APPENDIX F

NUHOMS®-24P Topical Report Revision 1

Intentionally Removed-Proprietary Information

APPENDIX G

DELETED

APPENDIX H

NUHOMS®-24P LONG CAVITY DSC EVALUATION FOR STORING PWR FUEL WITHOUT BURNABLE POISON ROD ASSEMBLIES (BPRAs)

H.1 General Description

H.1.1 Introduction

As stated in Section 1.2.1, the body of this SAR is dedicated to two canister types: the standard length NUHOMS®-24P and the NUHOMS®-52B. The purpose of this Appendix is to perform a safety evaluation of the long cavity NUHOMS®-24P DSC design for use with the standardized NUHOMS® system. The long cavity DSC is intended for use with fuel assemblies longer than those suitable for storage in the standard DSC, either because of longer fuel rods/end fittings or because of Burnable Poison Rod Assemblies (BPRAs) which extend beyond the normal assembly length. However, as described in this Appendix, the long cavity DSC is also qualified to store PWR fuel without BPRAs per NUHOMS Technical Specification 1.2.1. The thermal and radiological capacity of the long cavity DSC is similar to that of the standard length DSC described in the remainder of this SAR. The format of this Appendix follows that of Regulatory Guide 3.61, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask."

The long cavity DSC has an external length and diameter identical to those of the standard length PWR DSC. To obtain a longer cavity length, the standard carbon steel shield plugs used in both ends of the DSC are replaced with composite lead/steel (hereinafter referred to as "lead") shield plugs. The lead thickness has been selected to provide equivalent shielding while reducing the overall plug thickness. The basket for the long cavity DSC is identical to that of the standard 24P DSC except that the support rods have been extended by six inches.

The long cavity canister design has been previously approved by the NRC in the NUHOMS®-24P Topical Report [H.1, H.2] and under a site license at the Oconee Nuclear Station [H.3] for storage of B&W 15x15 fuel (the design basis for this SAR) with control components. A design similar to the long cavity DSC has also been previously approved by the NRC for use at the Calvert Cliffs Nuclear Power Plant.

Control components, as defined herein, are non-fuel hardware used to initiate, control, and monitor the chain reaction in the core which are used within assemblies but are not necessarily permanently attached. The control components are usually retired from service on a schedule different from that of the assemblies. Evaluation of the NUHOMS® 24P Long Cavity DSC for storage of PWR fuel with BPRAs is presented in Appendix J of this SAR. The balance of this Appendix provides a safety analysis for storing PWR fuel without BPRAs in accordance with Fuel Specification 1.2.1 of NUHOMS® COC.

As described in SAR Section 7.3.2.3 and below, the lead shield plug design provides equivalent shielding to the solid carbon steel shield plug design, and therefore is suitable to maintain doses ALARA in accordance with the design basis for the Standardized NUHOMS® system. A detailed discussion of the features of the lead shield plug design, a

comparative evaluation with the solid steel shield plug design, and a discussion of the licensing acceptability basis is provided in this Appendix.

As shown in this Appendix, the margins of safety demonstrated in the CSAR are not reduced by use of the long cavity DSC design. Operations and technical specifications are also unaffected. Therefore, the long cavity DSC described herein and in Appendix E is acceptable for use with the NUHOMS® system, and meets all applicable requirements of 10 CFR 72.

H.1.2 Description

With the exception of the shield plugs and support rods, the long cavity DSC is nearly identical to the standardized NUHOMS®-24P DSC. The external dimensions of the long cavity DSC are identical to those of the standard length DSC, as shown in Figure 4.2-1.

The top shield plug assembly is a stand-alone, cast lead-encased component, located between the support ring of the DSC shell and the inner top cover plate. The top shield plug is not welded or otherwise connected to the DSC shell or inner top cover plate. The casing is fabricated of welded carbon steel plates and the interior is cast lead. Both, the inner and outer top cover plates remain unchanged from the standard 24P design. The overall composite thickness of the plug is 5.0 inches

The bottom shield plug is a cast lead-encased component, structurally integrated with the outer bottom cover plate and the grapple ring assembly. The bottom shield plug is placed into the bottom end of the DSC during fabrication. This is similar to the process for installing the solid steel shield plug. The composite thickness of the bottom plug is 5.25 inches. The resulting cavity length of the DSC is a nominal 173 inches. The outer bottom cover plate is welded to the DSC shell and is part of the DSC pressure and containment boundary. The design of the top and bottom lead shield plug is shown in Figures H-1 and Figure H-2.

Both the top and bottom shield plugs designs incorporate radial stiffeners and a ring stiffener located at the center of the shield plug. Radial stiffeners connect the top and bottom plates of the shield plugs, provide shear transfer capability between the top and bottom plates, and allow the plug steel assembly to act as a composite section. The ring stiffener provides additional connection between the top and bottom plates and also serves to ease the connection of the radial stiffeners at the center of the shield plug.

The incorporation of lead shield plugs into the DSC results in some minor changes to certain DSC components. At the bottom end of the DSC, the inner bottom cover plate location is moved 2.75 inches closer to the bottom of the canister. At the top end of the DSC, the support ring, lifting lugs, and the machined step on the vent and siphon block are moved 3.25 inches closer to the top end of the canister. This reflects the thinner lead shield plugs, and results in an additional 6.00 inch long cavity, as shown in Figure 4.2-1.

The cover plates to DSC shell weld details are unaffected. Also, as shown in Figure 4.2-2, the support rod length is 6.00 inches longer because of the longer cavity.

H.2 Principal Design Criteria

The design criteria for the long cavity DSC are identical to those defined in Chapter 3 for the standard length NUHOMS®-24P DSC. These include the spent fuel to be stored, the environmental conditions and natural phenomena, the safety protection systems, and the decommissioning considerations. The design criterion for the lead shield plugs is to provide shielding equivalent to that of the steel shield plugs described in the remainder of this SAR.Structural Evaluation

H.3 Structural Evaluation

H.3.1.1 Long Cavity DSC Structural Evaluation

The structural evaluation for the long cavity DSC is based on the standard cavity DSC evaluations documented in Chapter 8 of this SAR, and additional evaluations as described in this Appendix. These additional evaluations address specific design differences between the long cavity DSC and the standard cavity DSC. These differences are limited to the design of top and bottom shield plug assemblies, and an increase of 6 inches in the length of the support rods of the basket assembly. The top shield plug assembly consists of a 5.00 inch thick carbon steel/lead composite compared with a 8.25 inch thick solid carbon steel plug in the standard design. The bottom shield plug consists of a 5.25 inch thick assembly, that is integral with the outer bottom cover plate. This is compared with a 8.00 inch thick assembly made up of the outer bottom cover plate and solid carbon steel plate for the standard DSC design.

A comparison of the weights for the lead shield plug canister and the solid steel shield plug canister is provided in Table H-1. As can be seen, the weight of the lead shield plugs is significantly less than that of the solid steel plugs, and more than offsets the increased basket weight due to the longer support rods. The long cavity DSC has no effect on the thermal analyses and the maximum internal pressure is bounded by the standard cavity DSC as described in Chapter 8 of this SAR. Therefore, the stress analyses described in Chapter 8 of this SAR bound the long cavity DSC design, with the exception of the shield plug assemblies which are addressed below.

The top and bottom shield plug assemblies are evaluated using material properties, loading combinations, and evaluation criteria as described in Chapter 3 for the standard 24P DSC. The top and bottom shield plug assemblies are evaluated for vertical deadweight, horizontal deadweight, handling, seismic, pressure, temperature, vertical bottom end drop, and horizontal side drop loads. In addition, thermal expansion analyses are performed to verify that the expansion of the lead material does not impose an unacceptable load condition on the shield plugs. The most critical load combinations for

each service level of the shell and the top inner and outer cover plates are bounded by the results in Section 8 of this SAR. Tables H-3 and H-4 summarize the results of the stress evaluation of the top shield plug and bottom shield plug assembly. The results for the inner and outer bottom cover plates are summarized in Table H-5.

The analyses of the standard DSC basket assembly documented in Chapter 8 of this SAR, incorporate the long cavity design parameters. Specifically, the basket assembly analyses are bounding evaluations which considered the increased length of the support rods in the long cavity design. Therefore, the analysis results described in Chapter 8 for the standard 24P basket design are bounding and thus, applicable to the 24P long cavity basket design.

The differential thermal expansion between the spacer disks and the DSC shell is addressed in Section 8.1.1.3.B and available gaps are deemed acceptable. Due to the similarities in materials of construction and the temperatures of the spacer disks and DSC shell, and based on the results of these analyses presented in Section 8.1.1.3.B, it is concluded that the long cavity DSC is also acceptable with respect to differential thermal expansion..

H.3.1.2 Long Cavity DSC Cavity Length

As described in Section 8.1.1.2, Paragraph C, adequate space exists in the cavity of the DSC between the spent fuel assemblies and the shield plug assemblies to allow for free thermal expansion. For storing PWR fuel without BPRAs in the long cavity DSC, the evaluation presented in SAR Section 8.1.1.3.B remains bounding.

H.4 Thermal Evaluation

Use of the long cavity DSC has no effect on the thermal analyses described in Section 8.1.3. When storing PWR fuel without BPRA, the canister heat load is unchanged. Also, since the increased cavity length provides a greater surface area for the dissipation of heat, temperatures in the long cavity DSC will be bounded by those currently reported in Tables 8.1-6 and 8.1-26.

H.5 Shielding Evaluation

The shielding effectiveness of the long cavity DSC shield plugs is evaluated below. Because the shield plug design for the long cavity DSC differs from that described in Chapter 7, an evaluation of the shield plug has been documented using measured data [H.5]. Table H-2 provides a summary of measured data from two fuel loads at Oconee into NUHOMS® canisters. The first canister shown, Oconee Canister No. 23, is a solid steel shield plug canister identical to that documented in the body of this SAR. The second canister shown, Oconee Canister No. 19, is a lead shield plug canister which is similar to that described in this Appendix.

As can be seen from examining Table H-2, the fuel radiation source is slightly higher in the lead shield plug canister, including the contribution from the irradiated control components. However, the table also shows that the resulting radiation doses on the top shield plug are essentially the same, and sometimes less, than the dose rates for the solid steel shield plug canister despite the higher fuel radiation source. This shows that the lead shield plug design is at least as effective as the solid steel shield plug design. Use of lead shield plugs in the DSC, therefore, will not result in any environmental impact or a significant increase in occupational exposure. To provide assurance that the lead shield plugs will provide adequate shielding during use (i.e. the thickness is within tolerance, the density is suitable, and no significant voids exist), all plugs are gamma scanned.

H.6 Criticality Evaluation

The criticality analysis is documented in Section 3.3.4.1 of this SAR for the NUHOMS®-24P design. As noted therein, the criticality analysis model is finite in the lateral direction and infinite in the axial direction, with the exception of the analysis of axial burnup variation which assumed water reflection at both axial ends. These assumptions are more conservative than modeling the actual canister ends. Since the canister ends do not affect the criticality analysis when storing PWR fuel without BPRAs, the use of the lead shield plug has no effect on the criticality safety of the design as demonstrated by the CSAR analyses.

H.7 Confinement

All confinement features of the long cavity DSC are identical to those of the standard length NUHOMS®-24P DSC described throughout this SAR.

H.8 Operating Procedures

Operating procedures for the long cavity DSC are identical to those provided in Chapter 5 for the standard 24P DSC.

H.9 Acceptance Criteria and Maintenance Program

All acceptance criteria and maintenance requirements for the long cavity DSC are identical to those of the standard DSC described throughout the body of this SAR.

H.10 Radiation Protection

As discussed in Section H.5, use of the long cavity DSC will not significantly affect dose rates during closure operations, handling, or storage. The ALARA and radiation protection discussions provided in Chapter 7 of this SAR are applicable to the long cavity DSC.

H.11 Accident Analyses

Because the long cavity DSC is functionally identical to the standard DSC, the type and likelihood of any postulated accident described in Section 8.2 is unchanged. The most severe accident for the DSC is the postulated cask drop. The structural evaluations provided in Section H.3 and Reference [H.1] demonstrate the adequacy of the long cavity DSC during the cask drop. Evaluations of other postulated accidents for the long cavity DSC, provided in detail in Reference [H.1], demonstrate its suitability for these conditions as well.

H.12 Operating Controls and Limits

The operating controls and limits specified in the Technical Specifications of the NUHOMS® Certificate of Compliance are applicable, without modification, to the long cavity DSC when storing PWR fuel without BPRAs.

H.13 Quality Assurance

The quality assurance program and requirements applicable to the long cavity DSC are identical to those defined for the standard length DSC in Chapter 11 of this SAR.

H.14 References

H.1 Pacific Nuclear Fuel Services, Inc., "Approved Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel - NUHOMS®-24P," NUH-002, Revision 2A, San Jose, California (Docket M-49).

H.2 U.S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safeguards, "Safety Evaluation Report Related to the Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel NUHOMS®-24P Submitted by NUTECH Engineers, Inc.," NUH-002, Revision 1A, April 1989.

H.3 Oconee Nuclear Station Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report (Docket No. 72-04).

H.4 U.S. Government, "Licensing Requirements for the Independent Storage of Spent Fuel and High-level Radioactive Waste," Title 10, Code of Federal Regulations, Part 72.

H.5 Duke Power Letter to Pacific Nuclear, Subject: "Oconee Nuclear Station Dry Canister Storage Loading No. 23," November 8, 1993.

Table H-1

Weight Comparison for Standard and Long Cavity Canister Designs

Component Description	Calculated Weight (Pounds) Standard Length	Calculated Weight (Pounds) Long Cavity
1. Dry Shielded Canister Shell Assembly	15,778	14,720
2. DSC Top Shield Plug Assembly	7,859	6,526
3. DSC Internal Basket Assembly	12,189	12,246
4. DSC Inner and Outer Top Cover Plates	1,934	1,934
5. 24 PWR Spent Fuel Assemblies	≤40,368[1]	≤40,368
6. Weight of Water in DSC Cavity	14,843	15,587
7. Total Wet DSC Loaded Weight (w/o top inner and outer cover plates)	91,037	89,447
8. Total Wet DSC Loaded Weight (w/ top inner and outer cover plates)	78,128	75,794

Note: Drawing changes may be referenced by more than one Safety Evaluation.

[1] For structural evaluation, weight of 24 PWR fuel assemblies was conservatively taken to be 1682 lbs

Table H-2

Comparison of Measured Dose Rates for Steel and Lead Shield Plugs(1)

Condition	Type	Steel Plug	Lead Plug
Measured Dose Rates at Centerline of Top Shield (mrem/hr)			
Standing Water Removed	Gamma	120	30
	Neutron	na(2)	<1
30 Gallons of Water Drained from DSC(1)	Gamma	180	80
	Neutron	na	1.5
DSC Closure Complete	Gamma	200	280
	Neutron	na	20
Fuel Information(3)			
Average Fuel Burnup (GWD/MTU)		28.6	29.7
Average Cooling Time (years)		10.9	10.2

Notes:

1. Reported lead plug dose rates are with 60 gallons of water removed.
2. Neutron dose rates not available at these locations.
3. Fuel batch information (i.e., burnup and cooling time with similar enrichment) indicates that fuel loaded in lead shield plug canister was somewhat more radioactive (higher source term) than that loaded into steel shield plug canister.

Table H-3
Long Cavity Top Shield Plug Analysis Results

Component	Controlling Load Combination	Service Level	Stress Type	Stress (ksi)	
				Calculated	Allowable(2)
Casing Plates	DD-2(3)	A/B	P_m	3.19	17.7
			$P_L + P_b$	3.44	26.6
			$P_L + P_b + Q$	21.3	53.1
	TR-11	D	P_m	39.4	40.6
	TR-10		$P_L + P_b$	55.3(1)	58.0
Stiffeners	DD-2(3)	A/B	$P_L + P_b$	7.3	26.6
			$P_L + P_b + Q$	25.2	53.1
	TR-10	D	$P_L + P_b$	56.9	58.0
Stiffener Welds	DD-2(3)	A/B	Weld Metal	2.1	21.0
			Base Metal	1.61	10.6
	TR-10	D	Weld Metal	16.3	42.0
			Base Metal	12.5	23.3

(1) Sum of global stress of 15.7 ksi plus local plate stress of 39.6 ksi
(2) All allowable stresses are taken at 600°F, unless noted otherwise.
(3) Uses an external pressure that is the maximum of the DD-2, HSM-9 and HSM-10 load combinations.
(4) Allowable stresses are taken at 500°F.

Table H-4
Long Cavity Bottom Shield Plug Analysis Results

Component	Controlling Load Combination	Service Level	Stress Type	Stress (ksi)	
				Calculated	Allowable[1]
Casing Plates	LD-1,LD-2	A/B	P_m	2.33	16.4
			$P_L + P_b$	2.72.	24.6
			$P_L + P_b + Q$	20.5	49.2
	UL-7[4]	C	P_m	4.94	19.7
			$P_L + P_b$	5.76	29.5
	TR-11	D	P_m	39.4	44.5
	TR-9		$P_L + P_b$	55.4[2]	63.0
Stiffeners	LD-1,LD-2	A/B	$P_L + P_b$	6.80	24.6
			$P_L + P_b + Q$	24.6	49.2
	UL-7[4]	C	$P_L + P_b$	14.4	29.5
	TR-9	D	$P_L + P_b$	37.8	63.0[3]
Stiffener Welds	LD-1,LD-2	A/B	Weld Metal	2.90	20.3
			Base Metal	2.90	7.28
	UL-7[4]	C	Weld Metal	6.16	30.5
			Base Metal	6.16	10.9
	TR-9	D	Weld Metal	15.4	40.6[3]
			Base Metal	15.4	15.5[3]

(1) All allowable stresses are taken at 600°F, unless noted otherwise.
(2) Sum of global stress of 15.1 ksi plus local plate stress of 40.3 ksi.
(3) Allowable stresses are taken at 500°F.
(4) Envelope of UL-7 and UL-8 loading conditions.

Table H-5
Long Cavity Bottom Cover Plate Analysis Results

Component	Controlling Load Combination	Service Level	Stress Type	Stress (ksi)	
				Calculated	Allowable[1]
Inner Bottom Cover Plate	TR-5	A/B	P_m	0.51	16.4
			$P_L + P_b$	17.20	24.6
			$P_L + P_b + Q$	45.2	49.2
	HSM-8a	C	P_m	0.01	19.7
	HSM-8a		$P_L + P_b$	0.30	29.5
	TR-11	D	P_m	39.4	44.5[4]
	TR-11		$P_L + P_b$	45.5	63.0[4]
Outer Bottom Cover Plate[2]	LD-1, LD-2	A/B	P_m	2.33	16.4
			$P_L + P_b$	2.72	24.6
			$P_L + P_b + Q$	20.5	49.2
	UL-7[3]	C	P_m	4.94	19.7
			$P_L + P_b$	5.76	29.5
	TR-11	D	P_m	39.4	44.5
	TR-11		$P_L + P_b$	45.5	63.0[4]

(1) All allowable stresses are taken at 600°F, unless noted otherwise.
(2) The Outer Bottom Cover Plate is integral with the Bottom Shield Plug.
(3) Envelope of UL-7 and UL-8 loading combination.
(4) Allowable stresses are taken at 500°F.

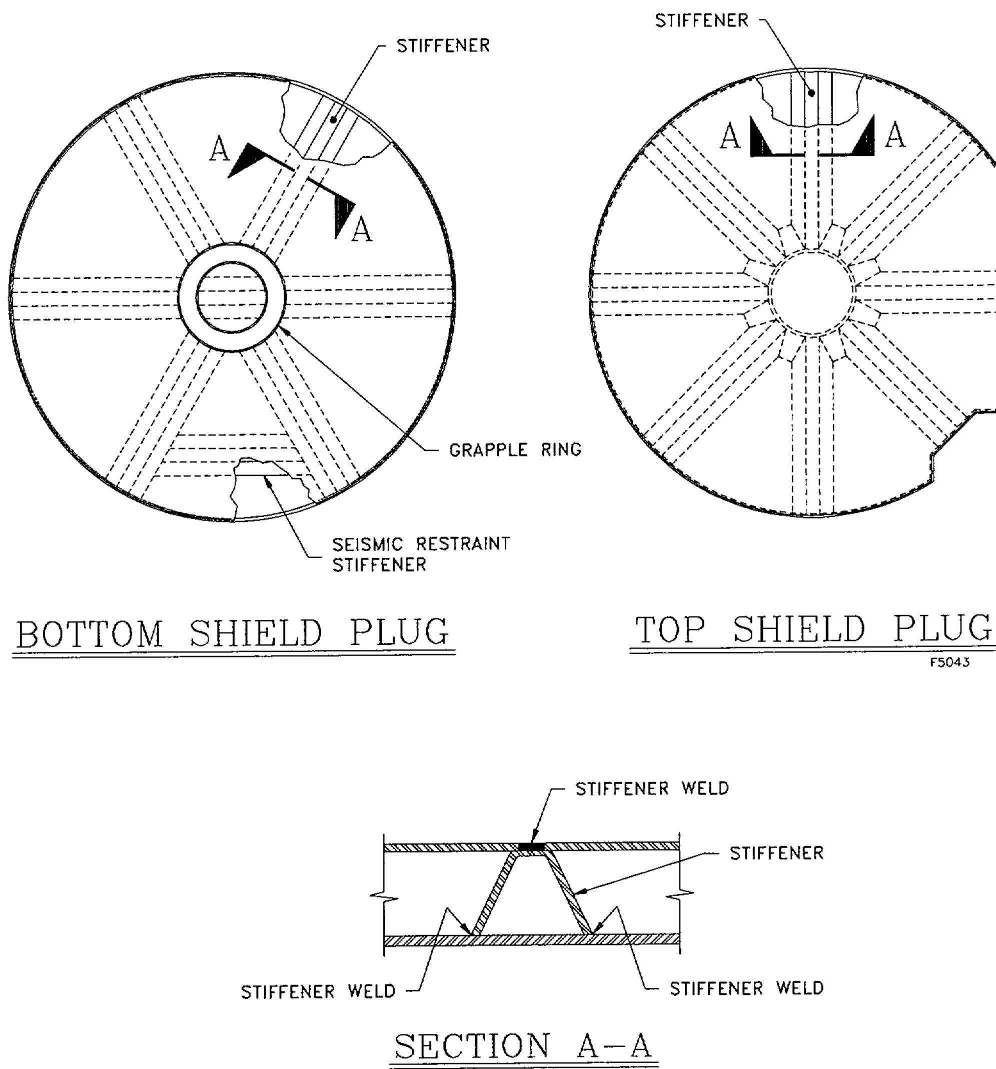


Figure H-1

Top and Bottom Lead Shield Plug Design

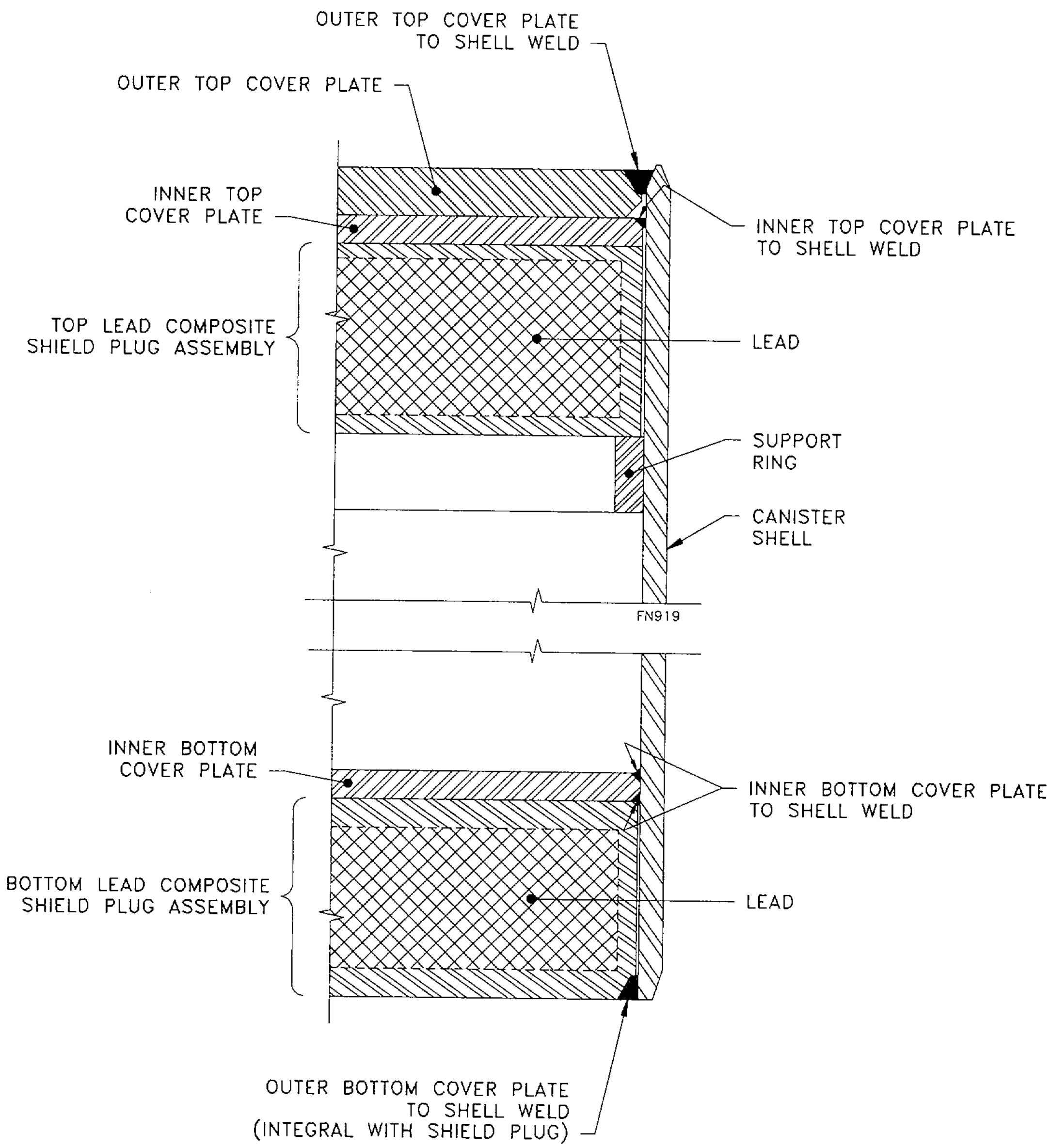


Figure H-2

Top and Bottom Lead Shield Plug Cross Section

APPENDIX I

DELETED

APPENDIX J

Evaluation of Burnable Poison Rod Assemblies (BPRAs) for the Standardized NUHOMS® -24P Long Cavity DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

J.1 General Description

J.1.1 Introduction

The purpose of this amendment application is to add intact Burnable Poison Rod Assemblies (BPRAs) and BPRAs with cladding failures for the B&W 15x15 and Westinghouse 17x17 fuel assembly types as authorized contents of the NUHOMS® 24P Long Cavity DSC. The evaluations in this Appendix address the impact of these BPRAs on the previously approved structural, thermal, shielding and criticality analyses.

J.1.2 General Description of the Storage Cask

The Cask description remains unchanged. Only the authorized contents of the Cask changes with this amendment request.

J.2 Principle Design Criteria

The design criteria remain unchanged. Only the authorized contents of the Cask changes with this amendment request.

J.3 Structural Evaluation

J.3.1 Discussion

The addition of BPRAs does not affect the structural evaluation for the Long Cavity DSC presented in Appendix H of the SAR. The maximum payload weight analyzed in the structural analysis of Appendix H and the various SAR chapters envelop the additional weight and center of gravity of the BPRAs. Section J.4 evaluates the effect of the addition of BPRAs on the maximum DSC internal pressure and temperatures. This evaluation confirms that the maximum DSC temperature during normal, off-normal and accident conditions with the addition of BPRAs remains bounded by the existing analysis. The structural evaluation for the long cavity DSC presented in Appendix H considers internal pressures that envelop those listed in Table J.4–2 and Table J.4–3 for consideration of BPRAs.

J.4 Thermal Evaluation

J.4.1 Discussion

The fuel qualification tables for the PWR fuel were adjusted for the inclusion of BPRAs, which generate a small amount of heat during transfer and storage in the NUHOMS® system. No recalculation of any thermal analysis was necessary to qualify the BPRAs as discussed in the following subsections. However, the pressure analysis is amended to analyze the BPRAs, which add gas during normal, off-normal, and accident conditions. The pressure analysis assumes intact BPRAs are loaded into the DSC. Failed BPRAs loaded in the DSC will have less available gas for release, therefore reducing the internal DSC pressures as compared to the case with intact BPRAs.

J.4.2 Summary of Thermal Properties of Materials

The only thermal property potentially affected by the addition of BPRAs is the effective conductivity of the fuel assemblies used in the thermal analysis. The effect of the addition of BPRAs will slightly increase the effective conductivity due to the addition of conduction paths and surface area available for radiation; however, no credit is taken for their presence.

J.4.3 Technical Specifications of Components

Refer to Section J.5 for the type of BPRA assemblies being considered. The BPRA assemblies generate a maximum 8 watts per BPRA of heat during transfer and storage operations in the NUHOMS® system, as calculated in Section J.5 of this appendix.

J.4.4 Thermal Evaluation for Normal and Off-Normal Conditions

Decay Heat

The addition of the BPRA components adds a small amount of decay heat to the fuel assemblies, which needs to be addressed. The maximum heat generation of the BPRA components is calculated to be 8 watts, as described in Section J.5.2. A new fuel qualification table (Table 10.3-5 of the SAR) has been added to address the addition of the heat generated by the BPRAs. The same methodology as presented in Chapter 8 of the SAR is used. The total decay heat of each assembly is taken to be that generated by the fuel plus the decay heat generated by the BPRAs. The criteria for fuel cladding temperature limit remains the same, but the allowable decay heat from the fueled rods in an assembly is reduced by 8 watts to accommodate the BPRAs. Therefore, the results from Chapter 8 of the SAR for normal and off-normal conditions remain valid for the maximum design basis decay heat of 1 kW per assembly, including the BPRA contribution.

Pressure Evaluation

The BPRAs generate Helium gas during reactor operation. Therefore, an evaluation of the impact on the existing DSC internal pressure calculations was performed. The B&W BPRA initially contains 22.3 lbs. of an aluminum oxide (Al_2O_3) composite. This composite contains 4 w/o Boron, of which 18.16 w/o is B-10. If, conservatively, 100% of the B-10 is assumed to react to generate Helium, this corresponds to the amount of Helium generation calculated below. There are 10 grams per mole of B-10 [J-8].

$$n_{B-10} = 22.3\,lbs \cdot 453.6\frac{g}{lbs} \cdot 0.04 \cdot 0.1816 \cdot \frac{1\,gmole}{10\,g} \cdot = 7.35\,gmoles$$

Conservatively, 30% of this Helium gas is assumed to be released into the BPRA rod void volume and is available for release into the DSC cavity in the case of BPRA rod rupture. Therefore, the total number of gas moles that are generated and released into the BPRA rod void volume is 24*7.35*0.3 = 52.9 gmoles. In addition, the BPRA rods are prepressurized with helium to one atmosphere. The void volume in each BPRA rod is 3.55 in^3. Assuming there are 24 BPRAs per DSC with 16 rods each, the total number of initial fill gas (Helium) moles is calculated below.

$$n_{He} = \frac{(14.7\,\mathrm{psia})(6894.8\,\mathrm{Pa/psi})(24 \cdot 16 \cdot 3.55\,\mathrm{in}^3)(1.6387\mathrm{x}10^{-5}\,\mathrm{m}^3/\mathrm{in}^3)}{(8.314\mathrm{J}/g\,\mathrm{mol}\cdot\mathrm{K})(293.15\mathrm{K})} \cdot \frac{kg/m \cdot s^2}{Pa} \cdot \frac{J}{kg \cdot m^2/s^2}$$

$$= 0.93\,\mathrm{g\,moles}$$

Therefore, the total number of gas moles per 24 B&W BPRAs is 52.9 + 0.93 = 53.8 gmoles.

For the Westinghouse 17x17 BPRA the total Helium released into the BPRA rodlet void volume is 2e-4 lb-moles, or 0.0907 gmoles per rodlet. The total Helium gas generated and released for 24 BPRA assemblies assuming the maximum 24 rodlets per BPRA assembly is 24*24*0.0907 = 52.2 gmoles. Thus, the B&W 15x15 BPRAs bound the WE 17x17 BPRAs for the DSC internal pressure calculations.

For normal and off-normal conditions, 1% and 10% release of the Helium gas from the BPRAs into the DSC cavity is assumed similar to the fuel assembly rods, as shown in Table J.4–1.

Table J.4–1 24P DSC BPRA Gas Quantities

Case	Percentage of BPRA Gas Released (%)	Moles of He Gas from BPRAs Released Per DSC
Normal	1	0.538
Off-Normal	10	5.38

Pressures are calculated for the long cavity DSC for normal and off-normal conditions for 40 and 45 GWd/MTU burnup fuel, which have total decay heat limits of 1.0 kW and 0.8 kW per fuel assembly including BPRAs respectively. The average gas temperatures are calculated according to the methodology described in Chapter 8 of the SAR. For the normal and off-normal cases, the 40 GWd/MTU burnup design basis fuel with 1 kW per assembly bounds the 45 GWd/MTU burnup fuel because the effect of the increased decay heat out weighs the effect of less fission gases. The following equations summarize the bounding DSC internal pressures calculated for normal and off-normal conditions. The partial pressures from the BPRAs are added to the results from Chapter 8 of the SAR.

$$P_{nor,40} = 21.7 + \frac{(1.4504\,x\,10^{-4}\,\frac{psia}{Pa})(0.538\,\text{mol})(8.314\,\text{J/mol}\bullet\text{K})(497°F + 459.67\,\text{R})}{(403{,}084\,\text{in}^3)(1.6387\,x\,10^{-5}\,\text{m}^3/\text{in}^3)(1.8\,\text{R/K})}$$
$$= 21.8\,psia\,(7.1\,psig)$$

$$P_{off-nor,40} = 24.8 + \frac{(1.4504\,x\,10^{-4}\,\frac{psia}{Pa})(5.38\,\text{mol})(8.314\,\text{J/mol}\bullet\text{K})(497°F + 459.67\,\text{R})}{(403{,}084\,\text{in}^3)(1.6387\,x\,10^{-5}\,\text{m}^3/\text{in}^3)(1.8\,\text{R/K})}$$
$$= 25.3\,psia\,(10.6\,psig)$$

Using similar methods, the maximum normal and off-normal condition DSC internal pressures with 45 GWd/MTU burnup fuel are 6.8 psig and 10.2 psig respectively.

The results are summarized in Table J.4–2 for the most limiting normal and off-normal conditions.

Table J.4–2 DSC Internal Pressure Summary for 24P DSC with BPRAs for Normal and Off-Normal Conditions.

Operating Condition	Limiting Case Description	40 GWd/MTU Burnup Pressure (psig)	45 GWd/MTU Burnup Pressure (psig)	Design Basis Pressure (psig)
Normal	DSC in Cask, 100°F	7.1	6.8	10
Off-Normal	DSC in Cask, 100°F	10.6	10.2	10

The pressures fall below the design basis DSC internal pressures under the worst case normal conditions. For Off-Normal conditions, the maximum calculated pressure of 10.6 psig exceeds the design pressure of 10.0 psig. This is acceptable based on paragraph NB-3223(a) of ASME Section III, Subsection NB.

J.4.5 Hypothetical Accident Thermal Evaluation

Decay Heat

The accident thermal analysis is not affected due to the addition of BPRAs. The allowable fuel assembly decay heats and cooling times are adjusted to accommodate the additional heat generation of the BPRAs.

Pressure Evaluation

The amount of Helium gas released to the DSC cavity from BPRAs during the accident condition assuming 100% release is 53.8 gmoles as calculated in the Section J.4.4. For accident conditions, 100% release of the Helium is assumed.

The effect on the DSC internal pressures are calculated for the long cavity DSC during accident conditions for 40 and 45 GWd/MTU burnup fuel. The average DSC cavity gas temperatures are calculated according to the methodology described in Chapter 8 of the SAR. The 45 GWd/MTU burnup fuel bounds the 40 GWd/MTU burnup fuel for the accident case because the effect of the increased number of moles of fission gases at higher burnups out weigh the effect of the lower allowable decay heats due to required

longer cooling times for higher burnup fuel. The following equation calculates the bounding DSC internal pressure for accident conditions. For the accident case, the limiting pressure is calculated at 45 GWd/MTU burnup fuel with 0.8 kW decay heat. The DSC cavity average gas temperature is calculated assuming a conservative 5 day blocked vent transient takes place during storage with 0.8 kW/assembly fuel using the methodology described in Chapter 8 of the SAR.

$$P_{acc,45} = \frac{(1.4504 \times 10^{-4} \frac{psia}{Pa})(53.8 + 586\,\text{moles})(8.314\,\text{J/mol}\bullet\text{K})(688.7 + 459.67\,\text{R})}{(403{,}084\,\text{in}^3)(1.6387 \times 10^{-5}\,\text{m}^3/\text{in}^3)(1.8\,\text{R/K})}$$
$$= 74.5\,\text{psia}\,(59.8\,\text{psig})$$

Using similar methods, the maximum accident condition DSC internal pressure with 40 GWd/MTU burnup fuel is 56.1 psig.

The resulting accident and design basis DSC internal pressures are compared in Table J.4–3.

Table J.4–3 DSC Internal Pressure for 24P DSC with BPRAs for Accident Conditions

Operating Condition	Limiting Case Description	40 GWd/MTU Burnup Pressure (psig)	45 GWd/MTU Burnup Pressure (psig)	Design Basis Pressure (psig)
Accident	Blocked HSM vents, 125°F	56.1	59.8	60

These pressures fall below the design basis pressure under the worst case accident conditions. Therefore, the DSC internal accident pressure with BPRAs is bounded by the previous analysis for the NUHOMS® system.

J.5 Shielding Evaluation

The radiation shielding evaluation for the Standardized NUHOMS® System (during loading, transfer and storage) with design basis PWR fuel is discussed in Section 3.3.5, 7.0 and 8.0 of the SAR. The following radiation shielding evaluation discussion specifically addresses the dose rate due to the design basis BPRAs for the B&W 15x15 and Westinghouse 17x17 class assembly types. This shielding evaluation is applicable to both intact and failed BPRAs loaded in the DSC because the shielding source terms are the same for intact and failed BPRAs.

Table J.5–1 lists the BPRA components considered in this application along with the assembly types in which each BPRA component is used.

Table J.5–1 BPRA Hardware Considered

Applicable Assembly Type	BPRA Component
B&W 15x15	B&W 15x15 Burnable Absorber Assembly
WE 17x17 Standard	WE 17x17 Pyrex Burnable Absorber 2 – 24 Rodlets
WE 17x17 OFA/Vantage 5	WE 17x17 WABA Burnable Absorber 3 – 24 Rodlets

J.5.1 Discussion and Results

The effect on dose rates due to the inclusion of 24 irradiated BPRAs into B&W 15x15 or Westinghouse 17x17 fuel assemblies, loaded into the Standardized NUHOMS® System are summarized in Table J.5–2 and Table J.5–3. It is expected that the dose rate on the shield plug surface during welding operations will increase due to the addition of BPRAs. However, by using proper ALARA, the total committed dose increase for one canister load will be small. Welding operations are performed remotely by use of the automated Welding Machine. In addition, temporary shielding can be used to keep dose rates ALARA. Increasing the required cooling time, as required, to assure that the HSM and transfer cask general area dose rates remain bounded by the SAR analysis mitigates the impact of the dose rates due to BPRAs. A discussion of the method used to determine the design basis BPRA source term is included in Section J.5.2. The method used to determine the increase in dose rates due to 24-design basis BPRAs is provided in Section J.5.4.

Table J.5–2 Dose Rate Due to the 24 Design Basis BPRAs

DOSE RATE LOCATION	BPRA Fuel Region (mrem/hr)	BPRA Plenum Region (mrem/hr)	BPRA Top Region (mrem/hr)	BPRA Total (mrem/hr)
HSM Roof (centerline)	7.2	NA	NA	7.2
HSM End Shield Wall Surface	1.1	NA	NA	1.1
HSM Door Exterior Surface (PWR) (centerline)	1.0	NA	NA	1.0
Center of HSM Door Opening (PWR) (w/out door)	53	NA	NA	53
HSM Back Shield Wall	0.001	0.010	0.035	0.046
Revised Centerline Top DSC Shield Plug Wet Welding	2	54	520	576(1)
Centerline Top DSC Cover Plate w/ 3"ns3+1"steel Dry Welding	1	4	14	19
Cask Surface (Radial) Contact Normal Condition	86	NA	NA	86
3 ft from Cask Surface (Radial) Normal Condition	43	NA	NA	43
Cask Surface (Radial) Contact Accident Condition	185	NA	NA	185
3 ft from Cask Surface (Radial) Accident Condition	87	NA	NA	87
15 ft from Cask Surface(Radial) Accident Condition	32	NA	NA	32
2000 ft from Cask Surface (Radial) Accident Condition	0.05	NA	NA	0.05
Cask Top Axial Surface	0.1	0.5	1.8	2.4
Cask Bottom Axial Surface PWR	1.7	NA	NA	1.7

(1) This high dose rate is based on conservative estimates of initial Cobalt content of the BPRA materials

Table J.5–3 Bounding Dose Rates for Fuel plus BPRAs NUHOMS®-24P System

Location	Neutron Dose Rate (mrem/hr.) Direct	Neutron Dose Rate (mrem/hr.) Reflected	Gamma Dose Rate (mrem/hr) Primary and Secondary Direct	Gamma Dose Rate (mrem/hr) Primary and Secondary Reflected	Total Dose Rate (mrem/hr.)
DSC in HSM					
1. HSM Surface(5)					
1.1 HSM Wall or Roof(5)	0.4	(1)	48.2	(1)	48.6
1.2 HSM Front Bird Screen(5)	0.8	15.7	75.2	234.2	325.9
1.3 HSM Roof Bird Screen(5)	0.8	35.2	103.6	427.8	567.4
1.4 Center of Door (exterior)(5)	28.2	(1)	23.2	(1)	62.3(4)
1.5 Center of Door Opening (door removed) (5)	966.1	(1)	1188.2	(1)	2329(4)
DSC IN CASK					
1. Centerline Top Shield Plug(2)	0.4	(1)	655.4	(1)	655.8
2. Top Cover Plate (cavity drained with water in annulus and 3 inches of temporary neutron and 1" of gamma shielding)					
2.1 Centerline(2)(6)	10.2	(1)	50.1	(1)	60.3
2.2 Outer Edge(3)(2)(6)	8.1	(1)	40.1	561.8	610.0

(1) The reflected dose at these locations is negligible.

(2) The DSC/cask annulus is filled with water and additional neutron shielding material is utilized as required. In addition, the DSC inner cavity is assumed to be filled with water for this operation. This high dose rate is based on conservative estimates of the initial Co60 content of the BPRA materials

(3) The same gap dose rate applies for cases where only the top shield plug is in place. The dose rates reported are with water in the DSC/cask annulus (however, no water is assumed to be in the DSC).

(4) Represents the greatest total dose rate calculated for fuel assemblies suitable for storage per Table 10.3-3, which bounds the dose rate due to fuel assemblies suitable for storage per Table 10.3-5 plus BPRAs.

(5) The cooling time for fuel assemblies suitable for storage is increased to account for the additional dose rate due to BPRAs. Therefore, the total dose rate remains bounded by 5 yr cooled fuel and/or Table 10.3-3 fuel

(6) Conservatively Bounded by assuming 5 year cooled design basis fuel plus BPRAs.

Table J.5-3 Bounding Dose Rates for Fuel plus BPRAs NUHOMS®-24P System (Concluded)

Location	Neutron Dose Rate (mrem/hr.)		Gamma Dose Rate (mrem/hr) Primary and Secondary		Total Dose Rate (mrem/hr.)
	Direct	Reflected	Direct	Reflected	
DSC IN CASK (continued)					
3. Transfer Cask					
3.1 Radial					
a. Surface(5)	163.9	(1)	427.9	(1)	591.8
b. 3 Ft. from Surface(5)	75.4	(1)	211.8	(1)	287.2
3.2 Top axial(6)	16.8	(1)	6.8	(1)	33.3(4)
3.3 Bottom axial(6)	28.3	(1)	40.8	(1)	69.8(4)

J.5.2 Source Specification

The design basis fuel radiation source strength used for the Standardized NUHOMS® System shielding calculations are described in Section 7.2 of the SAR. The addition of 24 irradiated BPRAs will add to the gamma source strength and therefore increase the gamma dose rates. BPRAs do not radiate neutrons, therefore, no increase in the neutron dose rates will occur. The BPRA design basis gamma source strength used in the BPRA evaluation is presented in Section J.5.2.1.3.

J.5.2.1 Gamma Source

J.5.2.1.1 BPRA Data

The important properties of the BPRAs for calculating a design basis source term are the material weights, locations, length of time in core (number of cycles of use) and suitable flux scaling factors. This data is primarily obtained from the OCRWM characteristics database [J-1]. The data used in this evaluation is shown in Table J.5–4 for each BPRA. Table J.5–5 lists the flux scaling factors used in this calculation which are taken from Reference [J-1].

Table J.5–4 Burnable Poison Rod Assembly Weight Data

Component Name: B&W 15 X 15 Burnable Absorber Assembly
Burnup: 2 cycle, 5 year cooled

Region	SS304 (Kgs)	Inconel-750 (Kgs)	Poison (Kgs)	Zr-4 (Kgs)
Top	3.602	0.058(1)	0	0
Plenum	1.068(1)	0	0.724(1)	1.197(1)
Core	2.468(1)	0	9.146(1)	11.98(1)

Component Name: WE 17 X 17 Pyrex Burnable Absorber, 2 - 24 Rodlets (Worst Case)
Burnup: 2 cycle, 10 year cooled

Region	SS304 (Kgs)	Inconel 718 (Kgs)	Poison (Kgs)	Zr-4 (Kgs)
Top	2.62	0.42	0	0
Plenum	2.85	0	0	0
Core	11.9(1)	0	5.08	0

Component Name: WE 17 X 17 WABA Burnable Absorber, 3 - 24 Rodlets (Worst Case)
Burnup: 2 cycle, 10 year cooled

Region	SS304 (Kgs)	Inconel 718 (Kgs)	Poison (Kgs)	Zr-4 (Kgs)
Top	2.95	0	0	0
Plenum	2.76	0	0	2.61
Core	0	0	2.5	14.8

(1) Bounding Data Assumed

Table J.5–5 Flux Scaling Factors for each Region

Region	Flux Factor
Above Top	0.01
Top Nozzle	0.10
Gas Plenum	0.20
In-Core	1.00

J.5.2.1.2 ORIGEN2 Calculation

The ORIGEN2 [J-2] code is used to calculate the source terms for the BPRAs. Two ORIGEN2 input files were created to calculate the BPRA source terms. The first model is for the B&W 15x15 assembly type with B&W BPRA materials and the second for the WE 17x17 assembly type with WE BPRA materials. The first ORIGEN2 model assumes an in-core composition for fuel of 475-kg uranium, 117.72-kg zircalloy and 4.9-kg of inconel-718, consistent with previous standard NUHOMS® source term calculations. The input file also includes 1 kg of SS304 and Inconel X-750 in the Top Region; 1 kg of SS304, Poison Material, and Zr-4 in the Plenum Region; and 1 kg of SS304, Poison Material and Zr-4 in the core region. The second ORIGEN2 model assumes an in-core composition for fuel of 461-kg uranium, 111.15-kg zircalloy, 0.54-kg Stainless Steel and 4.9-kg of inconel-718. The input file also includes 1 kg of SS304 and Inconel 718 in the Top Region; 1 kg of SS304, Poison Material, and Zr-4 in the Plenum Region; and 1 kg of SS304, Poison Material and Zr-4 in the core region.

The in-core regions in both cases were irradiated for two (2) cycles. The fuel is burned to 22,000 MWd/MTU during the first cycle and an additional 14,000 MWd/MTU during the second cycle for a total burnup of 36,000 MWd/MTU. The ORIGEN2 flux calculated for each irradiation step is then scaled by the factors listed in Table J.5–5 and applied to the BPRA materials in each region. The outputs from ORIGEN2 are source terms per kg of each BPRA material by region. Therefore, to determine the source term for a specific BPRA one must multiply the appropriate ORIGEN2 output source terms by the number of kgs of each material in a specific region and sum the results. In Section J.5.2.1.3, the source terms for each BPRA are tabulated. Material compositions and impurities are taken directly from Reference [J-3] with the exception of the poison material, which is modeled as B_4C.

J.5.2.1.3 BPRA Gamma Source Term Results

The calculated source terms for each BPRA component are summarized below. Based on a review of the heat generation and region-wise gamma source terms the design basis BPRA is:

- The WE 17x17 Pyrex Burnable Absorber with 10 year cooling for heat generation and dose rates dominated by BPRA core region (DSC side and bottom), and

- The B&W 15x15 Burnable Poison Assembly with 5 year cooling for dose rates dominated by the BPRA top and plenum regions (DSC top).

Table J.5–6 and Table J.5–7 summarize the gamma source term by region and energy group and the total heat generation for the two bounding BPRA designs.

Table J.5–6 B&W 15 X 15 Burnable Absorber Assembly

Burnup: 2 cycles, 5 year cooled

Region	Top	Plenum	Core
Mean Energy (MeV)	γ/sec/BPRA	γ/sec/BPRA	γ/sec/BPRA
1.00E-02	6.581E+10	4.430E+10	4.694E+11
2.50E-02	1.104E+10	7.287E+10	7.336E+11
3.75E-02	6.292E+09	1.998E+10	2.024E+11
5.75E-02	7.090E+09	4.457E+09	4.734E+10
8.50E-02	2.788E+09	1.780E+09	1.888E+10
1.25E-01	1.071E+09	1.064E+09	1.106E+10
2.25E-01	3.521E+08	5.952E+09	5.971E+10
3.75E-01	9.879E+07	3.424E+10	3.427E+11
5.75E-01	5.681E+06	4.394E+10	4.398E+11
8.50E-01	1.110E+10	6.678E+09	7.624E+10
1.25E+00	2.403E+12	1.329E+12	1.422E+13
1.75E+00	1.478E+02	1.360E+02	1.153E+03
2.25E+00	1.274E+07	7.040E+06	7.535E+07
2.75E+00	3.942E+04	2.179E+04	2.332E+05
3.50E+00	3.947E-15	6.520E-14	6.526E-13
5.00E+00	0.000E+00	0.000E+00	0.000E+00
7.00E+00	0.000E+00	0.000E+00	0.000E+00
9.50E+00	0.000E+00	0.000E+00	0.000E+00
Total	2.509E+12	1.564E+12	1.662E+13
Total Heat Gen.:	3.9 watts per BPRA		

Table J.5–7 WE 17 X 17 Pyrex Burnable Absorber, 2 - 24 Rodlets (Worst Case)

Burnup: 2 cycles, 10 year cooled

Region	Top	Plenum	Core
Mean Energy (MeV)	γ/sec/BPRA	γ/sec/BPRA	γ/sec/BPRA
1.00E-02	4.189E+10	4.654E+10	9.043E+11
2.50E-02	7.161E+09	7.952E+09	1.545E+11
3.75E-02	4.073E+09	4.523E+09	8.783E+10
5.75E-02	4.587E+09	5.093E+09	9.889E+10
8.50E-02	1.803E+09	2.002E+09	3.888E+10
1.25E-01	6.926E+08	7.686E+08	1.493E+10
2.25E-01	2.279E+08	2.528E+08	4.909E+09
3.75E-01	6.388E+07	7.091E+07	1.377E+09
5.75E-01	3.667E+06	4.073E+06	7.909E+07
8.50E-01	5.386E+08	3.668E+08	7.507E+09
1.25E+00	1.554E+12	1.726E+12	3.351E+13
1.75E+00	8.097E-05	4.215E-06	1.413E-04
2.25E+00	8.236E+06	9.146E+06	1.775E+08
2.75E+00	2.548E+04	2.830E+04	5.495E+05
3.50E+00	7.871E-15	2.745E-21	7.266E-18
5.00E+00	0.000E+00	0.000E+00	0.000E+00
7.00E+00	0.000E+00	0.000E+00	0.000E+00
9.50E+00	0.000E+00	0.000E+00	0.000E+00
Total	1.615E+12	1.793E+12	3.482E+13
Total Heat Gen.:	7.7 watts per BPRA		

J.5.2.2 Neutron Source Term

BPRAs do not radiate neutrons. Therefore, there is no neutron dose rate contribution from the BPRAs.

J.5.3 Model Specification

The insertion of BPRAs does not affect the axial and radial shielding configurations of the Standardized NUHOMS® System.

J.5.4 Shielding Evaluation

Dose rate contributions from the Core, Plenum and Top regions, as appropriate, from 24 BPRAs are calculated at each location on the Standardized NUHOMS® HSM, DSC and Transfer Cask.

The radiation shielding evaluation for the Standardized NUHOMS® HSM, DSC and Transfer Cask with design basis PWR fuel is summarized in Section 7.0 of the SAR. The following shielding evaluation discussion specifically addresses the loading of the design basis BPRA components as determined in Section J.5.2.

The analysis presented in Section 7.0 of the SAR accounts for the neutron and gamma ray dose rate contributions from a DSC loaded with design basis fuel assemblies. In this evaluation, the dose rate due to 24-design basis BPRAs loaded in the DSC is calculated. The calculations are performed by replacing the source terms in the SAR shielding model with the design basis BPRA source terms. The exact models used in the SAR shielding evaluation are duplicated in this evaluation with the exception of the model for the Centerline Top DSC Shield Plug during wet welding, which are modified as follows:

- the DSC cavity was assumed to be completely filled with water, and
- the 0.75-inch DSC top inner cover plate is removed.

This was done to account for a clarification to Specification 10.3.6 which requires the user to measure the dose rate at the top of the shield plug when the cask is full of water prior to welding.

The assumptions, technical approach and flux-to-dose rate conversion factors, etc. are identical to those documented in Section 7.0 of the SAR. Table J.5–2 presents the results of the dose rate calculations for the 24-design basis BPRAs. Table J.5–3 presents the bounding results using the same locations as in SAR Table 7.3-2.

J.6 Criticality Evaluation

The criticality analysis is documented in Section 3.3.4.1 of this SAR for the NUHOMS®-24P design. As noted therein, the criticality analysis model is finite in the lateral direction and infinite in the axial direction, with the exception of the analysis of axial burnup variation which assumed water reflection at both axial ends. Because BPRAs can displace borated moderator in the assembly guide tubes, an evaluation has been performed to determine the potential impact of BPRA storage on the system reactivity. KENO-VI (CSAS26 of SCALE 4.4)[J-4] is used to demonstrate that the B&W 15x15, Westinghouse 17 x 17 Standard, and OFA/Vantage 5 fuel designs with BPRAs are bounded by the B&W 15x15 fuel design without BPRAs, for storage in the standardized NUHOMS-24P system. No credit was taken for BPRA cladding and absorbers, rather the BPRA is modeled as $^{11}B_4C$ in the entire guide tube. Thus, the highly borated moderator between the guide tube and the BPRA rodlet is modeled as $^{11}B_4C$. The inclusion of more Boron-11 and carbon enhances neutron scattering causing the neutron population in the fuel assembly to be slightly increased which increases reactivity.

J.6.1 Discussion and Results

The results demonstrate that when BPRAs are added to the NUHOMS System payload, the reactivity effect is negative with a moderator density less than 0.85 g/cc. This is demonstrated by modeling the guide and instrument tube volumes as filled with fully depleted boron carbide, $^{11}B_4C$, and showing that k_∞ decreased with respect to the case of borated moderator filled tube volumes. The B&W 15x15, Westinghouse 17x17 Standard and OFA/Vantage 5 fuel types show a small positive change in k_∞ at 1 g/cc and 0.9 g/cc, however, the change is small. The positive $\Delta k_\infty/k_\infty$ is only seen at the higher, much less reactive borated moderator density, not at the more reactive lower densities. The positive reactivity change seen at moderator densities of 1 g/cc and 0.9 g/cc for these fuel types, does not result in a payload more reactive than when at the optimum moderator density (0.2 –0.3 g/cc). To demonstrate that the overall system reactivity is below 0.95 for moderator densities greater than 0.85, a set of 2-D, infinite height, DSC models are evaluated. In these calculational models, the actual DSC geometry (2-D) in the radial direction is modeled. For each of the three fuel designs, simulations at three moderator densities (0.9982, 0.90 and 0.85 g/cc) were performed to ensure that the resulting k_{eff} does not exceed 0.95 for the worst case conditions, including the effects of the BPRA. These additional calculations demonstrate that the maximum final k_{eff} (0.9203) in the moderator density in the range of 0.85 to 1.0, meets the acceptance criteria of less than 0.95. Thus, the value of k_{eff} is within acceptable bounds for the entire moderator density range.

J.6.2 Package Fuel Loading

The only change to the package fuel loading is the addition of BPRAs that are modeled as $^{11}B_4C$. The fuel assembly dimensions and layout used in this evaluation are given in Table J.6–1 through Table J.6–3 and Figure J.6-1 through Figure J.6-3.

Table J.6–1 B&W 15x15 Fuel Assembly

Fuel Assembly Design:	**B&W 15x15**		
Parameter	Value		Reference
Maximum number of fuel rods	208		[J-5]
Fuel density, % theoretical	95%		[J-5]
Quantity of guide tubes	16		[J-5]
Quantity of instrument tubes	1		[J-5]
Parameter	inches	cm	Reference
Pellet diameter	0.3686	0.9362	[J-5]
Active fuel length	141.8	360.1720	[J-5]
Cladding thickness	0.0265	0.0673	[J-5]
Fuel rod OD	0.43	1.0922	[J-5]
Fuel rod pitch	0.568	1.4427	[J-5]
Guide tube OD	0.53	1.3462	[J-6]
Guide tube thickness	0.016	0.0406	[J-6]
Instrument tube OD	0.493	1.2522	[J-6]
Instrument tube thickness	0.026	0.0660	[J-6]

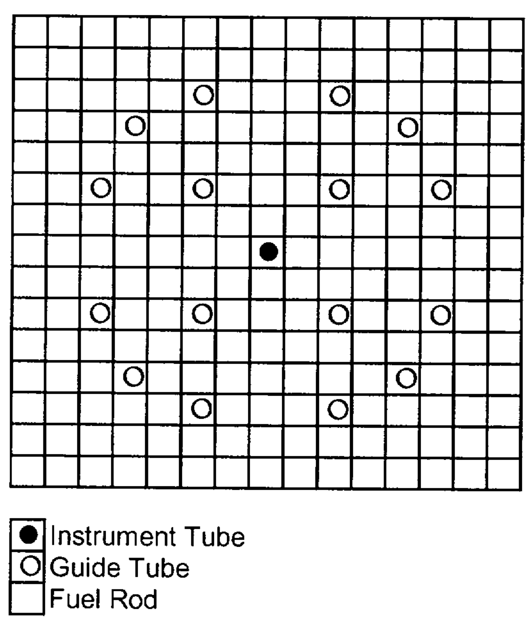


Figure J.6-1 B&W 15x15 Fuel Assembly

Table J.6–2 Westinghouse 17x17 Standard Fuel Assembly

Fuel Assembly Design:	**Westinghouse 17x17 Standard Fuel Assembly**		
Parameter	Value		Reference
Maximum number of fuel rods	264		[J-5]
Fuel density, % theoretical	95%		[J-5]
Quantity of guide tubes	24		[J-7]
Quantity of instrument tubes	1		[J-7]
Parameter	inches	cm	Reference
Pellet diameter	0.3225	0.8192	[J-5]
Active fuel length	144	365.7600	[J-5]
Cladding thickness	0.0225	0.0572	[J-5]
Fuel rod OD	0.374	0.9500	[J-5]
Fuel rod pitch	0.496	1.2598	[J-5]
Guide tube OD	0.474	1.2040	[J-7]
Guide tube thickness	0.016	0.0406	[J-7]
Instrument tube OD	0.48	1.2192	[J-7]
Instrument tube thickness	0.015	0.0381	[J-6]

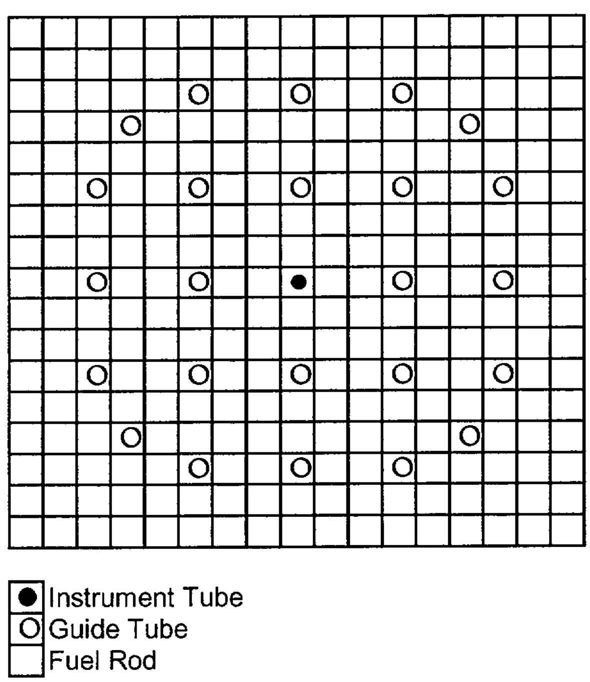


Figure J.6-2 Westinghouse 17x17 Standard Assemblies

Table J.6–3 Westinghouse 17x17 OFA/Vantage 5 Fuel Assembly

Fuel Assembly Design:	**Westinghouse 17x17 OFA/Vantage 5 Fuel Assembly**		
Parameter	Value		Reference
Maximum number of fuel rods	264		[J-5]
Fuel density, % theoretical	95%		[J-5]
Quantity of guide tubes	24		[J-7]
Quantity of instrument tubes	1		[J-7]
Parameter	inches	cm	Reference
Pellet diameter	0.3088	0.7844	[J-5]
Active fuel length	144	365.7600	[J-5]
Cladding thickness	0.0225	0.0572	[J-5]
Fuel rod OD	0.36	0.9144	[J-5]
Fuel rod pitch	0.496	1.2598	[J-5]
Guide tube OD	0.482	1.2243	[J-6]
Guide tube thickness	0.016	0.0406	[J-6]
Instrument tube OD	0.476	1.2090	[J-7]
Instrument tube thickness	0.015	0.0381	[J-6]

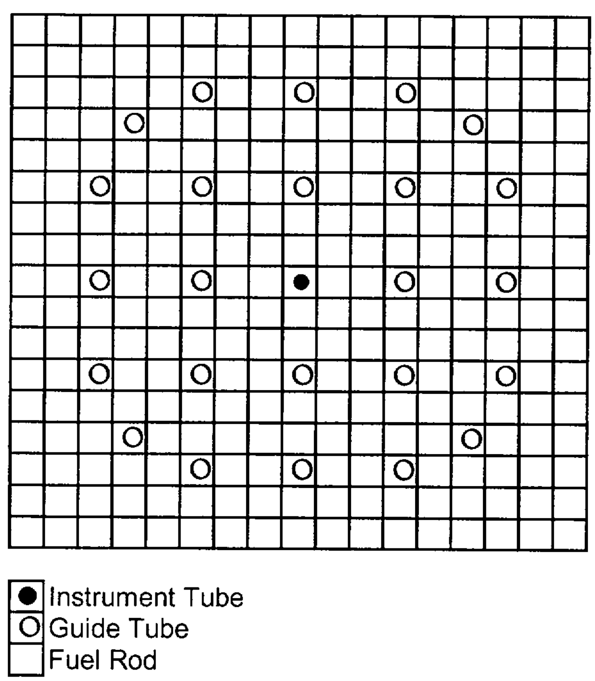


Figure J.6-3 Westinghouse 17x17 OFA/Vantage 5 Fuel Assembly

J.6.3 Model Specification

The KENO-VI monte-carlo computer code was used to:

1) Compare the reactivity of an infinite array of fuel assemblies with moderator and with BPRAs in the guide and instrument tubes. These calculations show that fuel with BPRAs is less reactive than fuel without BPRAs with moderator densities up to 0.85 g/cm^3.

2) Determine the maximum final k_{eff} for fuel with BPRAs with moderator densities greater than 0.85 g/cm^3.

For step 1, infinite array modeling techniques were used to represent the fuel in the basket. In all cases, the fuel was modeled as an array of fuel rods, guide tubes, and instrument tube surrounded by a NUHOMS®-24P guide sleeve. Each of these fuel assembly components is modeled discretely according to the dimensions listed in Table J.6–1 through Table J.6–3. The assembly and guide sleeve are immersed in borated moderator. The total width of the model is 10.28 inches. A specular albedo is specified on all four sides of the model. This approach results in high computed system reactivity due to neglected leakage effects but it will not bias the conclusions of the study.

The active fuel length was set to 141.8", however, specular albedo conditions are specified on the top and bottom. Fuel assembly grid straps and other related hardware are not included in the models. Partially inserted BPRAs are not modeled. This condition will not affect the outcome of the calculation since the change in system reactivity is governed by the fueled region and dominated by a fueled height approximately equal to the diameter of the fueled region.

The inside volume of the guide tubes and instrument tubes were specified as $^{11}B_4C$ for the models of the assemblies with BPRAs and borated water for those models without BPRAs. The dimensional data for the DSC used in the CSAS26 (KENO-VI) models is given in Table J.6–4.

Table J.6–4 Infinite Array DSC Dimensional Data

	inches	cm
Nominal guide sleeve inside dimension	8.9	22.606
Nominal guide sleeve thickness	0.1046	0.265684
Nominal guide sleeve outside dimension	9.1092	23.13737
Nominal guide sleeve pitch	10.28	26.1112

For Step 2, calculational models are generated for the actual DSC geometry (2-D, infinite height). For each of the three fuel designs, simulations at three moderator densities

(0.9982, 0.90 and 0.85 g/cc) were performed to ensure that the resulting k_{eff} does not exceed 0.95 including the effects of the BPRA.

For each run, the model (see Figure J.6-4) includes the 24 fuel assemblies within the guide sleeves, the four axial support rods, and the steel DSC shell with inner radius of 83.74 cm and outer radius of 85.33 cm. The fuel is modeled consistent with the nominal cases described in Section 3.3.4 of the SAR. The DSC is flooded with borated water and an additional 12 inches of borated water outside the DSC is added for increased reflection. For conservatism, boundary conditions were infinite water reflectors in the x and y directions and the fuel assembly was made infinitely long by adding specular reflection boundary conditions in the +z and –z directions.

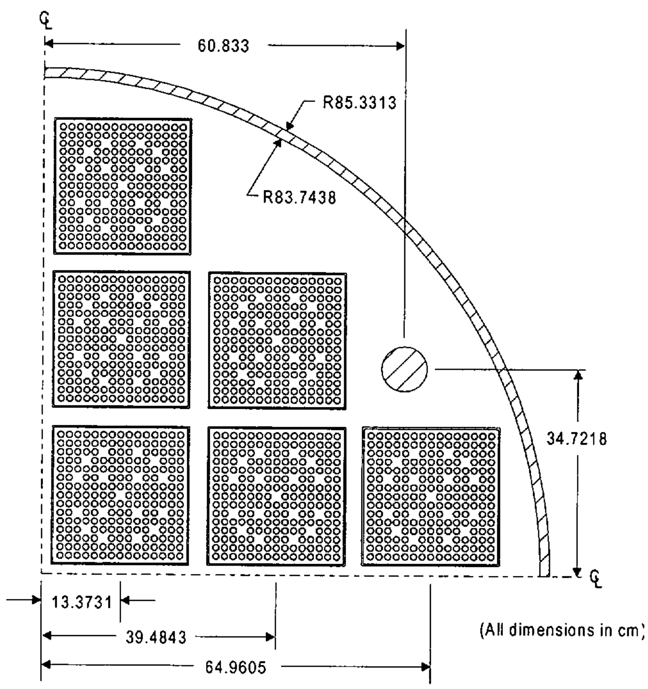


Figure J.6-4 DSC Geometry

J.6.4 Criticality Calculation

Figure J.6-5 through Figure J.6-7 show $\Delta k_\infty / k_\infty$ as a function of moderator density from 0.1 g/cc to 1.0 g/cc for the B&W 15x15, Westinghouse 17x17 Standard, and OFA/Vantage 5 fuel types, respectively.

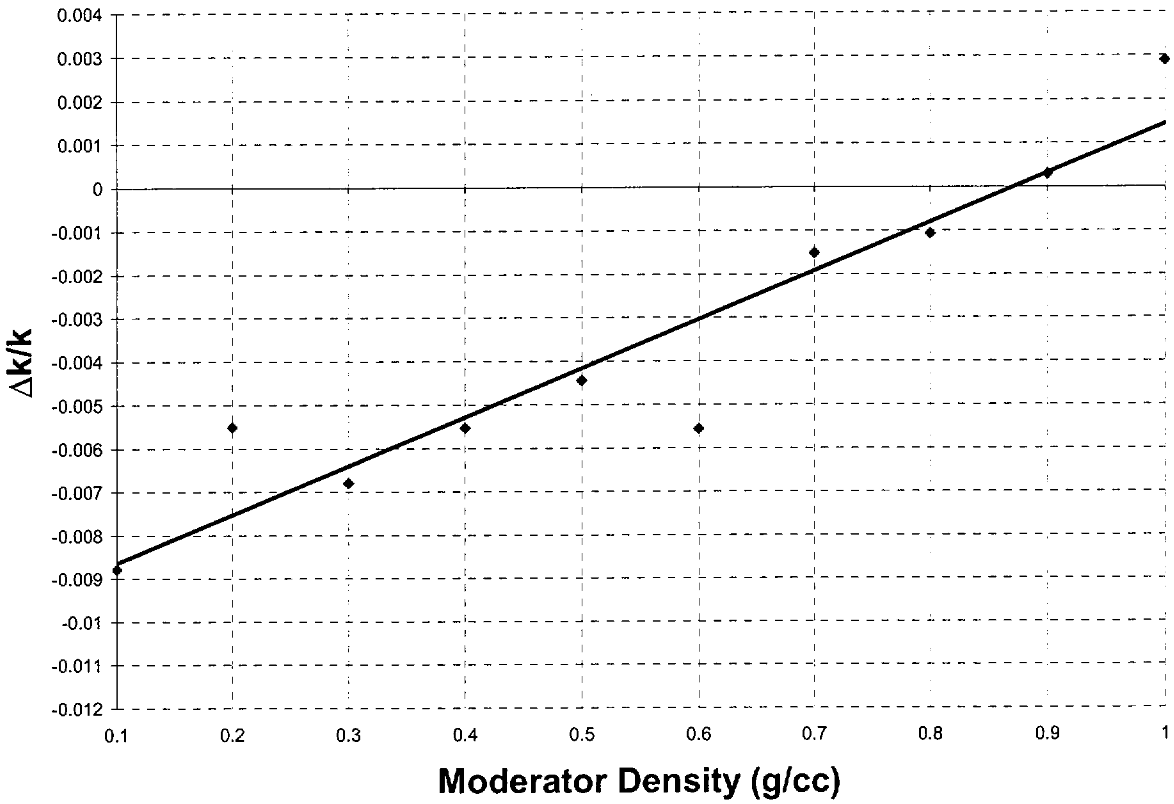


Figure J.6-5 $\Delta k_\infty / k_\infty$ as a Function of Moderator Density – B&W 15x15

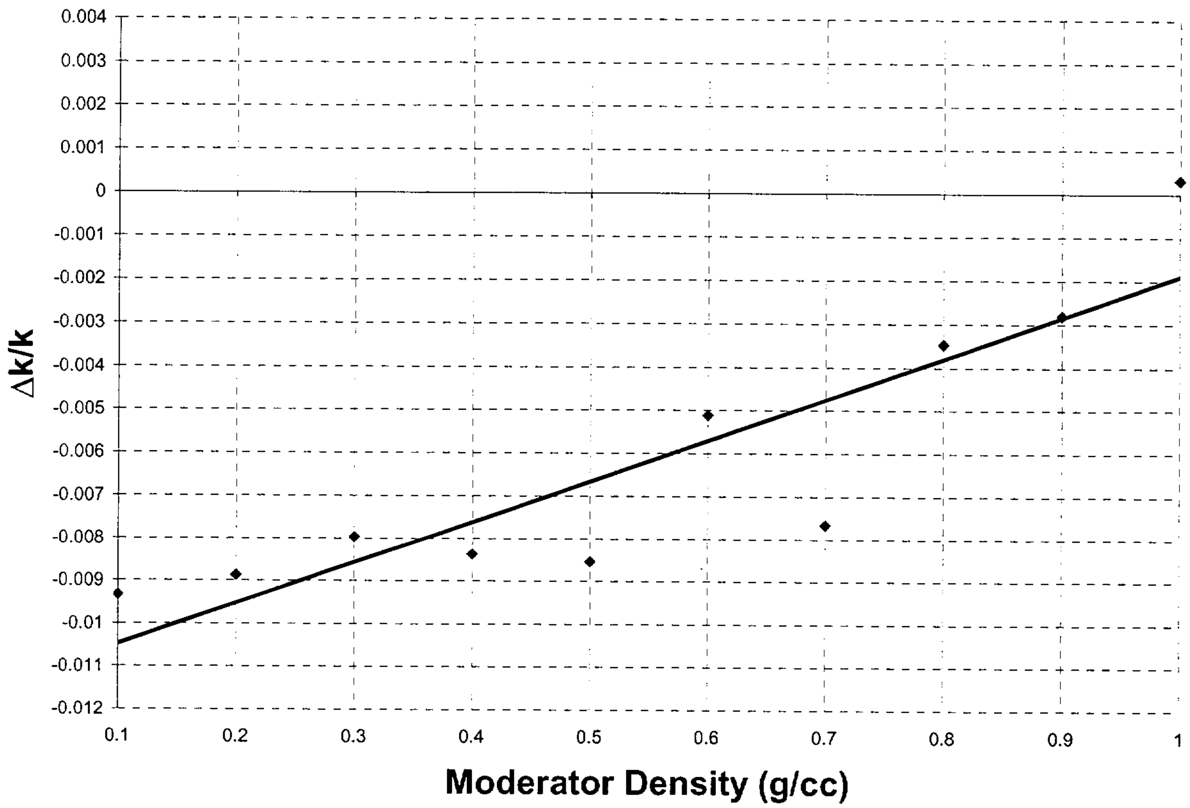


Figure J.6-6 $\Delta k_\infty / k_\infty$ as a Function of Moderator Density – Westinghouse 17x17 Standard

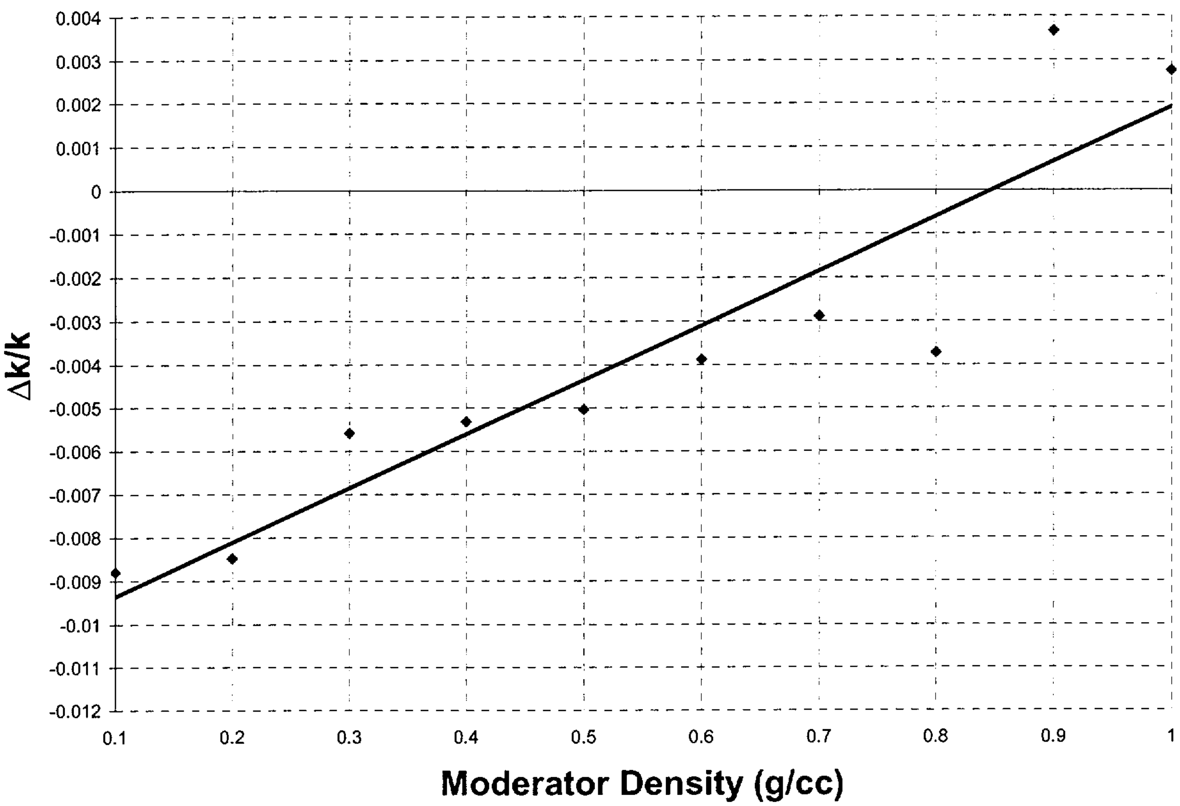


Figure J.6-7 $\Delta k_\infty / k_\infty$ as a Function of Moderator Density – Westinghouse 17 x17 OFA/Vantage 5

Table J.6–5 provides the results of the DSC nominal case analysis for moderator densities greater than 0.85 g/cm^3. The maximum k_{eff}, including all uncertainties is calculated using the methodology described in Section 3.3.4 of the SAR. The "nominal" k_{eff} values from the KENO-VI runs are then adjusted for worst case conditions as discussed in Section 3.3.G of the SAR.

Thus,

$$k_{eff} = k_{\text{-nominal}} + B_{\text{-method}} + B_{\text{-axial}} + B_{\text{-mod}} + B_{\text{-ref}} +$$

$$[(ks_{\text{-nominal}})^2 + (ks_{\text{-method}})^2 + (ks_{\text{-axial}})^2 + (ks_{\text{-mechanical}})^2 +$$

$$(ks_{\text{-reflector}})^2 + (ks_{\text{-burnup}})^2 + (ks_{\text{-mod}})^2]^{1/2}$$

where B_{-XX} represents the bias in the calculated k_{eff} value due to the given parameter "XX"(where "XX" represents the mathematical model/method, non-uniform axial burnup, worst case moderator density, and worst case reflector positioning, respectively). Similarly, ks_{-XX} represents, the 95/95 uncertainty in the k_{eff} value resulting from the uncertainty in the given parameter "XX", (with "XX" representing the mathematical model/method, non-uniform axial burnup, material mechanical construction/positioning tolerances, the burnup/enrichment-equivalence method, and moderator bias, respectively).

These adjustments to $k_{-nominal}$ can be calculated for the DSC with 24 PWR assemblies. For this analysis, $(B_{-method} + ks_{-method}) = (0.95 - USL + 2\sigma_{KENO})$, because both biases and uncertainties due to method are already included in the limiting k_{eff}, that is, the USL calculated in USLSTATS [J-9] (i.e. it has already been subtracted from the Upper Safety Limit of $k_{eff} = 0.95$) and the $2\sigma_{KENO}$ is the uncertainty from the current KENO-VI run. B_{-axial} and B_{-mod} are also 0.0 because the fuel is assumed to be fresh (no burnup credit) and the effect of moderator density bias is being explicitly modeled. B_{-ref} is likewise 0.0 because conservative reflector assumptions in all directions are explicitly included in these models.

This leaves only $ks_{-mechanical}$ to be considered. There are several mechanical tolerances that contribute to this 95/95 k_{eff} uncertainty, namely the DSC-guidesleeve center-center spacing tolerance, the fuel assembly position uncertainty within the guidesleeves, the manufacturing tolerances in the guide sleeve thickness, and the axial sleeve straightness uncertainty (cell bowing). Values of the uncertainty on k_{eff} from each of these components are given in Section 3.3.4.1.3F of the SAR at the 2σ level. These uncertainties in k_{eff} due to mechanical tolerances are added in quadrature.

$k_{eff} = k_{-nominal}$ (from Table J.6–5) + $2\sigma_{KNEO}$ (from Table J.6–5) +

0.007($B_{-method}$ from[J-9]) + $[(0.01758)^2 + (0.02551)^2 + (0.02157)^2 + (0.01304)^2]^{1/2}$(Mechanical Uncertainties from Section 3.3.4.1.3F of the SAR)

$$k_{eff} = k_{-nominal} + 2\sigma_{KNEO} + 0.0469$$

This approach is conservative for at least two reasons. The $k_{-nominal}$ values assume an infinitely long fuel region axially through the use of specular boundary conditions on the ends and take no credit for residual neutron absorber in the BPRA's.

The B&W 15x15 Mark B with BPRAs at a moderator density of 0.85 g/cm^3 represents the most reactive configuration for the three fuel types at moderator densities greater than 0.85 g/cm^3. The maximum final k_{eff} for this configuration is 0.9203, as shown in Table J.6–5. This result is well below the 0.95 limit for reactivity.

Table J.6–5 DSC k_{eff} for PWR Assemblies vs Moderator Density

Fuel Assembly Type	Moderator Density, g/cc	Nominal k_{eff}	$1\sigma_{KENO}$	Final k_{eff}
B&W Mark B	0.9982	0.8508	0.0013	0.9002
	0.90	0.8637	0.0012	0.9129
	0.85	0.8710	0.0012	0.9203
Westinghouse Std	0.9982	0.8406	0.0014	0.8903
	0.90	0.8529	0.0013	0.9023
	0.85	0.8592	0.0013	0.9086
Westinghouse OFA	0.9982	0.8250	0.0011	0.8740
	0.90	0.8339	0.0012	0.8833
	0.85	0.8477	0.0013	0.8971

J.6.5 Critical Benchmark Experiments

Verification and validation of SCALE4.4, CSAS26 (KENO-VI)and the 44 group ENDF/B-V cross section library has been performed by TN West to determine any method bias [J-9]. A series of 26 critical experiments with square lattice fuel were modeled with CSAS26. These critical experiments were selected for a range of ^{235}U enrichments, fuel rod pitches, moderator/fuel volume ratios that bracket the three PWR fuel designs. The results of the 26 computer simulations were entered into the USLSTATS code [J-9] to determine the USL and look for any correlation with these parameters. It was concluded that no significant correlations are present due to system parameters such as fuel enrichment, absorber characteristics, fuel/moderator volume ratios, soluble boron content, etc. In particular, the correlation between moderator/fuel volume ratio, which is a system parameter that is analogous to changing the moderator density, and k_{eff}, is not statistically significant (Pearson-r value =-0.34). The USL is calculated to be 0.9430. The method bias is therefore 0.95-0.9430 = 0.007.

J.7 Confinement

J.7.1 Confinement Boundary

The Confinement Boundary Analysis is not affected by the addition of BPRAs. The addition of BPRAs to the approved contents does not alter the previous Confinement Boundary analysis.

J.8 Operating Procedures

J.8.1 Procedures for Loading/Unloading the Cask

The addition of BPRAs to the approved contents does not alter any of the previous Loading/Unloading Procedures of the NUHOMS system as described in Chapter 5 of the SAR.

J.9 Acceptance Tests and Maintenance Program

The addition of BPRAs to the approved contents does not alter any of the previous Acceptance Tests, Maintenance and Surveillance Actions for the NUHOMS® system as described in Chapters 9 and 10 of the SAR.

J.10 Radiation Protection

Section 7.4.1 of the SAR discusses the anticipated cumulative dose exposure to site personnel during the fuel handling and transfer activities associated with unitizing one NUHOMS® HSM for storage of one DSC. Chapter 5 of the SAR describes in detail the NUHOMS® operational procedures, several of which involve potential exposure to personnel.

The expected occupational dose for placing a canister of spent fuel into dry storage for the operational steps listed in Table 7.4-1 is less than 1.2 person-rem. The additional occupational dose due to BPRAs is expected to be 0.3 person-rem. This increase is due mainly to the increase in the expected gamma dose rate during preparation for welding. All other exposures are expected to remain bounded by the previous evaluations because the fuel qualification tables in Section 10 of the SAR limit the fuel plus BPRA dose rates to be the same as for fuel only.

J.11 Accident Analyses

There are no changes to this Section. The radioactive material release source terms remain unchanged and the bounding dose rates and decay heat values considered during accident conditions remain unchanged. The increase in DSC pressure due to the presence of BPRAs is discussed in Section J.4 and is bounded by the SAR analysis.

J.12 Operation Controls and Limits

The revised operation controls and limits are included in the revised Certificate of Compliance and in Section 10 of the CSAR. Section 10.3 of the CSAR is revised to implement the C of C changes.

J.13 Quality Assurance

This Section remains unchanged. Only the authorized contents of the Cask changes with this amendment request.

J.14 References

J-1 "Characteristics of Potential Repository Wastes," DOE/RW-0184/R1," Office of Civilian Radioactive Waste Management, July 1992.

J-2 "ORIGEN2.1 – Isotope Generation and Depletion Code – Matrix Exponential Method," CCC-371, Oak Ridge National Laboratory, RSIC Computer Code Collection, File Number QA040.208.0002, August 1991.

J-3 Ludwig, S.B., and J.P. Renier, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018 Oak Ridge National Laboratory, December 1989.

J-4 Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers," NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6. (SCALE 4.4)

J-5 "Characteristics of Potential Repository Wastes," Department of Energy, Office of Civilian Radioactive Waste Management, DOE/RW-0184-R1, July, 1992.

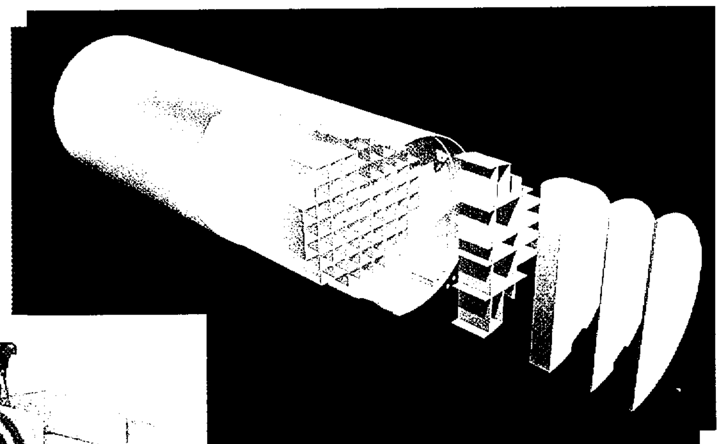
J-6 "Domestic Light Water Reactor Fuel Design Evolution, Volume III," Nuclear Assurance Corporation, September, 1981, DOE/ET/47912-3.

J-7 Roddy, J.W., et.al., "Physical and Decay Characteristics of Commercial LWR Spent Fuel," Oak Ridge National Laboratory October, 1985, ORNL/TM-9591/V1.

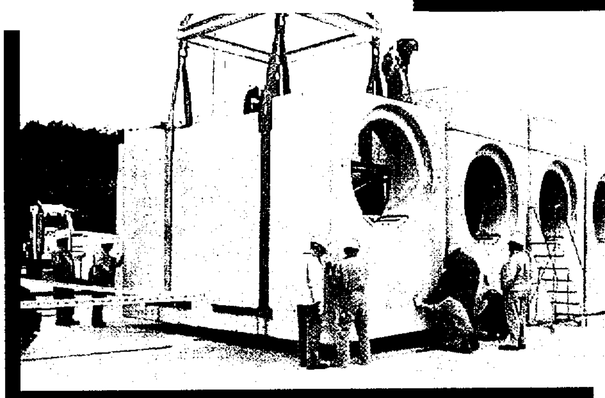
J-8 Nuclides and Isotopes, Chart of the Nuclides, 15th Edition, GE Nuclear Energy, 1996.

J-9 Verification and Validation Document SCALE4.4PC: CSAS26X and CSAS26, Transnuclear West Document SCALE4.4PCk-QA, Rev. 1, 6/15/99, Document No. QA040.223.0001.

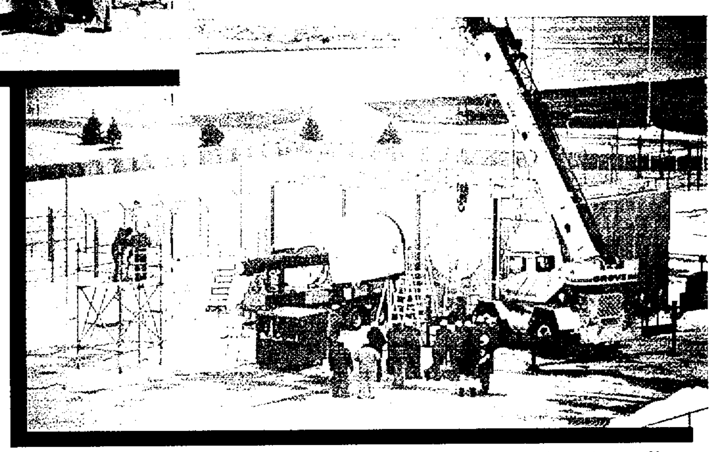
Standardized NUHOMS®
Horizontal Modular Storage System
for Irradiated Nuclear Fuel



61BT Dry Shielded Canister



HSM Placement



HSM Loading

FINAL SAFETY ANALYSIS REPORT
Volume 2 of 2

TRANSNUCLEAR WEST

TABLE OF CONTENTS

TABLE OF CONTENTS

(Continued)

Page

TABLE OF CONTENTS

(Continued)

TABLE OF CONTENTS

(Continued)

TABLE OF CONTENTS

(Continued)

TABLE OF CONTENTS

(Continued)

LIST OF TABLES

LIST OF TABLES

(Continued)

LIST OF TABLES

(Continued)

LIST OF FIGURES

LIST OF FIGURES

(Continued)

LIST OF FIGURES

(Continued)

Page

LIST OF FIGURES

(Continued)

K.1 General Discussion

Appendix K addresses the Important to Safety aspects of storing spent fuel in the NUHOMS®-61BT system. The NUHOMS®-61BT system consists of a NUHOMS®-61BT Dry Shielded Canister (DSC) stored in a NUHOMS® Horizontal Storage Module (HSM) and transferred in a OS197 Transfer Cask (TC). The format follows the guidance provided in NRC Regulatory Guide 3.61 [1.1].

The NUHOMS®-61BT system provides confinement, shielding, criticality control and passive heat removal independent of any other facility structures or components. The NUHOMS®-61BT DSC also maintains structural integrity of the fuel during storage.

The addition of NUHOMS®-61BT DSC to the standardized NUHOMS® system was approved by the NRC effective September 12, 2001 as documented in Amendment No. 3 to CoC 1004 and the associated Safety Evaluation Report [1.4]. As noted in SER Section 1.1, the NUHOMS®-61BT DSC is designed to store 61 intact, or a combination of up to 16 damaged and the remainder intact BWR fuel assemblies. However, damaged BWR fuel as specified in Table K.2.2, is not authorized for storage in NUHOMS®-61BT system.

NOTE: As noted in various sections of the SER [1.4], the safety analysis presented in Appendix K for storing damaged fuel has been reviewed and concurred with by the NRC with the exception of one issue as described in SER section 8.1.1. This issue relates to the absence of an analysis in Appendix K which evaluates the effects of stresses on the fuel cladding associated with normal and off-normal condition. TN West plans to address this issue by submitting a new amendment to CoC 1004. Accordingly, all safety analyses, including structural analyses, presented in Appendix K which supports storage of damaged fuel is being retained and will be updated upon approval of this future amendment.

K.1.1 Introduction

The standardized NUHOMS® system provides a modular canister based spent fuel storage and transport system. This system includes DSCs, HSMs; and the TC.

Appendix K adds the 61BT DSC to the standardized NUHOMS® system. Only those features that are being revised or added to the NUHOMS® system are addressed and evaluated in this Appendix. Sections of this Appendix which are not affected by the addition of 61BT DSC are indicated in this Appendix with "No Change." The HSM and the TC remain unchanged. The NUHOMS®-61BT DSC is similar to the existing DSCs with the following exceptions:

- The canister shell thickness is reduced from 0.625 inches to 0.5 inches.
- The canister has been upgraded to provide a leak tight confinement.
- The basket represents a new design.
- The thickness of the top and bottom shield plug is reduced slightly to accommodate the new basket design.

The NUHOMS®-61BT DSC is designed to store 61 intact, or up to 16 damaged and remainder intact, for a total of 61, standard Boiling Water Reactor (BWR) fuel assemblies with or without fuel channels. The NUHOMS®-61BT DSC is designed for a maximum heat load of 18.3 kW or 0.3 kW/assembly. The fuel which may be stored in the NUHOMS®-61BT DSC is presented in Section K.2.0.

K.1.2 General Description of the NUHOMS®-61BT DSC

K.1.2.1 NUHOMS®-61BT DSC Characteristics

Each NUHOMS®-61BT DSC consists of a fuel basket and a canister body (shell, canister inner bottom and top cover plates and shield plugs). A sketch of the 61BT DSC is shown in Figure K.1-1. A set of reference drawings is presented in Section K.1.5. Dimensions and the estimated weight of the NUHOMS®-61BT DSC are shown in Table K.1-1. The NUHOMS®-61BT DSC shell thickness is 0.50 inches instead of 0.625 inches as used for the NUHOMS®-24P or –52B DSC designs. The bottom and top shield plugs are 5.0 and 7.0 inches respectively as compared to the 5.75 and 8.0 inches used for the NUHOMS®-52B DSC designs. The materials used to fabricate the DSC are shown in the Parts List on Drawing NUH-61B1065 contained in Section K.1.5.

The confinement vessel for the NUHOMS®-61BT DSC consists of a shell which is a welded, stainless steel cylinder with an integrally-welded, stainless steel bottom closure assembly; and a stainless steel top closure assembly, which includes the vent and drain system.

There are no penetrations through the confinement vessel. The draining and venting systems are covered by the seal welded outer top closure plate and vent port plug. To preclude air in-leakage, the canister cavity is pressurized above atmospheric pressure with helium. The NUHOMS®-61BT DSC is designed and tested to meet the leak tight criteria of ANSI N14.5-1997.

The basket structure consists of assemblies of stainless steel fuel compartments held in place by basket rails and holddown rings. The four and nine compartment assemblies are held together by welded stainless steel boxes wrapped around the fuel compartments, which also retain the neutron poison plates between the compartments in the assemblies. The borated aluminum or boron carbide/aluminum metal matrix composite plates or Boral® (neutron poison plates) provide the necessary criticality control and provide the heat conduction paths from the fuel assemblies to the cask cavity wall. This method of construction forms a very strong structure of compartment assemblies which provide for storage of 61 fuel assemblies. The minimum open dimension of each fuel compartment is 5.8 in. x 5.8 in., which provides clearance around the fuel assemblies.

There are three NUHOMS®-61BT DSC basket types, A, B, and C, as shown on Drawing NUH-61B1065 contained in Section K.1.5. The types are identical with the exception of the minimum B10 content of the poison plates. The maximum lattice average enrichment of the fuel assemblies allowed by basket type is given in Table K.2-4. Damaged fuel is only stored in Type C baskets, in four corner compartment assemblies with endcaps installed on the respective damaged fuel compartments.

During dry storage of the spent fuel in the NUHOMS®-61BT system, no active systems are required for the removal and dissipation of the decay heat from the fuel. The NUHOMS®-61BT DSC is designed to transfer the decay heat from the fuel to the basket, from the basket to the canister body and ultimately to the ambient via HSM or TC.

Each canister is identified by a Mark Number, NUHOMS®-61BT DSC -XX, Type Y. where XX is a sequential number corresponding to a specific canister, and Y refers to the basket type. Each canister is also marked with the patent number.

K.1.2.2 Operational Features

K.1.2.2.1 General Features

The NUHOMS®-61BT DSC is designed to safely store 61 intact, or up to 16 damaged and remainder intact, for a total of 61, standard BWR fuel assemblies with or without fuel channels. The NUHOMS®-61BT DSC is designed to maintain the fuel cladding temperature below 649°F (343°C) during storage. It is also designed to maintain the fuel cladding temperature below 1058°F (570°C) during short-term accident conditions, short-term off-normal conditions and fuel transfer operations.

The criticality control features of the NUHOMS®-61BT DSC are designed to maintain the neutron multiplication factor k-effective less than the upper subcritical limit equal to 0.95 minus benchmarking bias and modeling bias under all conditions.

K.1.2.2.2 Sequence of Operations

The sequence of operations to be performed in loading fuel into the NUHOMS®-61BT DSC is presented in Chapter K.8. The operations are the same as presented in Chapter 5 with the exception of the handling of the OS197 TC with NUHOMS®-61BT DSC using a 100-ton rated crane.

K.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

K.1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry and by utilizing neutron poison in the fuel basket. These features are only necessary during the loading and unloading operations that occur in the loading pool (underwater). During storage, with the DSC cavity dry and sealed from the environment, criticality control measures within the installation are not necessary because of the low reactivity of the fuel in the dry NUHOMS®-61BT DSC and the assurance that no water can enter the DSC cavity during storage.

K.1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the NUHOMS®-61BT system.

K.1.2.2.3.3 Operation Shutdown Modes

The NUHOMS®-61BT DSC is a totally passive system so that consideration of operation shutdown modes is unnecessary.

K.1.2.2.3.4 Instrumentation

No change.

K.1.2.2.3.5 Maintenance Techniques

No change.

K.1.2.3 Cask Contents

The NUHOMS®-61BT DSC is designed to store 61 intact, or up to 16 damaged and remainder intact, for a total of 61, standard Boiling Water Reactor (BWR) fuel assemblies with or without fuel channels. The NUHOMS®-61BT DSC is designed for a maximum heat load of 18.3 kW or 0.3 kW/assembly. The fuel which may be stored in the NUHOMS®-61BT DSC is presented in Table K.2-3.

Chapter K.5 provides the shielding analysis. Chapter K.6 covers the criticality safety of the NUHOMS®-61BT DSC and its contents, listing material densities, moderator ratios, and geometric configurations.

K.1.3 Identification of Agents and Contractors

Transnuclear West, Inc. (TNW), provides the design, analysis, licensing support and quality assurance for the NUHOMS®-61BT system. Fabrication of the NUHOMS®-61BT system cask is done by one or more qualified fabricators under TNW's quality assurance program. TNW's quality assurance program is described in Chapter K.13. This program is written to satisfy the requirements of 10 CFR 72, Subpart G and covers control of design, procurement, fabrication, inspection, testing, operations and corrective action. Experienced TNW operations personnel provide training to utility personnel prior to first use of the NUHOMS®-61BT system and prepare generic operating procedures.

Managerial and administrative controls, which are used to ensure safe operation of the casks, are provided by the host utility. NUHOMS®-61BT system operations and maintenance are performed by utility personnel. Decommissioning activities will be performed by utility personnel in accordance with site procedures.

Transnuclear West, Inc. provides specialized services for the nuclear fuel cycle that support transportation, storage and handling of spent nuclear fuel, radioactive waste and other radioactive materials. TNW is the holder of CoC 1004.

K.1.4 Generic Cask Arrays

No change.

K.1.5 Supplemental Data

The following Transnuclear West drawings are enclosed:

1. NUHOMS® -61B Transportable Canister for BWR Fuel General Arrangement, Drawing NUH-61B1060NP-SAR.

2. NUHOMS® -61B Transportable Canister for BWR Fuel Shell Assembly, Drawing NUH-61B1061NP-SAR.

3. NUHOMS® -61B Transportable Canister for BWR Fuel Canister Details, Drawing NUH-61B1062NP-SAR.

4. NUHOMS® -61B Transportable Canister for BWR Fuel Basket Assembly, Drawing NUH-61B1063NP-SAR.

5. NUHOMS® -61B Transportable Canister for BWR Fuel Basket Details, Drawing NUH-61B1064NP-SAR.

6. NUHOMS® -61B Transportable Canister for BWR Fuel Parts List, Drawing NUH-61B1065NP-SAR.

7. NUHOMS® -61B Transportable Canister Top & Bottom Cap Details for Failed BWR Fuel, Drawing NUH-61B1066NP-SAR.

Security-Related Information Figure Withheld Under 10 CFR 2.390.

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ± .06
6.00 TO 48.00: ± .13
GREATER THAN 48.00: ± .25
ANGLES: ± 1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

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

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL GENERAL ARRANGEMENT

FILE NO.	DWG NO.	SCALE	SHEET	REV NO.
NUH61B-1060-SAR	NUH-61B-1060NP-SAR	NONE	1 OF 2	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ± .06
6.00 TO 48.00: ± .13
GREATER THAN 48.00: ± .25
ANGLES: ± 1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL GENERAL ARRANGEMENT

FILE NO.	DWG NO.	SCALE	SHEET	REV. NO.
NUH61B-1060-SAR	NUH-61B-1060NP-SAR	NONE	2 OF 2	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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

ALL DIMENSIONS ARE APPLICABLE AT 80°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ±.06
6.00 TO 48.00: ±.13
GREATER THAN 48.00: ±.25
ANGLES: ±1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL SHELL ASSEMBLY

NUH61B-1061-SAR	NUH-61B-1061NP-SAR	NONE	1 OF 1	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ± .06
6.00 TO 48.00: ± .13
GREATER THAN 48.00: ± .25
ANGLES: ± 1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

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

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL CANISTER DETAILS

NUH61B-1062-SAR	NUH-61B-1062NP-SAR	NONE	1 OF 2	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ± .06
6.00 TO 48.00: ± .13
GREATER THAN 48.00: ± .25
ANGLES: ± 1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL CANISTER DETAILS

NUH61B-1062-SAR	NUH-61B-1062NP-SAR	NONE	2 OF 2	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ± .06
6.00 TO 48.00: ± .13
GREATER THAN 48.00: ± .25
ANGLES: ± 1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL BASKET ASSEMBLY

NUH61B-1063-SAR	NUH-61B-1063NP-SAR	NONE	1 OF 1	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ± .06
6.00 TO 48.00: ± .12
GREATER THAN 48.00: ± .25
ANGLES: ± 1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL BASKET DETAILS

NUH61B-1064-SAR | NUH-61B-1064NP-SAR | NONE | 1 OF 2 | 0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:
DIMENSIONS LESS THAN 6.00: ± .06
6.00 TO 48.00: ± .13
GREATER THAN 48.00: ± .25
ANGLES: ± 1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL BASKET DETAILS

NUH61B-1064-SAR	NUH-61B-1064NP-SAR	NONE	2 OF 2	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:

DIMENSIONS LESS THAN 8.00: ± N/A
8.00 TO 48.00: ± N/A
GREATER THAN 48.00: ± N/A
ANGLES: ± N/A

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

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

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT
NUHOMS® 61B TRANSPORTABLE CANISTER FOR BWR FUEL PARTS LIST

FILE NO.	DWG NO.	SCALE	SHEET	REV NO.
NUH61B-1065-SAR	NUH-61B-1065NP-SAR	NONE	1 OF 1	0

Security-Related Information Figure Withheld Under 10 CFR 2.390.

ALL DIMENSIONS ARE APPLICABLE AT 68°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.

UNLESS NOTED OTHERWISE, TOLERANCES SHALL BE AS FOLLOWS:

DIMENSIONS LESS THAN 8.00: ±.06
8.00 TO 48.00: ±.13
GREATER THAN 48.00: ±.25
ANGLES: ±1°

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

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

TRANSNUCLEAR WEST

FINAL SAFETY ANALYSIS REPORT

NUHOMS® 61B TRANSPORTABLE CANISTER TOP & BOTTOM CAP DETAILS FOR DAMAGED BWR FUEL

FILE NO.	DWG NO.	SCALE	SHEET	REV NO.
NUH61B-1066-SAR	NUH-61B-1066NP-SAR	NONE	1 OF 1	0

K.1.6 References

1.1 US Nuclear Regulatory Commission, Regulatory Guide 3.61, Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask, February, 1989.

1.2 10CFR72, Rules and Regulations, Title 10, Chapter 1, Code of Federal Regulations - Energy, U.S. Nuclear Regulatory Commission, Washington, D.C., "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

1.3 10CFR71, Rules and Regulations, Title 10, Chapter 1, Code of Federal Regulations - Energy, U.S. Nuclear Regulatory Commission, Washington, D.C., "Packaging and Transportation of Radioactive Material."

1.4 U.S. Nuclear Regulatory Commission, Office of the Nuclear Materials Safety and Safeguards, "Safety Evaluation Report, Addition of the NUHOMS®-61BT Dry Shielded Canister and Additional Fuel Types," September 17, 2001.

1.5 NUHOMS® Certificate of Compliance for Dry Spent Fuel Storage Casks, Certificate of Number 1004, Amendment No. 3, September 2001 (Docket 72-1004).

Table K.1-1
Nominal Dimensions and Weight of the NUHOMS®-61BT DSC

Overall length (with grapple, in)	199.7
Outside diameter (in)	67.25
Cavity diameter (in)	66.25
Cavity length (in)	179.3
Nominal DSC weight:	
Loaded on storage pad (kips)	88.5

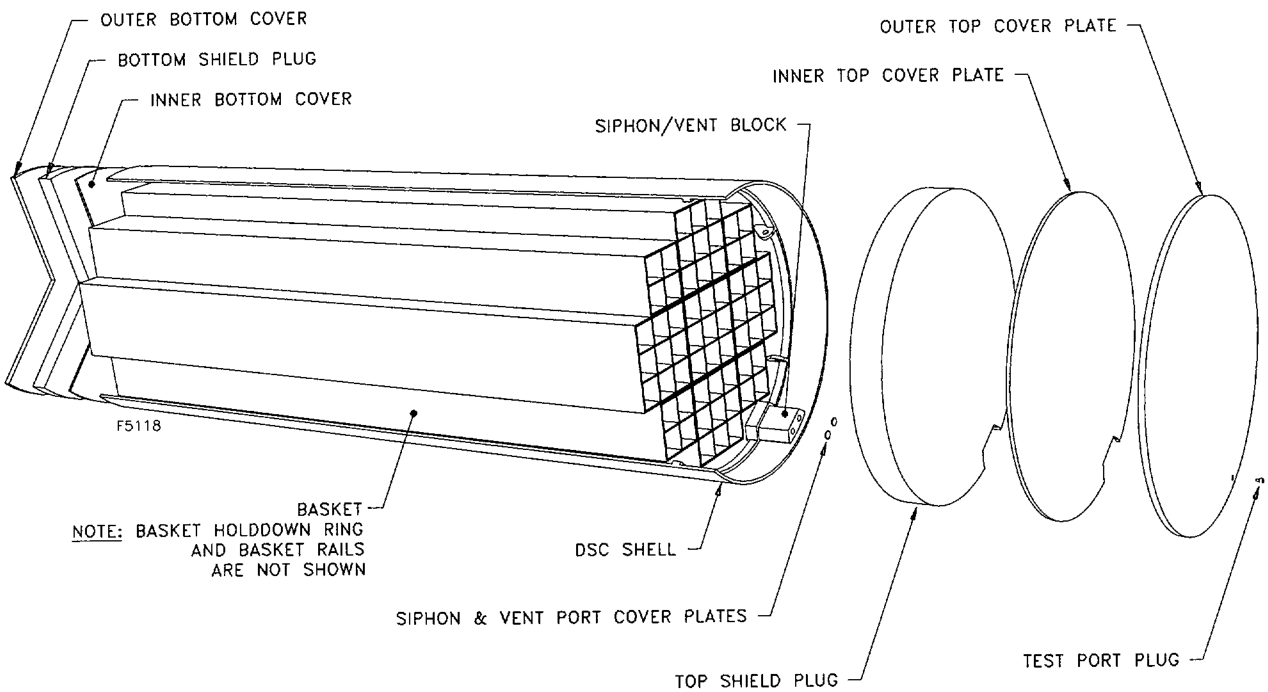


Figure K.1-1
NUHOMS®-61BT DSC Components

K.2 Principal Design Criteria

This section provides the principal design criteria for the NUHOMS®-61BT system. The NUHOMS®-61BT Dry Shielded Canister (DSC) is handled, transferred and stored in the same manner as the existing NUHOMS®-52B DSC. There is no change to the NUHOMS® OS197 TC or the standard NUHOMS® Horizontal Storage Module (HSM). Only those principal design criteria that have changed from Chapter 3, are described in this chapter. Section K.2.1 presents a general description of the spent fuel to be stored. Section K.2.2 provides the design criteria for environmental conditions and natural phenomena. This section contains an assessment of the local damage due to the design basis environmental conditions and natural phenomena and the general loadings and design parameters used for analysis in subsequent chapters. Section K.2.3 provides a description of the systems which have been designated as important to safety. Section K.2.4 discusses decommissioning considerations. Section K.2.5 summarizes the NUHOMS®-61BT DSC design criteria.

K.2.1 Spent Fuel To Be Stored

The NUHOMS®-61BT DSC is designed to store 61 intact, or up to 16 damaged and the remainder intact, for a total of 61, standard BWR fuel assemblies with or without fuel channels. The NUHOMS®-61BT DSC can store intact BWR fuel assemblies with the characteristics described in Table K.2-1, or damaged and intact BWR fuel assemblies with the characteristics described in Table K.2-2, which include a variety of cooling times, enrichment and maximum bundle average burnup. Damaged BWT fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks or with cracked, bulging, or discolored cladding. Missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through the gap created opening during handling and retrievability is assured following Normal/Off-Normal conditions.

Note: Damaged BWR fuel as described in Table K.2-2 is currently not authorized for storage in NUHOMS®-61BT system. See page K.1-1 for explanation.

The NUHOMS®-61BT DSC may store BWR fuel assemblies with a maximum decay heat of 300 watts/assembly, or a total of 18.3 kW. The NUHOMS®-61BT DSC is inserted and backfilled with helium at the time of loading. The maximum fuel assembly weight with channel is 705 lbs.

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, heat load and confinement. The fuel assemblies considered are listed in Table K.2-3. It was determined that the GE 7x7 is the enveloping fuel design for the shielding source term calculation. However, for criticality safety, the GE 10x10 assembly is the most reactive, and is evaluated for configurations that bound all normal, off-normal and accident conditions.

The NUHOMS®-61BT DSC has three basket configurations, based on the boron content in the poison plates. The maximum lattice average enrichment authorized for Type A, B and C NUHOMS®-61BT DSCs is 3.7, 4.1 and 4.4 weight percent (wt. %) U-235, respectively.

Intact BWR fuel assemblies may be stored in any of the three NUHOMS®-61BT DSC Types provided the loading meets the maximum lattice average enrichment limit for the NUHOMS®-61BT DSC type, as given on Table K.2-4. Damaged BWR fuel assemblies may only be stored in Type C NUHOMS®-61BT DSCs with endcaps installed on each four compartment assembly where a damaged fuel assembly is stored.

Fuel assemblies with various combinations of burnup, enrichment and cooling time can be stored in the NUHOMS®-61BT DSC as long as the fuel assembly parameters fall within the design limits specified in Table K.2-1 or Table K.2-2, and Table K.2-4.

For calculating the maximum internal pressure in the NUHOMS®-61BT DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event. A minimum of 100% of the fill gas and 30% of the fission gases (e.g., H-3, Kr

and Xe) within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

The maximum design basis internal pressures for the NUHOMS®-61BT DSC are 10, 20 and 65 psig for normal, off-normal and accident conditions of storage, respectively.

K.2.1.1 General Operating Functions

No change.

K.2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The NUHOMS®-61BT DSC is handled and stored in the same manner as the existing NUHOMS®-52B system. The environmental conditions and natural phenomena are the same as described in Chapter 3. Updated criteria are given in the applicable section. Table K.2-10 summarizes the design criteria for the 61BT DSC. This table also summarizes the applicable codes and standards utilized for design. Design criteria for the NUHOMS® HSM and TC remain are as shown in Table 3.2-1.

K.2.2.1 Tornado Wind and Tornado Missiles

No change.

K.2.2.2 Water Level (Flood) Design

No change.

K.2.2.3 Seismic Design

No change.

K.2.2.4 Snow and Ice Loading

No change.

K.2.2.5 Combined Load Criteria

The NUHOMS®-61BT system is subjected to the same loads as the existing NUHOMS®-24P or -52B system. The criteria applicable to the HSM and the OS197 TC are the same as those found in Chapter 3. The criteria applicable to the NUHOMS®-61BT DSC are found in the following subsections.

K.2.2.5.1 NUHOMS®–61BT DSC Structure Design Criteria

The NUHOMS®-61BT DSC is designed using the ASME Boiler and Pressure Vessel Code [2.2] criteria given in Chapter 3, except as noted in the following sections. A summary of the NUHOMS®-61BT DSC load combinations is presented in Table K.2-5.

K.2.2.5.1.1 NUHOMS®–61BT DSC Shell Stress Limits

The stress limits for the NUHOMS®-61BT DSC shell are taken from the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-3200 [2.2] for normal condition loads (Level A) and Appendix F for accident condition loads (Level D).

- The stress due to each load shall be identified as to the type of stress induced, e.g., membrane, bending, etc., and the classification of stress, e.g., primary, secondary, etc.

- Stress limits for Level A and D service loading conditions are given in Table K.2-6. Local yielding is permitted at the point of contact where the Level D load is applied. If elastic stress limits cannot be met, the plastic system analysis approach and acceptance criteria of Appendix F of ASME Section III shall be used.

- Reference to ASME, Section III, Subsection NB, Paragraph NB-3223 and 3224 for Level B and Level C stress limits.

- The allowable stress intensity value, S_m, as defined by the Code shall be taken at the temperature calculated for each service load condition.

K.2.2.5.1.2 NUHOMS®–61BT DSC Basket Stress Limits

The basket fuel compartment wall thickness is established to meet heat transfer, nuclear criticality, and structural requirements. The basket structure must provide sufficient rigidity to maintain a subcritical configuration under the applied loads.

The primary stress analyses of the basket for Level A (Normal Service) and sustained Level D conditions do not take credit for the neutron poison plates except for through thickness compression. The poison plate strength is, however, considered when determining secondary stresses in the stainless steel.

Normal Conditions

- The basis for the stainless steel basket assembly stress allowables is the ASME Code, Section III, Subsection NG. The primary membrane stress intensity and membrane plus bending stress intensities are limited to S_m (S_m is the code allowable stress intensity) and 1.5 S_m, respectively, for Level A (Normal Service) load combinations. The average primary shear stress is limited to 0.6 S_m.

- The ASME Code provides a basic $3S_m$ limit on primary plus secondary stress intensity for Level A conditions. That limit is specified to prevent ratcheting of a structure under cyclic loading and to provide controlled linear strain cycling in the structure so that a valid fatigue analysis can be performed.

- Reference to ASME Section III, Subsection NG, paragraph NG-3223 and NG-3224 for Level B and Level C stress limits.

Accident Conditions

- The basket shall be evaluated under Level D Service loadings in accordance with the Level D Service limits for components in Appendix F of Section III of the Code. The hypothetical impact accidents are evaluated as short duration Level D conditions. For elastic quasistatic analysis, the primary membrane stress (P_m) is limited to the smaller of $2.4S_m$ or $0.7S_u$ and

membrane plus bending stress intensities are limited to the smaller of $3.6S_m$ or $1.0S_u$. The average primary shear stress is limited to 0.42 S_u. When evaluating the results from the non-linear elastic-plastic analysis for the accident conditions, the general primary membrane stress intensity, P_m, shall not exceed $0.7S_u$ and the maximum stress intensity at any location (P_l or $P_l + P_b$) shall not exceed 0.9 S_u.

- The fuel compartment walls and basket rails, when subjected to compressive loadings, are also evaluated using ASME Code Subsection NF rules to ensure that buckling will not occur. The acceptance criteria (allowable buckling loads) are taken from ASME Code, Section III, Appendix F, paragraph F-1341.3, Collapse Load. The allowable buckling load is equal to 100% of the calculated plastic analysis collapse load or 100% of the test collapse load.

- The stress and load limits for the basket are summarized in Table K.2-7.

K.2.3 Safety Protection Systems

K.2.3.1 General

The NUHOMS®-61BT DSC is designed to provide storage of spent fuel for at least 40 years. The cask cavity pressure is always above atmospheric during the storage period as a precaution against the in-leakage of air which could be harmful to the fuel. Since the confinement vessel consists of a steel cylinder with an integrally-welded bottom closure, and a seal welded top closure that is verified to be leak tight after loading, the cavity gas cannot escape.

Only those features that are not addressed in Chapter 3, or have been revised, are addressed in this Section. Those features include the thermal and nucleonic performance of the poison plates, and their acceptance. Components of the NUHOMS®-61BT DSC that are "Important to Safety" and "Not Important to Safety" are listed in Table K.2-8.

K.2.3.2 Protection By Multiple Confinement Barriers and Systems

The NUHOMS®-61BT DSC provides a leak tight confinement of the spent fuel. Although similar to the existing -52B DSC, sealing of the NUHOMS®-61BT DSC involves leak testing in accordance with ANSI N14.5 [2.3] after loading and sealing the canister, as described in Section K.9.

The NUHOMS®-61BT DSC poison plates are required to meet the minimum uniform boron concentration limits of Table K.2-4 in support of criticality safety. A detailed acceptance program for the neutron poison material is given in Section K.9. The program also requires that the plates be tested to verify they meet the minimum thermal conductivity limits given in Section K.4.

K.2.3.3 Protection By Equipment and Instrumentation Selection

No change.

K.2.3.4 Nuclear Criticality Safety

K.2.3.4.1 Control Methods for Prevention of Criticality

The design criterion for criticality is that an upper subcritical limit (USL) of 0.95 minus benchmarking bias and modeling bias will be maintained for all postulated arrangements of fuel within the DSC. The intact fuel assemblies and the damaged fuel assemblies are assumed to stay within their basket compartment based on the DSC and basket geometry.

The control method used to prevent criticality is incorporation of poison material in the basket material and favorable geometry. The quantity and distribution of boron in the poison material is controlled by specific manufacturing and acceptance criteria of the poison plates. The acceptance of the plates is described in Section K.9.

The basket has been designed to assure an ample margin of safety against criticality under the conditions of fresh fuel in a DSC flooded with fresh water. The method of criticality control is in accordance with the requirements of 10CFR72.124.

The criticality analyses are described in Section K.6.

K.2.3.4.2 Error Contingency Criteria

Provision for error contingency is built into the criterion used in Section K.2.3.4.1 above. The criterion used in the criticality analysis is common practice for licensing submittals. Because conservative assumptions are made in modeling, it is not necessary to introduce additional contingency for error.

K.2.3.4.3 Verification Analysis-Benchmarking

The verification analysis-benchmarking used in the criticality safety analysis is described in Section K.6.

K.2.3.5 Radiological Protection

No change.

K.2.3.6 Fire and Explosion Protection

No change.

K.2.4 Decommissioning Considerations

No change.

K.2.5 Summary of NUHOMS®-61BT DSC Design Criteria

The additional principal design criteria for the NUHOMS®-61BT DSC are presented in Table K.2-1. The NUHOMS®-61BT DSC is designed to store 61 intact, or up to 16 damaged and the remainder intact, for a total of 61, standard BWR fuel assemblies with or without fuel channels with assembly average burnup, initial enrichment and cooling time as described in Table K.2-1, Table K.2-2 and Table K.2-4.

The maximum total heat generation rate of the stored fuel is limited to 0.3 kW per fuel assembly and 18.3 kW per NUHOMS®-61BT DSC in order to keep the maximum fuel cladding temperature below the limit necessary to ensure cladding integrity for 40 years storage [2.4]. The fuel cladding integrity is assured by the NUHOMS®-61BT DSC and basket design which limits fuel cladding temperature and maintains a nonoxidizing environment in the cask cavity [2.5], as described in Section K.4.

The NUHOMS®-61BT DSC (shell and closure) is designed and fabricated to the maximum practicable extent as a Class I component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-3200.

The NUHOMS®-61BT DSC is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. Poison materials in the fuel basket are employed to maintain the upper subcritical limit of 0.9414. The basket is designed and fabricated to the maximum practicable extent in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200.

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

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

K.2.6 References

2.1 NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," 1997.

2.2 American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Section III, Division 1 - Subsections NB and NG, 1998 edition including 1999 addenda.

2.3 ANSI N14.5-1997, "Leakage Tests on Packages for Shipment," February 1998.

2.4 Levy, et. al., "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy - Clad Fuel Rods in Inert Gas," Pacific Northwest Laboratory, PNL-6189, 1987.

2.5 Johnson, et. al., "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases," PNL-4835, Pacific Northwest Laboratory, Richland, Wash., Sept. 1983.

Table K.2-1
Intact BWR Fuel Assembly Characteristics

PHYSICAL PARAMETERS:	
Fuel Design:	7x7, 8x8, 9x9, or 10x10 intact BWR fuel assemblies manufactured by General Electric or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Table K.2-3.
Cladding Material:	Zircaloy
Fuel Damage:	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR Fuel."
Channels:	Fuel may be stored with or without fuel channels
RADIOLOGICAL PARAMETERS(1):	
Group 1:	
Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	5-years
Maximum Initial Enrichment:	See Table K.2-4
Minimum Initial Bundle Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 2:	
Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	8-years
Maximum Initial Enrichment:	See Table K.2-4
Minimum Initial Bundle Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 3:	
Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	6.5-years
Maximum Initial Enrichment:	See Table K.2-4
Minimum Initial Bundle Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 4:	
Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	10-years
Maximum Initial Enrichment:	See Table K.2-4
Minimum Initial Bundle Average Enrichment:	3.4 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly

(1) Fuel assemblies fully complying with any of the four groups of parameters are suitable for storage in the NUHOMS®-61BT DSC. No interpolation of Radiological Parameters is permitted between Groups.

Table K.2-2
Damaged BWR Fuel[1] Assemblies Characteristics

PHYSICAL PARAMETERS:	
Fuel Design:	7x7, 8x8 BWR damaged fuel assemblies manufactured by General Electric or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Table K.2-3 for the 7x7 and 8x8 designs only.
Cladding Material:	Zircaloy
Fuel Damage:	Damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks or with cracked, bulging, or discolored cladding. Missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through the gap created by the cladding opening during handling and retrievability is assured following Normal/Off-Normal conditions. Damaged fuel shall be stored with Top and Bottom Caps for Failed Fuel. Damaged fuel may only be stored in the 2x2 compartments of the "Type C" NUHOMS®-61B Canister.
Channels:	Fuel may be stored with or without fuel channels.
RADIOLOGICAL PARAMETERS[2]:	
Group 1:	
Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	5-years
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 2:	
Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	8-years
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 3:	
Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	6.5-years
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly

(1) Damaged fuel is not authorized for storage per note on page K.1-1.
(2) Fuel assemblies fully complying with any of the four groups of parameters are suitable for storage in the NUHOMS®-61BT DSC. No interpolation of Radiological Parameters is permitted between Groups.

Table K.2-2
Damaged BWR Fuel[1] Assemblies Characteristics

(Concluded)

RADIOLOGICAL PARAMETERS[3]: *Group 4:* Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	10-years
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	3.4 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly

(3) Fuel assemblies fully complying with any of the four groups of parameters are suitable for storage in the NUHOMS®-61BT DSC.

Table K.2-3
BWR Fuel Assembly Design Characteristics[(1) (2) (3)]

Transnuclear, ID	7 x 7-49/0	8 x 8-63/1	8 x 8-62/2	8 x 8-60/4	8 x 8-60/1	9 x 9-74/2	10x10-92/2
GE Designations	GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12
Max Length (in)	176.2	176.2	176.2	176.2	176.2	176.2	176.2
Max Width (in) (excluding channels)	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Channel Internal Width (in)	5.278	5.278	5.278	5.278	5.278	5.278	5.278
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Number of Fuel Rods	49	63	62	60	60	66 – Full 8 – Partial	78 – Full 14 - Partial
Number of Water Holes	0	1	2	4	1	2	2

(1) Any fuel channel thickness from 0.065 to 0.120 inch is acceptable on any of the fuel designs.

(2) Maximum fuel assembly weight with channel is 705 lb.

(3) Maximum Co-59 content in the Top End Fitting Region is 4.5 gm per assembly.
Maximum Co-59 content in the Plenum Region is 0.9 gm per assembly.
Maximum Co-59 content in the In-Core Region (including the whole fuel channel) is 4.5 gm per assembly.
Maximum Co-59 content in the Bottom Region is 4.1 gm per assembly.

Table K.2-4
BWR Fuel Assembly Poison Material Design Requirements

Boron-Aluminum Alloy or Boralyn®

NUHOMS®-61BT DSC Type	Maximum Lattice Average Enrichment[1] (wt. % U-235)	Minimum B10 Content in Poison Plates (g/cm2)	% Credit of B10 used in Critically Calculation	Poison Material[2] Coupon Testing
A	3.7	0.021	90	Neutron Transmission plus Radiography
B	4.1	0.032	90	Neutron Transmission plus Radiography
C	4.4	0.040	90	Neutron Transmission plus Radiography

Boral®[2] or Metamic®[2]

NUHOMS®-61BT DSC Type	Maximum Lattice Average Enrichment[1] (wt. % U-235)	Minimum B10 Content in Poison Plates (g/cm2)	% Credit of B10 used in Critically Calculation
A	3.7	0.025	75
B	4.1	0.038	75
C	4.4	0.048	75

(1) Maximum pin enrichment is 5% U235 in all cases.
(2) See Section K.9.1.7 for poison material acceptance listing requirements.

Table K.2-5
Summary of Canister Load Combinations

LOAD CASE	Horizontal DW		Vertical DW		Internal Pressure(9)	External Pressure	Thermal Condition	Liting Loads	Other Loads	Service Level	Enveloped By
	61BT DSC	Fuel	61BT DSC	Fuel							
Non-Operational Load Cases											
NO-1 Fab. Leak Testing	--	--	--	--	--	14.7 psi (101kpa)	70°F (21°C)	--	155 kip axial (689KN)	Test	
NO-2 Fab. Leak Testing	--	--	--	--	12 psi (83kpa)	--			155 kip axial	Test	
NO-3 DSC Uprighting	x	--	--	--	--	--	70°F	x	--	A	
NO-4 DSC Vertical Lift	--	--	x	--	--	--	70°F	x	--	A	
Fuel Loading Load Cases											
FL-1 DSC/Cask Filling	--	--	Cask	--	--	Hydrostatic	100°F Cask (38°C)	x	x	A	DD-2
FL-2 DSC/Cask Filling	--	--	Cask	--	Hydrostatic	Hydrostatic	100°F Cask	x	x	A	DD-2
FL-3 DSC/Cask Xfer	--	--	Cask	--	Hydrostatic	Hydrostatic	100°F Cask	--	--	A	
FL-4 Fuel Loading	--	--	Cask	x	Hydrostatic	Hydrostatic	100°F Cask	--	--	A	
FL-5 Xfer to Decon	--	--	Cask	x	Hydrostatic	Hydrostatic	100°F Cask	--	--	A	
FL-6 Inner Cover plate Welding	--	--	Cask	x	Hydrostatic	Hydrostatic	100°F Cask	--	--	A	
FL-7 Fuel Deck Seismic Loading	--	--	Cask	x	Hydrostatic	Hydrostatic	100°F Cask	--	Note 10	C	
Draining/Drying Load Cases											
DD-1 DSC Blowdown	--	--	Cask	x	Hydrostatic + 20 psi (138kpa)	Hydrostatic	100°F Cask	--	--	A	DD-2
DD-2 Vacuum Drying	--	--	Cask	x	0 psia	Hydrostatic + 14 psi (97kpa)	100°F Cask	--	--	A	
DD-3 Helium Backfill	--	--	Cask	x	12 psi (83kpa)	Hydrostatic	100°F Cask	--	--	A	
DD-4 Final Helium Backfill	--	--	Cask	x	3.5 psi (24kpa)	Hydrostatic	100°F Cask	--	--	A	DD-3
DD-5 Outer Cover Plate Weld	--	--	Cask	x	3.5 psi (24kpa)	Hydrostatic	100°F Cask	--	--	A	DD-3
Transfer Trailer Loading											
TL-1 Vertical Xfer to Trailer	--	--	Cask	x	10 psi (69kpa)	--	0°F Cask (-17°C)	--	--	A	
TL-2 Vertical Xfer to Trailer	--	--	Cask	x	10 psi	--	100°F Cask	--	--	A	
TL-3 Laydown	Cask	X	--	--	10 psi	--	0°F Cask	--	--	A	TR-1-TR-4
TL-4 Laydown	Cask	X	--	--	10 psi	--	100°F Cask	--	--	A	TR-5-TR-6

Table K.2-5
Summary of Canister Load Combinations
(Continued)

LOAD CASE	Horizontal DW 61BT DSC	Horizontal DW Fuel	Vertical DW 61BT DSC	Vertical DW Fuel	Internal Pressure(9)	External Pressure	Thermal Condition	Handling Loads	Other Loads	Service Level	Enveloped By
Transfer to/from ISFSI											
TR-1 Axial Load - Cold	Cask	X	--	--	10.0 psi (70kpa)	--	0°F (-17°C)	1g Axial	--	A	
TR-2 Transverse Load - Cold	Cask	X	--	--	10.0 psi	--	0°F	1g Transverse	--	A	
TR-3 Vertical Load - Cold	Cask	X	--	--	10.0 psi	--	0°F	1g Vertical	--	A	
TR-4 Oblique Load - Cold	Cask	X	--	--	10.0 psi	--	0°F	½ g Axial + ½ g Trans + ½ g Vert.	--	A	
TR-5 Axial Load - Hot	Cask	X	--	--	10.0 psi	--	100°F	1g Axial	--	A	
TR–6 Transverse Load - Hot	Cask	X	--	--	10.0 psi	--	100°F	1g Trans.	--	A	
TR-7 Vertical Load - Hot	Cask	X	--	--	10.0 psi	--	100°F	1g Vertical	--	A	
TR-8 Oblique Load - Hot	Cask	X	--	--	10.0 psi	--	100°F	½ g Axial + ½ g Trans + ½ g Vert.	--	A	
TR-9 25g Corner Drop	Note 1		--	--	20 psi	--	100°F(2)		25g Corner Drop	D	
TR-10 75g Side Drop	Note 1		--	--	20 psi	--	100°F(2)		75g Side Drop	D	
TR-11 Top or Bottom End Drops			Note 12		20 psi	--	100°F(2)		75g End Drop	D	

Table K.2-5
Summary of Canister Load Combinations
(Continued)

HSM LOADING	Horizontal DW		Vertical DW		Internal Pressure[9]	External Pressure[9]	Thermal Condition	Handling Loads	Other Loads	Service Level	Enveloped By:
	61BT DSC	Fuel	61BT DSC	Fuel							
LD-1 Normal Loading - Cold	Cask	X	--	--	10.0 psi (69kpa)	--	0°F Cask (-17°C)	+80 Kip (356KN)	--	A	LD-4
LD-2 Normal Loading – Hot	Cask	X	--	--	10.0 psi	--	100° F Cask	+80 Kip	--	A	LD-5
LD-3	Cask	X	--	--	10.0 psi	--	125° F w/shade[5]	+80 Kip	--	A	LD-2
LD-4 Off-Normal Loading – Cold	Cask	X	--	--	20.0 psi	--	0° F Cask	+80 Kip	FF	B	
LD-5 Off-Normal Loading - Hot	Cask	X	--	--	20.0 psi	--	100° F Cask	+80 Kip	FF	B	
LD-6	Cask	X	--	--	20.0 psi	--	125° F w/shade[5]	+80 Kip	-- FF	B	LD-5
LD-7 Accident Loading	Cask	X	--	--	20.0 psi	--	125° F w/shade[5]	+80 Kip	-- FF	C/D	
HSM STORAGE											
HSM-1 Off-Normal	HSM	X	--	--	10.0 psi	--	-40° F HSM	--	--	B	
HSM-2 Normal Storage	HSM	X	--	--	10.0 psi	--	0° F HSM	--	--	A	HSM-1
HSM-3 Off-Normal	HSM	X	--	--	10.0 psi	--	125° F HSM	--	--	B	
HSM-4 Off-Normal Temp. + Damaged Fuel	HSM	X	--	--	20.0 psi	--	125° F HSM	--	FF	C	
HSM-5 Blocked Vent Storage	HSM	X	--	--	65.0[8] psi	--	125° F HSM/BV[4]	--	--	D	
HSM-6 B.V. + Damaged Fuel Storage	HSM	X	--	--	65.0[8] psi	--	125° F HSM/BV[4]	--	FF	D	
HSM-7 Earthquake Loading – Cold	HSM	X	--	--	10.0 psi	--	0° F HSM	--	FF+EQ	C	
HSM-8 Earthquake Loading – Hot	HSM	X	--	--	10.0 psi	--	100°F HSM	--	FF+EQ	C	
HSM-9 Flood Load (50' H_2O) – Cold	HSM	X	--	--	0 psi	22 psi	0° F HSM	--	Flood[3]	C	
HSM-10 Flood Load (50' H_2O) - Hot	HSM	X	--	--	0 psi	22 psi	100°F HSM	--	Flood[3]	C	

HSM UNLOADING	Horizontal DW		Vertical DW		Internal Pressure[9]	External Pressure[9]	Thermal Condition	Handling Loads	Other Loads	Service Level	Enveloped By:
	61BT DSC	Fuel	61BT DSC	Fuel							
UL-1 Normal Loading - Cold	HSM	X	--	--	10.0 psi	--	0°F HSM	+60 Kip	--	A	UL-4
UL-2 Normal Loading – Hot	HSM	X	--	--	10.0 psi	--	100° F HSM	+60 Kip	--	A	UL-5
UL-3	HSM	X	--	--	10.0 psi	--	125° F w/shade	+60 Kip	--	A	UL-2
UL-4 Off-Normal Loading – Cold	HSM	X	--	--	20.0 psi	--	0° F HSM	+60 Kip	FF	B	
UL-5 Off-Normal Loading - Hot	HSM	X	--	--	20.0 psi	--	100° F HSM	+60 Kip	FF	B	
UL-6	HSM	X	--	--	20.0psi	--	125° F w/shade	+60 Kip	FF	B	UL-5
UL-7 Off. Norm. Unloading-FF/Hot[6,16]	HSM	X	--	--	20.0 psi	--	100° F HSM	+80 Kip	FF	C	
UL-8 Accident Unloading – FF/Hot[7,16]	HSM	X	--	--	65.0[7,8] psi	--	100° F HSM	+80 Kip	FF	D	

HSM UNLOADING / REFLOOD	Horizontal DW		Vertical DW		Internal Pressure[9]	External Pressure[9]	Thermal Condition	Handling Loads	Other Loads	Service Level	Enveloped By:
	61BT DSC	Fuel	61BT DSC	Fuel							
RF-1 DSC Reflood	--	--	Cask	X	20.0 psi (max)	Hydrostatic	100° F Cask	--	--	D	HSM-5&6

Table K.2-5
Summary of Canister Load Combinations
(Concluded)

1. 75g drop acceleration includes gravity effects. Therefore, it is not necessary to add an additional 1.0g load.
2. For Level D events, only maximum temperature case is considered. (Thermal stresses are not limited for level D events and maximum temperatures give minimum allowables).
3. Flood load is an external pressure equivalent to 50 feet (164m) of water.
4. BV = HSM Vents are blocked.
5. At temperature over 100°F (38°C) a sunshade is required over the TC. Temperatures for these cases are enveloped by the 100° F (without sunshade) case.
6. As described in Section K.4.1.2, this pressure assumes release of the fuel cover gas and 30% of the fission gas. Since unloading requires the HSM door to be removed, the pressure and temperatures are based on the normal (unblocked vent) condition. Pressure is applied to the inner pressure boundary.
7. As described in Section K.4.1.2, this pressure assumes release of the fuel cover gas and 30% of the fission gas. Although unloading requires the HSM door to be removed, the pressure and temperatures are based on the blocked vent condition. Pressure is applied to the outer pressure boundary.
8. This pressure is applied to the outer pressure boundary.
9. Unless noted otherwise, pressure is applied to the inner pressure boundary.
10. Fuel deck seismic loads are assumed enveloped by handling loads.
11. Load Cases UL-7 and UL-8 envelop loading cases where the insertion loading of 80 kips (356KN) is considered with an accident pressure (the insertion force is opposed by internal pressure).
12. The 75g top end drop and bottom end drop are not credible events, therefore these drop analyses are not required.

Table K.2-6
Canister Allowable Stress

STRESS CATEGORY	STRUCTURE ALLOWABLE STRESSES[3]	
	Normal Conditions	Accident Conditions
Primary Membrane General P_m	S_m	Lesser of $2.4S_m$ or $0.7\ S_u$ [1]
Local P_L	$1.5\ S_m$	Lesser of $3.6\ S_m$ or $1.0\ S_u$ [1]
Primary Membrane + Bending (P_m or P_L) + P_b	$1.5\ S_m$	Lesser of $3.6\ S_m$ or $1.0\ S_u$ [1]
Range of Primary + Secondary (P_m or P_L) + P_b + Q	$3.0\ S_m$	$2 \times S_a$ for 10 Cycles (Reg. Guide 7.6)
Bearing Stress	S_y	S_y for Seal Surface S_u Elsewhere
Buckling[2]	Factor of Safety = 2.0 Code Case N-284	Factor of Safety = 1.34 Code Case N-284
Pure Shear Stress	$0.6\ S_m$	$0.42\ S_u$
Fatigue	Usage Factor ≤ 1	Not Applicable

Notes:
1. When evaluating the results from the nonlinear elastic plastic analysis for the accident conditions, the general primary membrane stress intensity, P_m, shall not exceed $0.7\ S_u$ and the maximum primary stress intensity at any location (P_L or $P_L + P_b$) shall not exceed $0.9\ S_u$. These limits are in accordance with Appendix F of Section III of the Code.
2. Other acceptable criteria are also provided in Section III of the ASME Code and NUREG/CR-6322.
3. Reference to Section III, Subsection NB, Para. NB-3223 and NB-3224 for Level B and Level C stress limits.

Table K.2-7
Basket Stress Limits

STRESS CATEGORY	ALLOWABLE STRESSES[6]	
	Normal Conditions[1]	Accident Conditions[2]
Primary Membrane General P_m	S_m	Lesser of 2.4 S_m or 0.7 S_u [3]
Local P_L	1.5 S_m	Lesser of 3.6 S_m or 1.0 S_u [3]
Primary Membrane + Bending (P_m or P_L) + P_b	1.5 S_m	Lesser of 3.6 S_m or 1.0 S_u [3]
Range of Primary + Secondary (P_m or P_L) + P_b + Q	3.0 S_m	2S_a for 10 cycles [4]
Bearing Stress	S_y	Not applicable
Average. Primary Shear Stress	0.6 S_m	0.42 S_u
Buckling[7]	Compressive Stress limit per NF-3322.1(c)	100% of the plastic analysis collapse load or test collapse load[5]
Fatigue	Cumulative fatigue usage factor $\leq$ 1	Not applicable

Notes:
1. ASME Code, Section III, Appendix NG, service level A
2. ASME Code, Section III, Appendix F, service level D
3. When evaluating the results from the nonlinear elastic-plastic analysis for the accident conditions, the general primary membrane stress intensity, P_m, shall not exceed 0.7S_u and the maximum primary stress intensity at any location (P_L or P_L + P_b) shall not exceed 0.9 S_u.
4. ASME Code Section III, Appendix 1 and Reg. Guide 7.6.
5. ASME Code, Section III, Appendix F, Para. F-1341.3
6. Reference to Section III, Subsection NG, Para. NG-3223 and NG-3224 for Level B and Level C Stress Limits.
7. Other acceptable criteria are also provided in Section III of the ASME Code and NUREG/CR-6322.

Table K.2-8
Classification of NUHOMS®-DSC Components

IMPORTANT TO SAFETY	NOT IMPORTANT TO SAFETY
Canister Assembly	Siphon tube
	Quick connect coupling
Canister shell	Male connector
Bottom shield plug	Alignment key
Inner bottom cover	Canister lifting lug
Outer bottom cover	Electroless nickel coating
Grapple ring and support	
Top shield plug	
Inner top cover plate	
Outer top cover plate	
Siphon/vent port cover plate	
Siphon vent block	
Support ring segment	
Test port plug	
Weld filler metal	
Storage Basket Assembly	
Fuel compartment	
Fuel compartment wrap	
Poison plate	
Basket plate	
Weld Stud, washer, hex nut	
Basket plate insert	
Basket rail	
Basket holddown plate	
Spacer pad	
Alignment leg	
Weld filler metal	

Table K.2-9
Additional Design Criteria for NUHOMS®-61BT DSC

IMPORTANT TO SAFETY	NOT IMPORTANT TO SAFETY
The gross weight of the NUHOMS®-61BT DSC:	88.5 kips
NUHOMS®-61BT DSC Type:	A, B or C
Payload Capacity:	61 intact BWR assemblies 61 BWR assemblies (up to 16 damaged and remainder intact) (acceptable assemblies listed in Table K.2-3)
Spent Fuel Characteristics:	See Tables K.2-1, K.2-2, K.2-4

Table K.2-10
Summary of NUHOMS®-61BT Component Design Loadings[(1)]

Component	Design Load Type	FSAR Section Reference	Design Parameters	Applicable Codes
61BT-DSC:	---	---	---	ASME Code, 1998 Edition with 1999 Addenda, Section III, Subsection NB and Appendix F (Shell) and Subsections NG, NF and Appendix F (Basket) with exceptions noted in Table K.3.1-2.
	Flood	K2.2.2	Maximum water height: 50 ft.	10CFR72.122(b)
	Seismic	K.2.2.3	Horizontal ground acc.: 0.25g Vertical ground acc.: 0.17g	NRC Reg. Guides 1.60 & 1.61
	Dead Load	K.3.6.1.2 Table K.3.2-1	Weight of loaded DSC: 88,390 lbs.	ANSI 57.9-1984
	Normal and Off-Normal Pressure	K.3.6.1.2 Table K.3.4-5	Enveloping internal pressure of ≤10.0 psig (Normal) and < 20 psig (Off-Normal)	10CFR72.122(h)
	Test Pressure	K.3.6.1.2	Enveloping internal pressure of 12 psig applied w/o DSC outer top cover plate	10CFR72.122(h)
	Normal and Off-Normal Operating Temperature	K.3.6.1.2 K.3.6.2.2 K.4.4 K.4.5	DSC with spent fuel rejecting 18.3 kW (BWR) decay heat. Ambient air temperature -40°F to 125°F	ANSI 57.9-1984
	Normal Handling Loads	K.3.6.1.2 Table K.2-5	1. Hydraulic ram load of: 80,000 lb.(DSC HSM insertion) 60,000 lb (DSC HSM extraction) 2. Transfer (to/from ISFSI) loads of: 2a. +/-1.0g axial 2b. +/-1.0g transverse 2c. +/-1.0g vertical 2d. +/-0.5g axial +/-0.5g transverse +/-0.5g vertical	ANSI 57.9-1984
	Off-Normal Handling Loads	K.3.6.1.2 Table K.2-5	Hydraulic ram load of: 80,000 lb. (DSC HSM insertion) 80,000 lb (DSC HSM extraction)	ANSI-57.9-1984

Table K.2-10
Summary of NUHOMS®-61BT Component Design Loadings[1]

(Continued)

Component	Design Load Type	FSAR Section Reference	Design Parameters	Applicable Codes
	Accidental Cask Drop Loads	K.3.7.5	Equivalent static deceleration of 75g for vertical end drop and horizontal side drops, and 25g oblique corner drop	10CFR72.122(b)
	Accident Internal Pressure	K.4	Enveloping internal pressure of ≤65 psig based on 100% fuel cladding rupture and fill gas release, 30% fission gas release, and ambient air temperature of 125°F	10CFR72.122(h)
61BT-DSC Steel Support Structure:	---	---	---	AISC Specification for Structural Steel Buildings
	Dead Weight	8.1.1.4	Loaded DSC plus self weight	ANSI-57.9-1984
	Seismic	3.2.3	DSC reaction loads with horizontal ground acc. of 0.25g and vertical ground acc. of 0.17g	NRC Reg. Guides 1.60 & 1.61
	Normal Handling Loads	K.3.6.1.1 K.3.6.1.4	Friction load of 25,633 lbs applied to both rails for support structure evaluation.	ANSI-57.9-1984
	Off-normal Handling Loads	K.3.6.2.1 K.3.6.1.4	For steel support structure evaluation, this load is 80,000 lbs plus a vertical load of 22,100 lbs applied to each rail, one rail at a time.	ANSI-57.9-1984

Notes:

(1) The design criteria for the HSM (including the DSC Steel Support Structure) and the TC remain unchanged from Table 3.2-1. However, these components have been evaluated for the effect of the higher weight of the 61BT-DSC.