Enclosure 3

Public Versions of Standardized NUHOMS® System UFSAR, Revision 13, Cover Page, List of Effective Pages, and Proprietary Replacement Pages and Drawings

Binder 2 of 2

transition rails support the fuel tubes and transfer mechanical loads to the DSC shell. They also provide the thermal conduction path from the basket assembly to the canister shell wall, making the basket assembly efficient in rejecting heat from its payload. The nominal clear dimension of each fuel tube opening is sized to accommodate the limiting assembly with sufficient clearance around the fuel assembly.

The 24PTH DSC basket geometry and the materials used for its fabrication are shown on drawings NUH-24PTH-1003-SAR and NUH-24PTH-1004-SAR, included in Section P.1.5.

During dry storage of the spent fuel in the NUHOMS®-24PTH system, no active systems are required for the removal and dissipation of the decay heat from the fuel. The NUHOMS®-24PTH DSC is designed to transfer the decay heat from the fuel to the canister body via the basket and ultimately to the ambient via either the HSM-H in storage mode or the TCs in the transfer mode.

Each canister is identified by a Mark Number, **W-24PTH-X-Y-Z**, where:

W is user specific designations;

X refers to the DSC Type as described previously (X = S or L or S-LC);

Y refers to the basket type (1A, 2A, 1B, 2B, 1C or 2C) and

Z is a number corresponding to a specific canister.

P.1.2.1.2 NUHOMS®-HSM-H Module

The Standardized HSM Model 102 is described in Chapter 1 and in the drawings included in Appendix E of the FSAR.

The HSM-H module design is similar to the design of HSM Model 102 with the following features provided to improve the heat rejection and shielding capabilities:

- Use of a thicker roof with no uniform gap between the adjacent modules,
- Use of slotted plates and holes in the DSC support rails to increase airflow at the bottom portion of canister,
- Increased height of the module to increase module cavity and stack height and to minimize air flow resistance in the module cavity,
- Optimized DSC support structure to minimum airflow resistance.
- Use of finned side heat shields option for high heat loads to improve convection heat transfer by increasing surface area of heat shield, and
- Use of louvered top heat shield to minimize airflow resistance.

P.1-3 provides an overview of the module.

P.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

P.1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry, soluble boron in spent fuel pool and by utilizing fixed neutron poison material in the fuel basket. During storage, with the DSC cavity is 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®-24PTH DSC and the assurance that no water can enter the DSC cavity during storage.

P.1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the NUHOMS®-24PTH system.

P.1.2.2.3.3 Operation Shutdown Modes

The NUHOMS®-24PTH DSC system is a totally passive system so that consideration of operation shutdown modes is unnecessary.

P.1.2.2.3.4 Instrumentation

No change *to Section 5.1.3.4.*

P.1.2.2.3.5 Maintenance Techniques

No change *to Section 5.1.3.5.*

P.1.2.3 Cask Contents

The NUHOMS®-24PTH DSC system is designed to store 24 intact (or up to 12 damaged and remaining intact) PWR fuel assemblies with or without control components. The fuel that may be stored in the NUHOMS®-24PTH DSC is presented in Chapter P.2.

Chapter P.3 provides the structural analysis. Chapter P.4 includes the thermal analysis. Chapter P.5 provides the shielding analysis. Chapter P.6 covers the criticality safety of the NUHOMS®-24PTH DSC system and its contents, listing material densities, moderator ratios, and geometric configurations.

P.1.4 Generic Cask Arrays

No change *to Section 1.2.1.*

P.1.5 Supplemental Data

The following Transnuclear drawings are enclosed:

1. NUHOMS®-24PTH Transportable Storage DSC, for PWR Fuel, Main Assembly NUH24PTH-1001-SAR.

2. NUHOMS®-24PTH Transportable Storage DSC, for PWR Fuel, Shell Assembly, NUH24PTH-1002-SAR.

3. NUHOMS®-24PTH Transportable Storage DSC, for PWR Fuel Basket Assembly, NUH24PTH-1003-SAR.

4. NUHOMS®-24PTH Transportable Storage DSC, for PWR Fuel, Transition Rails, NUH24PTH-1004-SAR.

5. Standardized NUHOMS® ISFSI HSM-H, Main Assembly, NUH-03-7001-SAR.

6. General License NUHOMS® ISFSI OS197FC Onsite Transfer Cask Main Assembly, NUH-03-8006-SAR.

7. *Standardized NUHOMS® ISFSI, HSM-H/HSM-HS Dose Reduction Hardware, NUH-03-7004-SAR.*

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
5	REVISED PER FCN 721004-1063	01/08/14
4	REVISED PER FCN 721004-821	01/26/12
3	REVISED PER FCN 721004-452 AND 721004-655; UPDATED TITLE BLOCK	01/28/10
2	INCORPORATED FCN 721004-276 AND 721004-294, UPDATED LOGO	01/30/06

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 24PTH TRANSPORTABLE STORAGE DSC FOR PWR FUEL MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH24PTH-1001-SAR	NONE	1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-1001-SAR	4 OF 4	5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
4	REVISED PER FCN 721004-1063	01/08/14
3	REVISED PER FCNs 721004-821, 721004-828 AND 721004-855	01/26/12
2	REVISED PER FCN 721004-452; UPDATED TITLE BLOCK	01/28/10
1	INCORPORATED FCN 721004-276 AND 721004-294, UPDATED LOGO	01/30/06

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 24PTH TRANSPORTABLE STORAGE DSC FOR PWR FUEL BASKET ASSEMBLY

DRAWING NO. NUH24PTH-1003-SAR | SCALE NONE | SHEET 1 OF 7

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-1003-SAR	2 OF 7	4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH24PTH-1003-SAR 5 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-1003-SAR	5 OF 7	4

NUH24PTH-1003-SAR 6 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-1003-SAR	6 OF 7	4

NUH24PTH-1003-SAR 7 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-1003-SAR	7 OF 7	4

NUH24PTH-1004-SAR 1 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
4	REVISED PER FCN 721004-1063	01/08/14
3	REVISED PER FCNs 721004-821 AND 721004-856; UPDATED TITLE BLOCK	01/26/12
2	INCORPORATED FCN 721004-276 AND 721004-294, UPDATED LOGO	01/30/06

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 24PTH TRANSPORTABLE STORAGE DSC FOR PWR FUEL TRANSITION RAILS

DRAWING NO.	SCALE	SHEET
NUH24PTH-1004-SAR	NONE	1 OF 2

NUH24PTH-1004-SAR 2 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-1004-SAR	2 OF 2	4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
5	REVISED PER FCNS 721004-1161 AND 721004-1231	01/08/14
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3	INCORPORATED FCNS 721004-352, 721004-370, 721004-447 AND 721004-439 UPDATED TITLE BLOCK	01/29/08
2	INCORPORATED FCNS 721004-278, 721004-206 AND 721004-313, UPDATED LOGO	01/31/06

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
STANDARDIZED NUHOMS® ISFSI
HSM-H
MAIN ASSEMBLY

DRAWING NO. NUH-03-7001-SAR | SCALE NONE | SHEET 1 OF 11

DRAWING NO. NUH-03-7001-SAR SHEET 1 OF 11

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7001-SAR	3 OF 11	5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7001-SAR	4 OF 11	5

NUH-03-7001-SAR 5 OF 11

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-03-7001-SAR | SHEET 5 OF 11 | REVISION 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7001-SAR	6 OF 11	5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-03-7001-SAR SHEET 8 OF 11

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH-03-7001-SAR 10 OF 11

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7001-SAR	10 OF 11	5

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER FCN 721004-1063, UPDATED TITLE BLOCK	01/08/14
0	FOR NRC REVIEW AMENDMENT 8	–

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
GENERAL LICENSE NUHOMS® ISFSI
OS197-FC ONSITE TRANSFER CASK
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH-03-8006-SAR	NONE	1 OF 1

The 24PTH DSC basket is designed with 2 alternate options: Type 1 basket, which includes aluminum inserts in the R45 transition rails, and Type 2 basket which does not include any aluminum inserts. Type 1 basket is the preferred option for canisters with high decay heat loads, since the aluminum inserts allow a more direct heat conduction path from the basket edge to the DSC shell. Type 2 basket offers the advantage of an adequate thermal performance but with a lower lifting weight requirement.

The NUHOMS®-24PTH DSC basket is designed with three alternate poison materials: Borated Aluminum alloy, Boron Carbide/Aluminum Metal Matrix Composite (MMC) and Boral®. For criticality analysis, 90% of B10 content present in the borated aluminum and MMC poison plates is credited, while only 75% is credited for Boral®.

For each poison material, the NUHOMS®-24PTH DSC basket is analyzed for six alternate basket configurations, depending on the boron loadings analyzed (designated as "A" basket for low B10 loading, "B" basket for moderate B10 loading, and "C" basket for high B10 loading) and Basket-Type (Type 1 or Type 2).

A summary of the alternate poison loadings considered and the corresponding credit taken in the criticality analysis for each poison material as a function of basket types is presented below:

Poison Type	24PTH Basket Type(1)	Poison Loading (B10 mg/cm^2)	% Credit Used in Criticality Analysis
Borated Aluminum Alloy/MMC	1A or 2A	7	90
	1B or 2B	15	
	1C or 2C	32	
Boral®	1A or 2A	9	75
	1B or 2B	19	
	1C or 2C	40	

(1) Type 1A = Basket Type 1 with aluminum inserts in the R45 transition rails and Type A poison plate configuration;
Type 2A = Basket Type 2 without aluminum inserts in the R45 transition rails and Type A poison plate configuration;

Table P.2-6 through Table P.2-9 define the minimum required cooling time after reactor discharge for a fuel assembly without CCs for a given assembly heat load, burnup, and maximum initial enrichment parameters. These tables ensure that the fuel assembly decay heat load is less than that specified for each table and that the corresponding radiation source term is bounded by that analyzed in Chapter P.5. Similarly, Table P.2-10 through Table P.2-13 defines the minimum required cooling time after reactor discharge for a fuel assembly with CCs.

The NUHOMS®-24PTH DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with a CC is 1682 lbs.

The maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-24PTH DSC per Interim Staff Guidance (ISG) No. 11, Revision *3* [2.5]. In addition, ISG-11 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

All changes on this page are AMD 11

P.2.1.1 General Operating Functions

No change *to Section 3.1.2.*

evaluation as recommended in NUREG-0800, Section 3.5.3 [2.7]. The results of these evaluations are reported in Section P.11.

The evaluation of tornado-generated missile loads on the transfer cask summarized in Section 8.2 of the FSAR remains unchanged.

P.2.2.2 Water Level (Flood) Design

No change *to Section 3.2.2.*

P.2.2.3 Seismic Design

The seismic design criteria for the HSM-H is consistent with the criteria set forth in Section 3.2.3, with the exception that the NRC Regulatory Guide 1.60 (R.G. 1.60) [2.11] response spectra is anchored to a maximum ground acceleration of 0.30g (instead of 0.25g) for the horizontal components and 0.20g (instead of 0.17g) for the vertical component. The results of the frequency analysis of the HSM-H structure (which includes a simplified model of the DSC) yield a lowest frequency of 23.2 Hz in the transverse direction and 28.4 Hz in the longitudinal direction. The lowest vertical frequency exceeds 33 Hz. Thus, based on the R.G. 1.60 response spectra amplifications, the corresponding seismic accelerations used for the design of the HSM-H are 0.37g and 0.33g in the transverse and longitudinal directions respectively and 0.20g in the vertical direction. The corresponding accelerations applicable to the DSC are 0.41g and 0.36g in the transverse and longitudinal directions, respectively, and 0.20g in the vertical direction. The seismic analysis of the HSM-H and 24PTH DSC are further discussed in Section P.3.7.

The seismic design criteria for the TC and HSM Model 102 does not change from that documented in Section 8.2. Therefore, even though the HSM-H and 24PTH DSCs are analyzed for 0.3g horizontal and 0.2g vertical seismic loads, the seismic design criteria for the 24PTH system is still limited to the criteria documented in Section 8.2.

P.2.2.4 Snow and Ice Loading

No change *to Section 3.2.4.*

P.2.2.5 Combined Load Criteria

The NUHOMS®-24PTH system is subjected to the same types of loads as the existing NUHOMS®-24P or -52B System. The load combination criteria for the TCs for transfer are the same as those shown in the FSAR Table 3.2-7. The criteria applicable to the NUHOMS®-24PTH DSC and HSM-H are discussed in the following subsections.

P.2.2.5.1 NUHOMS®–24PTH DSC Structural Design Criteria

The NUHOMS®-24PTH DSC is designed using the ASME Boiler and Pressure Vessel Code [2.2] criteria given in the existing FSAR, Chapter 3, except as noted in the following sections. A summary of the NUHOMS®-24PTH DSC load combinations is presented in Table P.2-14.

A. Under axial loads, the DSC shell and transfer cask provide overall/global stability to the 24PTH basket structure. Thus, only local stability effects are specifically addressed.

For axial compression loads, stability criteria for the fuel compartment tubes are based on NF-3322.1(c)(2) (for austenitic members). Using a span length of 24.0 in, corresponding to the maximum distance between basket straps and a value of K=1.0 (pinned-pinned condition) a slenderness ratio (KL/ r) of 6.42 is calculated.

Application of elastic stability criteria to the fuel tubes is conservative as the low value of KL/ r indicates elastic buckling is not a likely failure mechanism. In addition, buckling is restricted by the adjacent tubes, transition rails, and DSC shell.

In addition, the width to thickness ratio of the tube wall is checked using NF-3322.2(d)(2)(b)(1) to verify that the tubes are fully effective in compression.

B. Under lateral loads, stability of the basket tube structure is demonstrated using hand calculations to evaluate the fuel compartment tubes "ligaments" as columns using the stability criteria of NF-3322.1(c)(2) for stainless steel compression members.

All changes on this page are AMD 11

Accident Conditions

Accident Condition Stress Criteria for Steel Elements

As summarized in Table P.2-16 the accident condition (Level D) stress criteria for the fuel support structure and the welded steel transition rails is based on Appendix F of the ASME Code, Section III. Criteria are provided for both linear elastic and elastic-plastic stress analyses.

Accident Condition Criteria for R90 Aluminum Transition Rails

For accident condition loading (i.e., the postulated drops), the R90 aluminum transition rail bodies must support the fuel tubes such that stresses and displacements in the fuel compartment tubes are acceptable. Since, the rail bodies are captured between the fuel compartment tube and the DSC shell, large displacements of the rails are prevented. Thus, no additional checks (of the aluminum) are required for accident/drop loading. Qualification of the fuel tubes demonstrates that the R90 rails perform their intended function.

Accident Condition Stability Criteria

Similar to the normal condition evaluations, stability criteria are addressed in two parts:

A. Accident condition axial stresses in the fuel compartment tubes are evaluated using the equation from F-1334.3(b)(1)[2.2] loads, stability of the basket structure is demonstrated using detailed finite element models and the Collapse Load criteria from F-1341.3 [2.2]. These criteria establish the allowable load as 90% of the Limit Analysis Collapse Load where the Limit Analysis Collapse Load is the maximum load determined using elastic-perfectly plastic material properties with a yield stress equal to the lesser of $2.3S_m$ or $0.7S_u$.

This load is described in Section P.2.2.1. The design pressures for the tornado wind load are shown in Table P.2-19.

(C) Flood Load (FL)

No change *to Section 3.2.2.*

(D) Earthquake Load (EQ)

The HSM-H is evaluated for amplified ground accelerations resulting from a design basis earthquake defined in NRC R.G. 160 [2.11] anchored at 0.30g horizontal and 0.2g vertical accelerations.

P.2.3 Safety Protection Systems

P.2.3.1 General

The NUHOMS®-24PTH DSC is designed to provide storage of spent fuel for at least 40 years. The DSC cavity is inerted and backfilled with helium and the internal pressure is always above atmospheric during the storage period as a precaution against 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 DSC cavity gas cannot escape.

Only those features that are not addressed in the existing FSAR, 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®-24PTH DSC that are "Important to Safety" and "Not Important to Safety" are listed in Table P.2-17.

P.2.3.2 Protection By Multiple Confinement Barriers and Systems

The NUHOMS®-24PTH DSC provides a leak tight confinement of the spent fuel. Although similar to the existing NUHOMS®-24P DSC, sealing of the NUHOMS®-24PTH DSC involves leak testing to the criteria of ANSI N14.5 [2.4] after loading and sealing the canister, as described in Section P.7.

P.2.3.3 Protection By Equipment and Instrumentation Selection

No change *to Section 3.3.3.*

P.2.3.4 Nuclear Criticality Safety

P.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 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, soluble boron in the pool 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 criteria of the plates is described in Section P.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 borated pool water. The method of criticality control is in accordance with the requirements of 10CFR72.124 [2.10].

The criticality analyses performed for the 24PTH system are described in Section P.6.

P.2.3.4.2 Error Contingency Criteria

Provision for error contingency is built into the criterion used in Section P.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.

P.2.3.4.3 Verification Analysis-Benchmarking

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

P.2.3.5 Radiological Protection

No change *to Section 3.3.5.*

P.2.3.6 Fire and Explosion Protection

No change *to Section 3.3.6.*

P.2.4 Decommissioning Considerations

No change *to Section 3.5.*

P.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, NG and NF, 1998 edition including 2000 Addenda.

2.3 Young, W.C., "Roark's Formulas for Stress and Strain," 6th Edition, McGraw-Hill Book Company, New York, 1989.

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

2.5 Interim Staff Guidance No. 11, Revision *3*, "Cladding Considerations for the Transportation and storage of Spent Fuel," dated *November 17, 2003*.

2.6 "Design Basis Tornado for Nuclear Power Plants," Regulatory Guide 1.76, U.S. Atomic Energy Commission, April 1974.

2.7 "Missiles Generated by Natural Phenomenon," Standard Review Plan, NUREG-0800, U.S. Nuclear Regulatory Commission.

2.8 American Society of Civil Engineers, ASCE 7-95, "Minimum Design Loads for Buildings and Other Structures" (formerly ANSI A58.1).

2.9 ANSI/ANS 57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," American Nuclear Society.

2.10 Title 10, Code of Federal Regulations, Part 72 (10CFR72), "Licensing Requirements for the Storage of Spent Fuel in the Independent Spent Fuel Storage Installation," U.S. Nuclear Regulatory Commission, August 31, 1988.

2.11 Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," U.S. Atomic Energy Commission, Revision 1, December 1973.

2.12 Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," U.S. Atomic Energy Commission, October 1973.

2.13 Bechtel Topical Report, "Design of Structures for Missile Impact," BC-TOP-9-A, Revision 2, September 1974.

All changes on this page are AMD 11

All changes on this page are AMD 11

Table P.2-1
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-24PTH DSC

PHYSICAL PARAMETERS:	
Fuel Class	*Intact or damaged unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table P.2-3. Equivalent reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table P.2-3 is also acceptable.*
Fuel Damage	*Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel assembly needs to be handled by normal means.*
Partial Length Shield Assemblies (PLSAs)	*WE 15x15 class PLSAs which have only ever been irradiated in peripheral core locations with following characteristics are authorized:* • *Maximum burnup, 40 GWd/MTU* • *Minimum cooling time, 6.5 years* • *Maximum decay heat, 900 watts.*
Reconstituted Fuel Assemblies: • *Maximum Number of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods* • *Maximum Number of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly* • *Maximum Number of Reconstituted Assemblies per DSC with unlimited number of low enriched UO_2 rods and/or Unirradiated Stainless Steel Rods and/or Zr Rods or Zr Pellets*	*4* *10* *24*
Control Components (CCs)	• *Up to 24 CCs are authorized for storage in 24PTH-L, 24PTH-S, and 24PTH-S-LC DSCs only.* • *Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Assembly Rods (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources.* • *Design basis thermal and radiological characteristics for the CCs are listed in Table P.2-2.*
Nominal Assembly Width	*8.536 inches*
Number of Intact Assemblies	*≤24*
Number and Location of Damaged Assemblies	*Maximum of 12 damaged fuel assemblies. Balance may be intact fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration.* *Damaged fuel assemblies are to be placed in Location A and/or B as shown in Figure P.2-6. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.*

Table P.2-1

PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-24PTH DSC (concluded)

All changes on this page are AMD 11

Maximum Assembly plus CC Weight	*1682 lbs*
THERMAL/RADIOLOGICAL PARAMETERS:	
Allowable Heat Load Zoning Configurations for each 24PTH DSC	*Per Figure P.2-1 or Figure P.2-2 or Figure P.2-3 or Figure P.2-4 or Figure P.2-5.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 1 (Without CCs)	*Per Table P.2-6 for Zone 1 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 2 (Without CCs)	*Per Table P.2-7 for Zone 2 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 3 (Without CCs)	*Per Table P.2-7 for Zone 2 fuel and Table P.2-8 for Zone 3 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 4 (Without CCs)	*Per Table P.2-9 for Zone 4 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 5 (Without CCs)	*Per Table P.2-8 for Zone 3 fuel and Table P.2-9 for Zone 4 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 1 (With CCs)	*Per Table P.2-10 for Zone 1 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 2 (With CCs)	*Per Table P.2-11 for Zone 2 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 3 (With CCs)	*Per Table P.2-11 for Zone 2 fuel and per Table P.2-12 for Zone 3 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 4 (With CCs)	*Per Table P.2-13 for Zone 4 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 5 (With CCs)	*Per Table P.2-12 for Zone 3 fuel and per Table P.2-13 for Zone 4 fuel.*
Maximum Planar Average Initial Fuel Enrichment	*Per Table P.2-4 or Table P.2-5*
Decay Heat	*Type 1 Basket:* *≤ 40.8 kW for 24PTH-S and 24PTH-L DSCs with decay heat limits for Zones 1, 2, 3 and 4 as specified in Figure P.2-1 or Figure P.2-2 or Figure P.2-3 or Figure P.2-4.*
	Type 2 Basket: *Same as Type 1 Basket except ≤ 31.2 kW/DSC and ≤ 1.3 kW/fuel assembly for 24PTH-S and 24PTH-L DSCs.* *≤ 24.0 kW for 24PTH-S-LC DSC with decay heat limits as specified in Figure P.2-5.*
Minimum Boron Loading	*Per Table P.2-4 or Table P.2-5.*

Table P.2-2

Thermal and Radiological Characteristics for Control Components Stored in the NUHOMS®-24PTH DSC

Parameter	***BPRAs, NSAs, CRAs, RCCAs, VSIs, Neutron Sources and APSRAs***	***TPAs and ORAs***
Maximum Gamma Source (γ/sec/DSC)	*9.3E+14*	*9.8E+13*
Decay Heat (Watts/DSC)	*192.0*	*192.0*

Note: NSAs and Neutron Sources shall only be stored in the interior compartments of the basket. Interior compartments are those compartments that are completely surrounded by other compartments, including the corners. There are four interior compartments in the 24PTH DSC.

All changes on this page are AMD 11

Table P.2-3
PWR Fuel Assembly Design Characteristics for the NUHOMS®-24PTH DSC

Assembly Class		*B&W 15x15*	*WE 17x17*	*CE 15x15*	*WE 15x15*	*CE 14x14*	*WE 14x14*
Maximum Unirradiated Length (in)(1)	*24PTH-S*	*165.75*	*165.75*	*165.75*	*165.75*	*165.75*	*165.75*
	24PTH-L	*171.93*	*171.93*	*171.93*	*171.93*	*171.93*	*171.93*
	24PTH-S-LC	*171.93*	*N/A*(3)	*N/A*(3)	*N/A*(3)	*N/A*(3)	*N/A*(3)
Fissile Material		UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/Assembly(2)		*0.49*	*0.49*	*0.49*	*0.49*(4)	*0.49*	*0.49*
Maximum Number of Fuel Rods		*208*	*264*	*216*	*204*	*176*	*179*
Maximum Number of Guide/ Instrument Tubes		*17*	*25*	*9*	*21*	*5*	*17*

(1) Maximum Assembly + Control Component Length (unirradiated).
(2) The maximum MTU/assembly is based on the shielding analysis. The listed value is higher than the actual.
(3) Not authorized for storage.
(4) The maximum MTU/assembly for WE 15x15 PLSA = 0.33.

All changes on this page are AMD 11

Table P.2-4
Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for the NUHOMS®-24PTH DSC (Intact Fuel)

Fuel Assembly Class	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)*			
	Minimum Soluble Boron (ppm)	*Basket Type*[3]		
		1A or 2A	*1B or 2B*	*1C or 2C*
CE 14x14 [1]	*2100*	*4.50*	*4.90*	*NR*
	2200	*4.60*	*5.00*	*NR*
	2300	*4.70*	*NR*	*NR*
	2400	*4.80*	*NR*	*NR*
	2500	*4.90*	*NR*	*NR*
	2600	*5.00*	*NR*	*NR*
WE 14x14 [2]	*2100*	*4.80*	*5.00*	*NR*
	2200	*4.90*	*NR*	*NR*
	2300	*5.00*	*NR*	*NR*
CE 15x15 [2]	*2100*	*3.90*	*4.20*	*4.60*
	2200	*4.00*	*4.40*	*4.70*
	2300	*4.10*	*4.50*	*4.80*
	2400	*4.20*	*4.60*	*4.90*
	2500	*4.30*	*4.70*	*5.00*
	2600	*4.40*	*4.80*	*NR*
	2700	*4.50*	*4.90*	*NR*
	2800	*4.50*	*5.00*	*NR*
	2900	*4.60*	*NR*	*NR*
	3000	*4.70*	*NR*	*NR*
WE 15x15 [2]	*2100*	*3.80*	*4.20*	*4.60*
	2200	*3.90*	*4.30*	*4.70*
	2300	*4.00*	*4.40*	*4.80*
	2400	*4.10*	*4.50*	*4.90*
	2500	*4.20*	*4.60*	*5.00*
	2600	*4.30*	*4.70*	*NR*
	2700	*4.30*	*4.80*	*NR*
	2800	*4.40*	*4.90*	*NR*
	2900	*4.50*	*5.00*	*NR*
	3000	*4.60*	*NR*	*NR*

All changes on this page are AMD 11

Table P.2-4

Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for the NUHOMS®-24PTH DSC (Intact Fuel)

(concluded)

All changes on this page are AMD 11

Fuel Assembly Class	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)*			
	Minimum Soluble Boron (ppm)	*Basket Type(3)*		
		1A or 2A	*1B or 2B*	*1C or 2C*
WE 17x17 (2)	2100	3.80	4.10	4.50
	2200	3.90	4.20	4.60
	2300	4.00	4.30	4.70
	2400	4.00	4.40	4.80
	2500	4.10	4.50	4.90
	2600	4.20	4.60	5.00
	2700	4.30	4.70	NR
	2800	4.40	4.80	NR
	2900	4.50	4.90	NR
	3000	4.60	5.00	NR
B&W 15x15 (2)	2100	3.60	4.00	4.30
	2200	3.70	4.10	4.50
	2300	3.80	4.20	4.60
	2400	3.90	4.30	4.70
	2500	4.00	4.40	4.80
	2600	4.10	4.50	4.90
	2700	4.20	4.60	5.00
	2800	4.20	4.70	NR
	2900	4.30	4.80	NR
	3000	4.40	4.90	NR

Notes:

(1) When CCs that extend into the active fuel region are stored, the maximum planar average initial enrichment shall be reduced by 0.2 wt. %.

(2) When CCs that extend into the active fuel region are stored, the maximum planar average initial enrichment shall be reduced by 0.05 wt. % or the soluble boron concentration shall be increased by 50 ppm.

(3) The fixed poison loading requirements as a function of Basket Type are specified in Table P.2-20.

NR = Not Required.

Table P.2-5
Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for the NUHOMS®-24PTH DSC (Damaged Fuel)

Assembly Class	*Maximum Number of Damaged Fuel Assemblies per DSC*	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)*			
		Minimum Soluble Boron (ppm)	*Basket Type*[3]		
			1A or 2A	*1B or 2B*	*1C or 2C*
CE 14x14 [1]	8	2150	NR	4.80	NR
	12	2150	NR	4.70	NR
	12	2450	4.50	5.00	NR
WE 14x14 [2]	12	2150	4.50	5.00	NR
CE 15x15 [2]	12	2150	NR	NR	4.50
	12	2550	NR	NR	5.00
WE 15x15 [2]	8	2150	NR	NR	4.50
	12	2250	NR	NR	4.50
	8	2550	NR	NR	5.00
	12	2650	NR	NR	5.00
B&W 15x15 [2]	12	2350	NR	NR	4.50
	12	2800	NR	NR	5.00
WE 17x17 [2]	12	2250	NR	NR	4.50
	12	2650	NR	NR	5.00

Notes:

(1) When CCs that extend into the active fuel region are stored, the maximum planar average initial enrichment shall be reduced by 0.2 wt. %.

(2) When CCs that extend into the active fuel region are stored, the maximum planar average initial enrichment shall be reduced by 0.05 wt. % or the soluble boron concentration shall be increased by 50 ppm.

(3 The fixed poison loading requirements as a function of Basket Type are specified in Table P.2-20.

NR = Not Required.

All changes on this page are AMD 11

Table P.2-6

PWR Fuel Qualification Table for Zone 1 Fuel with 1.7 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel without CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																															
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34		3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
38		4.5	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
39		4.5	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
40		4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
41		5.0	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
42										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
43										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
44										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
45										4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
46										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
47										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
48										5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0
49													5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
50													5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
51													5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
52														5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5
53														5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
54															5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
55															5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0
56															6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0
57																	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
58																	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5
59																	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5
60																	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
61																	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0
62																	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time.

Note: The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table P.2-7

PWR Fuel Qualification Table for Zone 2 Fuel with 2.0 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel without CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																															
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36		3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
38		3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
39		3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
40		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
41		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
42										3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
43										3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0
44										3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
45										4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
46										4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
47										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
48										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
49													4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5
50													4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
51													4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
52														4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
53														4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
54															4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0
55															5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
56															5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5
57																	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
58																	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
59																	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
60																	6.0	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
61																	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
62																	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time.

Note: *The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.*

All changes on this page are AMD 11

Table P.2-8

PWR Fuel Qualification Table for Zone 3 Fuel with 1.5 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel without CCs)

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

BU GWd/MTU	*Assembly Average Initial Enrichment (wt. % U-235)*																															
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34		4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36		4.5	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
38		5.0	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
39		5.0	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
40		5.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5
41		5.5	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
42										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
43										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
44										5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0
45										5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
46										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
47										5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
48										5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5
49													5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
50													5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
51													6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0
52														6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
53														6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
54															6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5
55															6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
56															7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0
57																	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
58																	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5
59																	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	7.0
60																	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0
61																	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
62																	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time.

Note: The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table P.2-9

PWR Fuel Qualification Table for Zone 4 Fuel with 1.3 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel without CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																															
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		4.5	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
34		4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
36		5.0	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
38		5.5	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
39		6.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
40		6.0	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
41		6.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
42										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
43										5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5
44										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
45										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
46										6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0
47										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
48										6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
49													6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5
50													7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
51													7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0
52														7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
53														7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5
54															8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
55															8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0
56															8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5
57																	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
58																	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0
59																	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5
60																	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0
61																	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5
62																	11.5	11.5	11.5	11.5	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0

Not Analyzed

Note: *If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time for cooling times less than 10 years.*

Note: The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table P.2-10

PWR Fuel Qualification Table for Zone 1 Fuel with 1.7 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel with CCs)

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

BU GWd/ MTU	*Assembly Average Initial Enrichment (wt. % U-235)*																														
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34		3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
38		4.5	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0
39		4.5	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
40		4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
41		5.0	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
42										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
43										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5
44										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
45										4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
46										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
47										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
48										5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0
49													5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
50													5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
51													5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
52														5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5
53														5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
54															5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
55															5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0
56															6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
57																	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
58																	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5
59																	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
60																	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
61																	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0
62																	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time.

Note: *The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.*

Table P.2-11

PWR Fuel Qualification Table for Zone 2 Fuel with 2.0 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel with CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																																
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36		3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
38		3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
39		3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
40		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
41		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
42										3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
43										3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0
44										3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
45										4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
46										4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
47										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
48										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
49													4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5
50													4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
51													4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
52														4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
53														4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0
54															4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0
55															5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
56															5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5
57																	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
58																	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
59																	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
60																	6.0	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
61																	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
62																	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time.

Note: *The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.*

All changes on this page are AMD 11

Table P.2-12

PWR Fuel Qualification Table for Zone 3 Fuel with 1.5 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel with CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																															
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34		4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36		4.5	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
38		5.0	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
39		5.0	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5
40		5.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
41		5.5	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
42										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
43										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
44										5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0
45										5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
46										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
47										5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5
48										5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
49													5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
50						Not Analyzed							5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
51													6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0
52														6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
53														6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5
54															6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5
55															6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
56															7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0
57																	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
58																	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5
59																	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
60																	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0
61																	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
62																	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time.

Note: The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table P.2-13

PWR Fuel Qualification Table for Zone 4 Fuel with 1.3 kW per Assembly for the NUHOMS®-24PTH DSC (Fuel with CCs)

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

BU GWd/ MTU	*Assembly Average Initial Enrichment (wt. % U-235)*																															
	0.7	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25		3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28		3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30		4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32		4.5	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
34		4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
36		5.0	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
38		5.5	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
39		6.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0
40		6.0	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
41		6.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
42										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
43										5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5
44										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
45										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
46										6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0
47										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
48										6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
49													6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
50													7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
51													7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0
52														7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
53														7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5
54															8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0
55															8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0
56															8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5
57																	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0
58																	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0
59																	10.0	10.0	9.5	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5
60																	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0
61																	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5
62																	12.0	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time for cooling times less than 10 years.

Note: *The page that follows Table P.2-13 provides the explanatory notes and limitations regarding the use of this table.*

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Notes: Tables P.2-6 through P.2-13:

- *BU = Assembly Average burnup.*
- *Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with an assembly average initial enrichment less than 0.7 wt. % U-235 (or less than the minimum provided above for each burnup) and greater than 5.0 wt. % U-235 is unacceptable for storage.*
- *Fuel with a burnup greater than 62 GWd/MTU is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 3-years cooling.*
- *WE 15x15 PLSAs shall be limited to a minimum assembly average initial enrichment of 1.2 wt. % U-235.*
- *See Figures P.2-1 through P.2-5 for the description of zones.*
- *For reconstituted fuel assemblies with UO_2 rods and/or Zr rods or Zr pellets and/or stainless steel rods, use the assembly average equivalent enrichment to determine the minimum cooling time.*
- *The cooling times for damaged and intact assemblies are identical.*
- *Example: An INTACT FUEL ASSEMBLY without CCs, with a decay heat load of 1.7 kW or less, an initial enrichment of 3.65 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a 4.0 year cooling time as defined by 3.6 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) in Table P.2-6.*

All changes on this page are AMD 11

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Table P.2-19
Design Pressures for Tornado Wind Loading

HSM-H Wall Orientation(1)	Velocity Pressure (psf)	Pressure Coefficient(2)	Max/Min Design Pressure (psf)
Front	344	+0.68	234
Left	344	-0.60	-207
Rear	344	-0.43	-148
Right	344	-0.60	-207
Roof	344	-0.60	-207

Notes:
(1) Wind direction assumed to be from front. Wind loads from other directions may be found by rotating table values to desired wind direction.
(2) Pressure coefficient (used) = Gust factor (0.85)* Max/Min pressure coefficient.

Table P.2-20
B10 Specification for the NUHOMS®-24PTH Poison Plates

NUHOMS®-24PTH DSC Basket Type(1)	*Minimum B10 Areal Density, (grams/cm²)*	
	Borated Aluminum or MMC	*Boral®*
1A or 2A	*0.007*	*0.009*
1B or 2B	*0.015*	*0.019*
1C or 2C	*0.032*	*0.040*

(1) Basket Type 1 contains aluminum inserts in the R45 transition rails of the basket, Type 2 does not contain aluminum inserts.

Table P.2-21
Maximum Allowable Heat Load for the NUHOMS®-24PTH DSC

24PTH DSC Type	Basket Type (2)(3)	Transfer Cask	Max. Heat Load (kW) per DSC
24PTH-S or 24PTH-L(1)	1A, 1B, or 1C	OS197FC	40.8
		OS197/OS197H	31.2
24PTH-S or 24PTH-L(1)	2A, 2B, or 2C	OS197FC	31.2
24PTH-S-LC(1)	2A, 2B, or 2C	Standardized TC (solid neutron shield)	24.0

Notes:
(1) Allows storage of control components.
(2) Basket Type 1 (1A, 1B, 1C) has heat conductive aluminum inserts in the R45 basket transition rails.
(3) Basket Type 2 (2A, 2B, 2C) does not have heat conductive aluminum inserts in the R45 basket transition rails.

Table P.3.1-1
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Confinement Boundary
(Part 1 of 2)

All changes on this page are AMD 11

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	All	Not compliant with NCA. Quality Assurance *is provided according to* 10 CFR 72 Subpart G, in lieu of NCA-4000.
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section P.2 may be used for construction, but in no case earlier than 3 years before that specified in Section P.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section P.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	Requirements for Code Stamping of Components, *Code reports and certificates, etc.*	Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-1132	*Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.*	*Bottom shield plug assembly, outer bottom cover plate, lifting posts, grapple ring, grapple ring support are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.*
NB-2130	Material must be supplied by ASME approved material suppliers.	Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability *and* certification are maintained in accordance with TN's NRC approved QA program.
NB-4121	Material Certification by Certificate Holder	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT	The joints between the top outer and inner cover plates (or top forging assembly for the 24PTH-S-LC) and containment shell are designed and fabricated per ASME Code Case N-595-2, which provides alternative requirements for the design and examination of spent fuel canister closures. This includes the inner top cover plate weld around the vent & siphon block and the vent and siphon block welds to the shell. The closure welds are partial penetration welds and the root and final layer are subject to PT examination (in lieu of volumetric examination) in accordance with the provisions of ASME Code Case N-595-2. The 24PTH closure system employs austenitic stainless steel shell, lid materials, and welds. Because austenitic stainless steels are not subject to brittle failure at the operating temperatures of the DSC, crack propagation is not a concern. Thus, multi-level PT examination provides reasonable assurance that flaws of interest will be identified. The PT examination is done by qualified personnel, in accordance with Section V and the acceptance standards of Section III, Subsection NB-5000. This alternative does not apply to other shell confinement welds, i.e., the longitudinal and circumferential welds of the DSC shell, and the inner bottom cover plate-to-shell weld (or bottom forging to shell weld, as applicable) which comply with NB-4243 and NB-5230.

Table P.3.1-1
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Confinement Boundary
(Part 2 of 2)

All changes on this page are AMD 11

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NB-6100 and 6200	All pressure retaining components and completed systems shall be pressure tested. The preferred method shall be hydrostatic test.	The NUHOMS®-24PTH DSC is pressure tested in accordance with ASME Code Case N-595-2. The shield plug support ring and the vent and siphon block are not pressure tested due to the manufacturing sequence. The support ring is not a pressure-retaining item and the vent and siphon block weld is helium leak tested after fuel is loaded to the same criteria as the inner top closure plate-to-shell weld (ANSI N14.5-1997 leaktight criteria).
NB-7000	Overpressure Protection	No overpressure protection is provided for the NUHOMS® *DSCs*. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.
NB-5520	*NDE Personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of the Recommended Practice SNT-TC-1A to include up to the most recent 2011 edition.*

Table P.3.1-2
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Basket Assembly
(Part 1 of 2)

All changes on this page are AMD 11

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section P.2 may be used for construction, but in no case earlier than 3 years before that specified in Section P.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section P.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG-1100	Requirements for Code Stamping of Components, *Code reports and certificates, etc.*	Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp or *to* be ASME Certified.
NG-2000	Use of ASME Material	*Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness.* Material properties in the ASME Code for Type 6061 aluminum are limited to 400°F to preclude the potential for annealing out the hardening properties. Annealed properties (as published by the Aluminum Association and the American Society of Metals) are conservatively assumed for the aluminum transition rails for use above the Code temperature limits.
NG-2130	Material must be supplied by ASME approved material suppliers.	Material is certified to meet all ASME Code criteria, but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NG-2130 is not possible. Material traceability & certification are maintained in accordance with TN's NRC approved QA program.
NG-4121	Material Certification by Certificate Holder	
NG-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA Data packages are prepared in accordance with the requirements of TN's approved QA program.
NG-3000/ Section II, Part D, Table 2A	Maximum temperature limit for Type 304 plate material is 800°F	Not compliant with ASME Section II Part D Table 2A material temperature limit for Type 304 steel for the postulated transfer accident case (117°F, loss of sunshade, loss of neutron shield). This is a post-drop accident scenario, where the calculated maximum steady state temperature is 862°F, the expected reduction in material strength is small (less than 1 ksi by extrapolation), and the only primary stresses in the basket grid are deadweight stresses. The recovery actions following the postulated drop accident are as described in Section 8.2.5 of the FSAR.

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Table P.3.1-2
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Basket Assembly
(Part 2 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NG-3352	Table NG 3352-1 lists the permissible welded joints	The fusion (spot) type welds between the stainless steel insert plates (straps) and the stainless steel fuel compartment tube are not permissible welds per Table NG-3352-1. These welds are qualified by testing. The required minimum tested capacity of the welded connection (at each side of the tube) shall be 36 Kips (at room temperature). This value is based on a margin of safety (test-to-design) of 1.6, which is larger than the Code-implied margin of safety for Level D loads. The minimum capacity shall be determined by shear tests of individual specimens made from production material. The tests shall be corrected for temperature differences (test-to-design) and for material properties (actual-to-ASME Code minimum values) to demonstrate that the capacity of the welded connection with ASME minimum properties, tested at design temperatures, will meet the 36 Kips test requirement. The capacity of the welded connection is determined from the test of the weld pattern of a typical insert plate to the tube connection. *The welds will be visually inspected to confirm that they are located over the insert plates, in lieu of the visual acceptance criteria of NG-5260 which are not appropriate for this type of weld. A joint efficiency (quality) factor of 1.0 is utilized for the fuel compartment longitudinal seam welds. Table NG-3352-1 permits a joint efficiency (quality) factor of 0.5 to be used for full penetration weld examined by ASME Section V visual examination (VT). For the 24PTH DSC, the compartment seam weld is thin and the weld will be made in one pass. Both surfaces of weld (inside and outside) will be fully examined b VT and therefore a factor of 2 x 0.5=1.0, will be used in the analysis. This is justified as both surfaces of the single weld pass/layer will be fully examined, and the stainless steel material that comprises the fuel compartment tubes is very ductile.*
NG-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A.*	*Permit use of the Recommended Practice SNT-TC-1A to include up to the most recent 2011 edition.*

Under these assumptions, the hydrogen concentration in the space between the water and the shield plug is a function of the time water is in the DSC prior to backfilling with helium. The hydrogen concentration is (0.25 ft^3 H_2/hr)*(8 hr) / (84.6 ft^3) = 2.36%. Monitoring of the hydrogen concentration before and during welding operations is performed to ensure that the hydrogen concentration does not exceed 2.4%, which is well below the ignitable limit of 4%. If the hydrogen concentration exceeds 2.4%, welding operations are suspended and the DSC is purged with an inert gas. In an inert atmosphere, hydrogen will not be generated.

Effect of Galvanic Reactions on the Performance of the System

There are no significant reactions that could reduce the overall integrity of the DSC or its contents during storage. The DSC and fuel cladding thermal properties are provided in Section P.4. The surface emissivity of the fuel compartment tube is 0.46, which is typical for non-polished stainless steel surfaces. If the stainless steel is oxidized, this value would increase, improving heat transfer. The fuel rod emissivity value used is 0.80, which is a typical value for oxidized Zircaloy. Therefore, the passivation reactions would not reduce the thermal properties of the component cask materials or the fuel cladding.

There are no reactions that would cause binding of the mechanical surfaces or the fuel to basket compartment boxes due to galvanic or chemical reactions.

There is no significant degradation of any safety components caused directly by the effects of the reactions or by the effects of the reactions combined with the effects of long term exposure of the materials to neutron or gamma radiation, high temperatures, or other possible conditions.

P.3.4.2 Positive Closure

Positive closure is provided by the OS197, OS197H, OS197FC and Standardized TCs. No change *to Section 3.3.2.*

P.3.4.3 Lifting Devices

As described in Section 8.1.1.9 (B), the evaluations for the OS197 and OS197H TC trunnions are based on critical lift weights (with water in the DSC) of 208,500 lbs and 250,000 lbs, respectively. These lifted weights capacities are not changed for the OS197FC since the only design feature that is different between the OS197/OS197H and the OS197FC is the top lid and the optional addition of wedge-shaped plates *at* the TC bottom. The maximum critical lift weight with a NUHOMS®-24PTH DSC is approximately 215,000 lbs. Therefore, an OS197FC TC that is based on the OS197H design is acceptable with any NUHOMS®-24PTH DSC. An OS197FC TC that is based on the OS197 design is limited to a total critical lift weight of 208,500 lbs. Water may be drained from the DSC cavity, if needed, to meet this weight limit.

For the Standardized Cask, the critical lift weight is limited to 200,000 lbs. Water may be drained from the DSC cavity, if needed, to meet this weight limit.

P.3.5 Fuel Rods

No change to *Section 3.1.1*.

intensity for each component of the DSC calculated for the enveloping normal operating, off-normal, and accident load combinations. For comparison, the appropriate ASME Code allowables are also presented in these tables.

P.3.7.11.2 DSC Fatigue Evaluation

Although the normal and off-normal internal pressures for the NUHOMS®-24PTH DSC are higher relative to the NUHOMS®-24P DSC, the range of pressure fluctuations due to seasonal temperature changes are essentially the same as those evaluated for the NUHOMS®-24P DSC. Similarly, the normal and off-normal temperature fluctuations for the NUHOMS®-24PTH DSC due to seasonal fluctuations are essentially the same as those calculated for the NUHOMS®-24P DSC. Therefore, the fatigue evaluation presented in Section 8.2.10.2 for the 24P DSC remains applicable to the NUHOMS®-24PTH DSC.

P.3.7.11.3 TC Load Combination Evaluation

There is no change to the TC load combination evaluations. Table P.3.7-15 is a summary of OS197/OS197H/OS197FC TC stresses. This table incorporates the thermal stress analysis results for these TCs loaded with a 24PTH DSC documented in P.3.6.1.5. In addition, the results of the modified top cask lid, documented in P.3.6.1.5, have been incorporated (primary stresses in Table P.3.7-15). The stress results summarized in Table P.3.7-15 use the stresses due to mechanical loads for the OS197H as summarized in Chapter 8. Allowable stresses have been adjusted for the higher temperatures associated with the TC loaded with a 24PTH DSC.

The evaluations performed in Sections 8.1 and 8.2 for the Standardized, OS197, and OS197H casks are based on payloads of 93,300 lbs, 97,250 lbs, and 116,000 lbs, respectively. The maximum total cask payload with a dry-loaded NUHOMS®-24PTH DSC is approximately 94,000 lbs. Therefore, a OS197FC TC that is based on either the OS197 or the OS197H TC is acceptable with any NUHOMS®-24PTH DSC as long as the total TC dry payloads are within their respective analyzed weights listed above. Water may be drained from the DSC cavity, if needed, to meet the above weight limits.

P.3.7.11.4 TC Fatigue Evaluation

No change *to Section 8.2.10.3.*

P.3.7.11.5 HSM –H Load Combination Evaluations

P.3.7.11.5.1 HSM –H Concrete Component Evaluation

The required strength, U, for critical sections of concrete is calculated in accordance with the requirements of ANSI 57.9 [3.14] and Chapter 9 of ACI 349 [3.27], including the strength reduction factors defined in ACI 349, Section 9.3.

The concrete design loads are multiplied by load factors and combined to simulate the most adverse load conditions. The load combinations described in Table P.3.7-16 are used to evaluate the concrete components.

load. The computed maximum ductility ratio for the door is less than 2 (compared to the allowable ductility of 20).

For the door anchorage, the controlling load is tornado generated differential pressure drop load. The maximum tensile force per bolt (there are four bolts that attach the door assembly to the front concrete wall of the HSM-H) is 8.56 kips. This is less than the allowable load per bolt of 44.3 kips. The concrete pull-out strength is conservatively estimated as 24 kips. Half of the concrete pull-out strength (12 kips) is greater than the tension load of 8.56 kips per bolt, thus satisfying the ductility requirements of the ACI Code.

P.3.7.11.6.6 Evaluation of the HSM-H Heat Shields

The top heat shield (louvers) consists of six panels. Each panel has two aluminum mounting bars. The aluminum louvers are mounted on the mounting bars. Each mounting bar is suspended from the roof by two threaded rods. The natural lateral frequency of a typical rod is conservatively estimated to be 9.0 Hz. The combined axial and bending stress in the hanger rods is 24.0 ksi. The allowable axial and bending stress is 84.3 ksi.

The side heat shields consists of three panels. Each panel is suspended from the roof by two threaded rods, and supported laterally and longitudinally by four rods. The maximum axial plus bending stress in the lateral and longitudinal support rods is 83.7 ksi. The allowable axial and bending stress is 84.3 ksi. The maximum temperature used in the stress analysis of the heat shields bounds the maximum temperature reported in Chapter P.4.

The alternate top heat shield consists of two panels made of stainless steel plate. The panels are suspended from the roof by fifteen ½" diameter rods threaded into concrete embedments. The combined axial and bending stress in the rods is 59.5 ksi. The allowable stress is 70.2 ksi.

The alternate side heat shield configuration may consist of four panels made from aluminum or stainless steel. The panels are supported off the base unit side wall by thirty four rod stand-offs threaded into concrete embedments. For the aluminum heat shield configuration, the maximum axial and bending stress in the rods is about 1 ksi and 53.7 ksi, respectively. For the stainless steel heat shield configuration, the maximum axial and bending stress in the rods is about 1.4 ksi and 79.3 ksi, respectively. The axial and bending stress allowable for the rods is 67.9 ksi and 112.3 ksi, respectively.

P.3.7.11.6.7 Evaluation of the HSM-H Seismic Retainers

The seismic retainer consists of a capped tube steel embedment located within the bottom center of the round access opening of the HSM-H, and a tube steel retainer assembly that drops into the embedment cavity after 24PTH-DSC transfer is complete. The drop-in retainer extends approximately 4" above the rail to provide axial restraint of the 24PTH-DSC. The maximum seismically induced shear load in the retainer is 61 kips. The maximum shear stress in the retainer is 15.25 ksi. The allowable shear stress is 17.8 ksi.

P.3.7.11.6.8 Thermal Cycling of the HSM-H

No change *to Section 8.2.10.5.*

P.4 Thermal Evaluation

P.4.1 Discussion

All changes on this page are AMD 11

This chapter presents the thermal evaluations which demonstrate that the NUHOMS®-24PTH system meets the thermal requirements of 10CFR72 for the dry storage of spent fuel. The NUHOMS®-24PTH system is designed to passively reject decay heat during storage and transfer for normal, off-normal and accident conditions while maintaining temperatures and pressures within specified regulatory limits.

Several thermal design criteria are established for the thermal analysis of the 24PTH DSC basket as discussed below.

Maximum temperatures of the confinement structural components must not adversely affect the confinement function,

Maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations including vacuum drying and helium backfilling of the 24PTH DSC per Interim Staff Guidance (ISG) No. 11, Revision *3* [4.19]. In addition, ISG-11 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65°C (117°F) during drying and backfilling operations,

Maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal thermal transients [4.19],

The maximum DSC cavity internal pressures during normal, off-normal and accident conditions must be below the design pressures of 15 psig, 20 psig and 120 psig, respectively, and

A total of five (5) Heat Load Zoning Configurations (HLZCs) are allowed for the 24PTH DSCs as shown in Figures P.2-1 through P.2-5. The maximum total heat load per DSC is 40.8 kW, 31.2 kW or 24 kW depending upon the specific DSC types and HLZCs. The thermal analysis is carried out for HLZC 1 shown in Figure P.4-36 with 40.8 kW heat load because it bounds the HLZCs 2 and 3. In HLZC 1, the fuel assemblies at the center 4 locations in the DSC basket have a maximum heat load of 1.7 kW per assembly as compared to zero (empty locations) for HLZC 2 and 1.5 kW for HLZC 3. Therefore, HLZC 1 results in higher fuel cladding and basket temperatures compared to HLZCs 2 or 3. To bound all possible combinations of heat loads from fuel assemblies allowed in HLZC 5 of Figure P.2-5, the thermal analysis is carried out for the HLZC shown in Figure P.4-38. In this thermal analysis configuration, the highest possible allowed decay heat assemblies are assumed to be at the center and upper half of the basket locations resulting in bounding fuel cladding and basket temperatures.

The thermal analysis is carried out for the three NUHOMS®-24PTH DSC configurations (24PTH-S, 24PTH-L, and 24PTH-S-LC DSC types in combination with six basket types of the NUHOMS®-24PTH system described in Section P.2.1). A summary of the three system configurations analyzed in this chapter are summarized below:

11. **Air [4.4]**

Temperature (°F)	K (Btu/min-in-°F)(1)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
71	2.075E-5	4.323e-5	0.240
107	2.199E-5	4.051e-5	0.241
206	2.528E-5	3.443e-5	0.242
314	2.869E-5	2.963e-5	0.243
404	3.139E-5	2.656e-5	0.245
512	3.447E-5	2.361e-5	0.248
602	3.693E-5	2.159e-5	0.251
692	3.929E-5	1.991e-5	0.253
764	4.114E-5	1.875e-5	0.256
800	4.203E-5	1.823e-5	0.257

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12. **Helium [4.4]**

Temperature (°F)	K (Btu/min-in-°F)(1)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
200	1.361E-4	4.81E-6	1.240
300	1.493E-4	4.18E-6	
400	1.635E-4	3.69E-6	
500	1.793E-4	3.31E-6	
600	1.949E-4	2.99E-6	
700	2.094E-4	2.74E-6	
800	2.232E-4	2.52E-6	
900	2.364e-4	2.33E-6	

(1) Thermal properties as listed are for atmospheric pressure, however, thermal conductivities for air and helium are assumed to be constant between atmospheric pressure and 3 Torr, absolute [4.18]. Air is not allowed for DSC blowdown operations.

13. **24PTH DSC basket effective thermal properties (See Section P.4.8.3)**

24PTH DSC Basket Effective Transverse Thermal Conductivity

24PTH S/L DSC with inserts,		24PTH-S and 24PTH-L DSC without inserts		24PTH-S-LC DSC without inserts	
Temperature (°F)	K (Btu/min-in-°F)	Temperature (°F)	K (Btu/min-in-°F)	Temperature (°F)	K (Btu/min-in-°F)
248	3.676E-03	251	2.759E-03	249	2.103E-03
338	3.949E-03	341	2.952E-03	340	2.249E-03
427	4.275E-03	430	3.195E-03	429	2.432E-03
518	4.615E-03	521	3.442E-03	520	2.619E-03
610	4.958E-03	613	3.697E-03	612	2.811E-03
703	5.292E-03	706	3.945E-03	705	3.000E-03
799	5.522E-03	801	4.119E-03	800	3.134E-03
897	5.618E-03	899	4.217E-03	898	3.208E-03

The total equivalent loss coefficients calculated for the steady state cases for the HSM-H regions are:

Decay Heat Load (kW)	Ambient Temperature (°F)	Total equivalent loss coefficient K (ft^{-4})$^{(1)}$	Total equivalent loss coefficient K (ft^{-4})$^{(2)}$
40.8	-40	0.0864	0.0980
	0	0.0872	0.0990
	100	0.0889	0.1011
	117	0.0890	0.1012
31.2	-40	0.0865	0.0984
	117	0.0892	0.1017
24	-40	0.0868	0.0988
	117	0.0896	0.1021

(1) Based on louvered top heat shields, finned side heat shields and with slots on plate on top of support rail.

(2) Based on flat top heat shields, flat side heat shields and without slots on plate on top of support rail.

Using these loss coefficients in equation for $\Delta T_{HSM\text{-}H}$, the exit and stack air temperature for the normal and off-normal cases are calculated.

The DSC outer surface is divided into three regions along the DSC circumference as shown in Figure P.4-2. The bulk air temperatures at each of these specified regions on the DSC are shown in Table P.4-1 for the range of ambient conditions. These bulk air temperatures are used in the subsequent HSM-H analyses to calculate the temperatures throughout HSM-H and 24PTH DSC shell.

P.4.4.3.1 *Effect of Dose Reduction Hardware on Airflow Analysis for HSM-H/HSM-HS*

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Proprietary Information Withheld Pursuant to 10 CFR 2.390

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Proprietary Information Withheld Pursuant to 10 CFR 2.390

Proprietary Information Withheld Pursuant to 10 CFR 2.390

P.4.4.4 Description of the Thermal Model of HSM-H with 24PTH DSC

A half symmetric, three dimensional, ANSYS [4.31] finite element model of the HSM-H loaded with 24PTH DSC is shown in Figure P.4-3.

The thermal model consists of SOLID70 conduction elements that simulate concrete and steel support structures of the HSM-H, and SHELL57 elements superimposed on SOLID70 elements, as required, to model radiating surfaces using MATRIX50 super elements. As such, radiation between the DSC shell, heat shields, and HSM-H walls is modeled using the ANSYS /AUX12 methodology. For elements wherein radiation is not applicable, the SHELL57 elements are unselected prior to solving the model. Additionally, to reduce the number of the nodes associated with the model's super-elements, the web of the supporting beam is modeled using only SHELL57 elements. As such, conservatively, radiation is not applied on the web of the supporting beam. This methodology is valid since the supporting beam's web greatly shields the support steel from the DSC radiation via its own flanges. The properties and dimensions of the support beam, such as the thickness of the web are given as real constants to the appropriate SHELL57 elements.

The base plate of the side heat shields are modeled as flat plates. Convection from the fins attached to the side shields is modeled using an equivalent convection coefficient. The purpose of adding fins on the side heat shield is to provide more surface area for the convection. Flat stainless steel heat shields are also evaluated.

P.4.6.6.4 Boundary Conditions, Off-Normal Transfer

In accordance with Section P.4.5, an analysis of a 24PTH DSC in the OS197FC cask and in the Standardized TC is performed for the following ambient conditions:

Maximum normal ambient temperature of 117°F with solar shield in place in OS197FC cask,
Maximum normal ambient temperature of 117°F with solar shield in place in Standardized cask.

These analyses determine maximum DSC shell surface temperatures. The maximum calculated DSC shell temperatures are applied to the exterior surface of the DSC shell in the DSC/basket/payload finite element model.

For Standardized TC, the DSC shell temperatures were calculated for 24 kW in Section 8.1.3. The results are summarized in Table P.4-39.

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P.4.6.6.5 24PTH DSC Thermal Model Results for Off-Normal Conditions of Storage and Transfer

According to the NUHOMS® CoC 1004, Technical Specification *5.3.1.B*, "TC/DSC Transfer Operations at High Ambient Temperatures" for transfer operations, when ambient temperatures exceed 100°F, a solar shield shall be used to provide protection against direct solar radiation.

Fuel Cladding Temperatures

The maximum fuel cladding temperatures during off-normal conditions of storage and transfer are evaluated for all HLZCs and compared with the corresponding fuel cladding temperature limits.

The results are reported in Table P.4-20 for HLZCs 1 (bounds HLZCs 2 and 3), 4 and 5.

DSC Basket Materials Component Temperatures

The maximum temperatures of the basket assembly for off-normal conditions of storage and transfer for HLZC 1 (bounds HLZCs 2 and 3), 4 and 5 are listed in Table P.4-21, Table P.4-22, and Table P.4-23, respectively.

P.4.6.6.6 Off-Normal 24PTH DSC Maximum Internal Pressure during Storage/Transfer

The maximum average helium backfill gas temperature for off-normal conditions of storage and transfer occurs when the 24PTH-L DSC with aluminum inserts, and a heat load of 31.2 kW (HLZC 4) is in OS197FC TC with an ambient temperature of 117°F and sunshade. The average helium temperature is 513°F (973°R). Per NUREG 1536 [4.7], the percentage of fuel rods ruptured for off-normal cases is 10%.

A summary of the maximum off-normal operating pressures for the various 24PTH DSC configurations are presented in Table P.4-24.

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P.4.7 Thermal Evaluation for Loading/Unloading Conditions

All fuel transfer operations occur when the NUHOMS®-24PTH DSC and TC are in the spent fuel pool. The fuel is always submerged in free-flowing pool water permitting heat dissipation. After fuel loading is complete, the TC cask and DSC are removed from the pool and the DSC is drained, dried, sealed and backfilled with helium.

The unloading operation considered is the reflood of the 24PTH DSC with water.

P.4.7.1 Maximum Fuel Cladding Temperatures During Vacuum Drying

The loading condition evaluated for the NUHOMS®-24PTH DSC is the heatup of the DSC before its cavity is backfilled with helium. This typically occurs during the performance of the vacuum drying operation of the DSC cavity with the TC in the vertical configuration inside the fuel handling building, and the annulus between the TC and the DSC is full of water.

Analyses were performed for the vacuum drying condition in order to ensure that the fuel cladding and 24PTH DSC structural component temperatures remain below the maximum allowable material limits.

During vacuum drying operation, water in the DSC cavity is forced out of the cavity (blowdown operation) *using helium* before the start of vacuum drying.

The vacuum drying of the DSC is assumed not to reduce the pressure sufficiently to reduce the thermal conductivity of the water vapor or helium in the DSC cavity [4.17] and [4.35]. Radiation in the gaps within the basket and rail components is conservatively neglected.

Thermal analysis is performed using the three-dimensional model developed in Section P.4.6, with decay heat loads for HLZCs 1, 4, and 5 and an initial DSC shell surface temperature of 215°F. The initial temperature of the DSC, basket and fuel is assumed to be 215°F, based on the saturation boiling temperature of the fill water. Table P.4-30 provides the maximum calculated temperatures for the fuel cladding and Table P.4-31 through Table P.4-33 provide the maximum calculated basket component temperatures for all three configurations. The maximum cladding temperature of *578°F for HLZC 4* during vacuum drying *is well* below the limit of 752°F [4.19].

P.4.7.2 Evaluation of Thermal Cycling of Fuel Cladding During Vacuum Drying, Helium Backfilling and Transfer Operations

ISG-11 [4.19] also states that thermal cycling is to be minimized and imposes a limit of 65°C (117°F) on thermal cycling (reduction in fuel clad temperature from previous peak temperature). The basis for the limit is that as the cladding temperature is reduced more than 65°C the concentration of hydrogen available for hydride reorientation becomes significant.

The thermal analysis of the 24PTH DSC during blowdown operation assumes helium is used to drain the water from the 24PTH DSC cavity and subsequent vacuum drying occurs with a helium environment. This option eliminates a fuel cladding temperature drop that would take place during helium backfilling of the 24PTH DSC subsequent to vacuum drying in a nitrogen environment and it eliminates the need for a time limit on the vacuum drying operation, since the thermal conductivity of helium does not change with pressure during vacuum drying operations.

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As shown in Table P.4-30 the maximum fuel cladding temperature limit of $T_{ISG\ limit}$ = 400°C (752°F) and the 65°C (117°F) ΔT_{limit} listed in ISG-11 [4.19] are satisfied for the 24PTH DSC.

P.4.7.3 Reflooding Evaluation

For unloading operations, the DSC is filled with the spent fuel pool water through its siphon port. During this filling operation, the DSC vent port is maintained open with effluents routed to the plant's off-gas monitoring system. The NUHOMS®-24PTH DSC operating procedures recommend that the DSC cavity atmosphere be sampled prior to introducing any reflood water in the DSC cavity.

Initially, the pool water is added to the DSC cavity containing hot fuel and basket components, some of the water will flash to steam causing internal cavity pressure to rise. This steam pressure is released through the vent port. The procedures specify that the flow rate of the reflood water be controlled such that the internal pressure in the DSC cavity does not exceed 20 psig. This is assured by monitoring the maximum internal pressure in the DSC cavity during the reflood event. The reflood for the DSC is considered as a Service Level D event and the design pressure of the DSC is 120 psig for 24PTH-S or –L DSCs and 90 psig for 24PTH-S-LC DSC. Therefore, there is sufficient margin in the DSC internal pressure during the reflooding event to assure that the DSC will not be over pressurized.

The maximum fuel cladding temperature during reflooding event is significantly less than the vacuum drying condition owing to the presence of water/steam in the DSC cavity. The analysis results presented in Table P.4-30 show that the maximum cladding temperature during vacuum drying is *578*°F. Hence, the peak cladding temperature during the reflooding operation will be less than *578*°F.

To evaluate the effects of the thermal loads on the fuel cladding during reflooding operations, a conservative assumption of high maximum fuel rod temperature of 750°F and a low quench water temperature of 50°F are used.

The material properties, corresponding to a temperature of 750°F, are used in the evaluation:

Modulus of Elasticity, E (psi) = 11.1 x10^6 [from Figure 4 of 4.10]
Coefficient of thermal expansion, α, (in/in/°F) = 3.73 x 10-6 [4.11]
Poison's Ratio, ν, = 0.38 [4.12]
Yield Stress (irradiated), Sy, = 50,500 psi [4.13 [4.10]

The fuel cladding is evaluated as a hollow cylinder with an outer surface temperature of T (50°F), and the inner surface temperature of T+ΔT (750°F) using [4.13] equations. The maximum thermal stress in the fuel cladding due to the temperature gradient during reflooding is calculated as follows:

The maximum circumferential stress at the outer surface is given by:

$$\sigma_t = \frac{\Delta T * \alpha \cdot E}{2(1-\nu)\log_e(c/b)} * (1 - \frac{2 * b^2}{(c^2 - b^2)} * \log_e c/b)$$

The maximum circumferential stress at the inner surface is given by:

4.19 Interim Staff Guidance No. 11, Revision *3*, "Cladding Considerations for the Transportation and Storage of Spent Fuel," *November 17, 2003.*

4.20 TN Response to RAI No. 2 and Submittal of Revision 4 of Application for Amendment No. 5 to the NUHOMS® Certificate of Compliance No. 1004 (TAC No. L23343), January 24, 2003 (Docket 72-1004).

4.21 J. M. Creer, et al, "The TN-24 PWR Spent Fuel Storage Cask: Testing and Analyses," PNL Report, Report No. PNL-6054, 1987.

4.22 SAND90–2806, Sanders, T. L., et al., "A Method for Determining the Spent Fuel Contribution to Transport Cask Containment Requirements," TTC-1019, UC-820, November 1992.

4.23 Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE, A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation for Workstations and Personal Computers," NUREG/CR-0200, Rev. 6, ORNL/NUREG/CSD-2/V3/R6.

4.24 Report, Azzazy, M., Emissivity Measurements of 304 Stainless Steel, Prepared for Southern California Edison, September 6, 2000, Transnuclear File No. NUH32PT.0100-01

4.25 Duke Energy's letter, Transnuclear, Inc., File No. NUH24PTH-0100-01.

4.26 Frank Kreith, Principles of Heat Transfer, Third Edition, Harper and Row Publishers.

4.27 Incropera and DeWitt, Handbook of Heat And Mass Transfer Fundamentals, 5th edition, Wiley Publishers, 2002, Table A.6 pp 924.

4.28 Bentz, "A Computer Model to Predict the Surface Temperature and Time-of-wetness of Concrete Pavements and Bridge Decks", Report # NISTIR 6551, National Institute of Standards and Technology, 2000.

4.29 Zoldners, "Thermal Properties of Concrete under Sustained Elevated Temperatures", ACI Publications, Paper SP 25-1, American Concrete Institute, Detroit, MI, 1970, Cavanaugh, Guide to Thermal Properties of Concrete and Masonry Systems, Reported by ACI Committee 122, Report #ACI 122R-02, American Concrete Institute, Detroit, MI, 2002.

4.30 Siegel, Howell, "Thermal Radiation Heat Transfer," 4th Edition, 2002.

4.31 ANSYS Computer Code and User's Manuals, Rev. 6.0, See Test Report E-19197, Rev. 0 and E-20184, Rev. 0 for validation of computer code.

4.32 NRC, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material," 2003.

4.33 I.E. Idelchik, Handbook of Hydraulic Resistance, 3rd Edition, 1994.

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Table P.4-14
Fuel Cladding Normal Condition Maximum Temperatures

Operating Condition	HLZC 1 (4) (°F)	HLZC 4 (°F)	HLZC 5 (°F)	Limit (°F)
DSC in HSM-H, 0°F ambient	656	<656 (1)	<561 (2) / 561 (3)	752 (6)
DSC in HSM-H, 100°F ambient	718	<718 (1)	<634 (2) / 634 (3)	
DSC in HSM-H, 100°F ambient (8)	734	<704 (7)	<628 (7)	
DSC in TC, 0°F ambient	<711 @ 12.8 hrs	<732 @ 30 hrs	630	
DSC in TC, 100°F ambient	711 @ 11.5 hrs	732 @ 27.3 hrs	714	
DSC in TC, 100°F ambient (5)	N/A	733	N/A	

1) Temperature is bounded by temperature for HLZC 1.
2) Temperature for storage in HSM-H is bounded by temperature for storage in HSM model 102.
3) Temperature for storage in HSM model 102.
4) Temperatures for HLZC 1 bounds the temperatures for HLZC 2 and 3.
5) Temperatures with aluminum inserts in R45 transition rails.
6) ISG-11, Revision *3* [4.*19*]
7) Temperatures are bounded by DSC in HSM-H, 117 °F ambient cases (Table P.4-20).
8) HSM-H with flat stainless steel heat shields.

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Table P.4-25
Fuel Cladding Accident Condition Maximum Temperatures

Heat Load Zoning	HLZC 1(2) (°F)	HLZC 4 (°F)	HLZC 5 (°F)	Limit (°F)
DSC in HSM, Blocked Vent, 117°F	891(1)	<891(2)	<821(3) / 821(4)	1058(5)
DSC in TC, loss of sun shade, neutron shield water and air circulation with fan, if used, 117°F	914	843	747	

1) Temperature at 38.5 hour and 30.0 hour of blocked vents for louvered top heat shield/finned side heat shield and stainless steel heat shields (top and side) respectively.
2) Temperatures for HLZC 1 bound the temperatures for HLZC 2, 3, and 4.
3) Temperature for storage in HSM-H is bounded by temperature for storage in HSM Model 102.
4) Temperature for storage in HSM Model 102 at 40 hours of blocked vent.
5) ISG-11, Revision *3* [*4.19*]

Table P.4-30
Vacuum Drying Fuel Cladding Maximum Temperatures (°F)

Operating Condition	HLZC 1	HLZC 2	HLZC 3	HLZC 4	HLZC 5	Limit (°F)
Vacuum Drying in helium environment (or helium backfilling), steady-state	573	(2)	(2)	578	555	752(5)

(1) *Deleted.*
(2) Temperature is bounded by temperature for HLZC 1.
(3) *Deleted.*
(4) *Deleted.*
(5) ISG-11, Revision *3* [4.*19*]

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Table P.4-31
DSC Basket Assembly Maximum Component Temperatures during Vacuum Drying, HLZC 1 (40.8 kW)

Operating Condition	T_{alum} (°F)	T_{poison} (°F)	T_{tube} (°F)	$T_{DSC\ shell}$ (°F)
Vacuum Drying in helium environment, steady-state	480	481	483	215

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Table P.4-32
DSC Basket Assembly Maximum Component Temperatures during Vacuum Drying, HLZC 4 (31.2 kW)

Operating Condition	T_{alum} (°F)	T_{poison} (°F)	T_{tube} (°F)	$T_{DSC\ shell}$ (°F)
Vacuum Drying in helium environment, steady-state	508	509	510	215

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Table P.4-33
DSC Basket Assembly Maximum Component Temperatures during Vacuum Drying, HLZC 5 (24 kW)

Operating Condition	T_{alum} (°F)	T_{poison} (°F)	T_{tube} (°F)	$T_{DSC\ shell}$ (°F)
Vacuum Drying in helium environment, steady-state	482	483	484	215

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Figure P.4-42 Maximum Fuel Temperatures during Vacuum Drying Transient with Nitrogen for HLZCs 1, 4, and 5

P.5 Shielding Evaluation

The radiation shielding evaluation for the Standardized NUHOMS® System (during loading, transfer and storage) for the other NUHOMS® canisters is discussed in other sections and appendices of the FSAR. The following radiation shielding evaluation specifically addresses the shielding evaluation of the NUHOMS® 24PTH system with design-basis PWR fuel and control components (CCs) loaded in a NUHOMS®-24PTH DSC.

The shielding analysis is carried out for the three DSC configurations (24PTH-L, 24PTH-S, and 24PTH-S-LC) of the NUHOMS®-24PTH system described in Section P.1. The 24PTH-L and 24PTH-S DSCs are transferred either in the OS197/OS197H Transfer Cask (TC) or the OS197FC TC depending upon the heat load and stored in the HSM-H. The 24PTH-S-LC DSC is transferred in the Standardized TC and stored in either the HSM-H or HSM-Model 102. The seven possible loading combinations are listed below:

(1) 24PTH-L DSC → OS197FC TC (bounds OS197/OS197H TCs)
(2) 24PTH-L DSC → HSM-H
(3) 24PTH-S DSC → OS197FC TC (bounded by #1)
(4) 24PTH-S DSC → HSM-H (bounded by #2)
(5) 24PTH-S-LC DSC → Standardized TC
(6) 24PTH-S-LC DSC → HSM-H (bounded by #7)
(7) 24PTH-S-LC DSC → HSM-Model 102

The design of HSM-H is similar to HSM Model 102 except the HSM-H has improved shielding performance due to the following design features:

- Elimination of 6" uniform gap between adjacent modules,
- Innovative shielded inlet and outlet ventilation openings,
- Increased concrete thickness in roof, front and backwalls and shield walls, and
- Increased shielding in the HSM door.

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These design features results in the occupational and site dose rates ALARA.

The basket layout for the three DSC configurations is identical except for the length of the DSC components and the shield plug design. The 24PTH-S DSC and 24PTH-L DSC differ in DSC and cavity length, while the 24PTH-S-LC DSC and 24PTH-S DSC differ in cavity length due to a different shield plug design. The 24PTH-L/S has carbon steel shield plugs, while the 24PTH-S-LC has thinner lead shield plugs to increase cavity length to allow for greater fuel lengths in a shorter canister.

Each DSC configuration is designed to store up to 24 intact (and up to 12 damaged, with remaining intact) PWR fuel assemblies. The 24PTH-L and 24PTH-S-LC DSCs are also designed to store up to 24 intact standard PWR fuel assemblies with or without CC; such as burnable poison rod assemblies (BPRAs), Control Rod Assemblies (CRAs), Thimble Plug

P.8.1 Procedures for Loading the Cask

All changes on this page are AMD 11

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

P.8.1.1 Preparation of the TC and DSC

1. Prior to placement in dry storage, the candidate intact and damaged fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification *2.1*.
2. Prior to being placed in service, the TC is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification *5.2.4.d*.
3. Place the TC in the vertical position in the cask decon area using the cask handling crane and the TC lifting yoke.
4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.
5. Remove the TC top cover plate and examine the cask cavity for any physical damage and ready the cask for service. If required by the plant lifting crane capacity limit, drain the TC neutron shield water to an acceptable location.
6. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed.
7. Verify that the DSC basket type (1A, 2A etc.) is appropriate for the specific fuel loading campaign.
8. Using a crane, lower the DSC into the cask cavity by the internal lifting lugs and rotate the DSC to match the cask and DSC alignment marks.
9. Fill the cask-DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask-DSC annulus by pressurizing the seal with compressed air.
10. If damaged fuel assemblies are included in a specific loading campaign, place the required number of bottom end caps provided (up to a maximum of 12) into the cell locations per Technical Specification *2.1*. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.
11. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification *3.2.1*.

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

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

   - A cask/DSC loading plan is developed to verify that the damaged and/or intact fuel assemblies, and CCs, if applicable, meet the burnup, enrichment and cooling time parameters of Technical Specification *2.1*. The loading plan for 24PTH-S-LC DSC is developed according to Figure P.2-5 for the orientation of fuel assemblies. AMD 11

   - The loading plan is independently verified and approved before the fuel load.

   - A fuel movement schedule is then written, verified and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance of the fuel movement schedule.

   - If loading damaged fuel assemblies, verify that the required number of bottom end caps are installed in appropriate fuel compartment tube locations.

6. Prior to loading of a spent fuel assembly (and CCs, if applicable) into the DSC, the identity of the assembly (and CCs, if applicable) is to be verified by two individuals using an underwater video camera or other means. Verification of CC identification is optional if the CC has not been moved from the host fuel assembly since it's last verification. Read and record the identification number from the fuel assembly (and CCs, if applicable) and check this identification number against the DSC loading plan which indicates which fuel assemblies (and CCs, if applicable) are acceptable for dry storage.

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Step 6 for each SFA loaded into the DSC. A maximum of 12 damaged fuel assemblies may be loaded into the basket per Technical Specification *2.1*. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly and CCs, if applicable, in the DSC. If loading damaged fuel assemblies, place top end caps over each damaged fuel assembly placed into the basket. AMD 11 & 72.48

8. After all the SFAs and CCs, if applicable, have been placed into the DSC and their identities verified, position the lifting yoke and the top shield plug and lower the shield plug onto the DSC.

   CAUTION: Verify that all the lifting height restrictions as a function of temperature specified in Technical Specification *5.3.1.A* can be met in the following steps which involve lifting of the TC. AMD 11

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

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

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

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

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

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

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

16. If loading 24PTH-S DSC or 24PTH-L DSC, drain water from the DSC as necessary to meet the plant lifting crane capacity limits.

    CAUTION: *The radiation dose rates around the surface of the transfer cask without water in the neutron shield and/or in the DSC cavity (through step P.8.1.2.19) are expected to be high. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.*

17. Lift the TC from the fuel pool. As the cask is raised from the pool, continue to spray the cask with demineralized water.

18. Move the TC with loaded DSC to the cask decon area.

19. If applicable to keep the occupational exposure ALARA, replace the water removed from the DSC in Step 16 with spent fuel pool water of the proper boron concentration. Fill the neutron shield with demineralized water if it was drained in Step P.8.1.1.5. Temporary shielding may be installed as necessary to minimize personnel exposure.

AMD 11

P.8.1.3 DSC Drying and Backfilling

CAUTION: During performance of steps listed in Section 8.1.3, monitor the Cask/DSC annulus water level and replenish *as* necessary *to maintain cooling.*

1. Check the radiation levels along the perimeter of the cask. The cask exterior surface should be decontaminated as necessary in accordance with the limits specified in Technical Specification *5.2.4.d* for the DSC surfaces. Temporary shielding may be installed as necessary to minimize personnel exposure. AMD 11

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

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3. Disengage the rigging cables from the top shield plug and remove the eyebolts. Disengage the lifting yoke from the trunnions and position it clear of the cask.

4. Decontaminate the exposed surfaces of the DSC shell perimeter and remove the inflatable cask/DSC annulus seal.

4a. In accordance with Technical Specification 5.2.4.a, verify that the neutron shield (NS) is filled before the draining operation in Step 5 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.

5. Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately twelve inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with the Technical Specification *5.2.4.d* limits.

5a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 6 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.

6. Prior to the start of welding operations, drain a minimum of 750 gallons of water from the DSC back into the fuel pool or other suitable location using the VDS or an optional liquid pump. Alternatively, all the water from the DSC may be drained if precautions are taken to keep the occupational exposure ALARA. Only helium may be used to assist in the removal of water.

7. Disconnect hose from the DSC siphon port.

8. Install the automatic welding machine onto the inner top cover plate and place the inner top cover plate with the automatic welding machine onto the DSC. Verify proper fit-up of the inner top cover plate with the DSC shell.

9. Check radiation levels along surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.

   CAUTION: Insert a 1/4-inch flexible tubing of sufficient length and adequate temperature resistance through the vent port such that it terminates just below the DSC shield plug. Connect the flexible tubing to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner cover plate, *in compliance with Technical Specification 5.2.6.* Optionally, other methods may be used for continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner top cover plate, *to comply with the Technical Specification.*

10. Cover the cask/DSC annulus to prevent debris and weld splatter from entering the annulus.

11. Ready the automatic welding machine and tack weld the inner top cover plate to the DSC shell. Install the inner top cover plate weldment and remove the automatic welding machine.

CAUTION: Continuously monitor the hydrogen concentration in the DSC cavity using the flexible tube arrangement or other alternate methods described in Step 9 during the inner top cover plate cutting/welding operations. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [8.2 and 8.3]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with approximately 2-3 psig helium (or any other inert medium) via the 1/4 inch flexible tubing to reduce the hydrogen concentration safely below the 2.4% limit.

12. Perform dye penetrant weld examination of the inner top cover plate weld in accordance with the Technical Specification *5.2.4.b* requirements. AMD 11

13. Connect the VDS to the DSC siphon and vent ports.

14. Install temporary shielding to minimize personnel exposure throughout the subsequent welding operations as required.

14a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 15 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

15. Engage helium supply and open the valve on the vent port and allow *helium* to force the water from the DSC cavity through the siphon port.

16. Once the water stops flowing from the DSC, close the DSC siphon port and disengage the gas source.

16a. Verify that the TC axial surface dose rates are compliant with limits specified in Technical Specification 5.2.4.e. The configuration for determining the TC axial dose rates shall be in accordance with Technical Specification 5.2.4.e. AMD 11

17. Connect the hose from the vent port and the siphon port to the intake of the vacuum pump. Connect a hose from the discharge side of the VDS to the plant's radioactive waste system or spent fuel pool. Connect the VDS to a helium source.

CAUTION: *During the vacuum drying evolution, personnel should be in the area of loading operations, or in nearby low dose areas in order to take proper action in the event of a malfunction.*

18. Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the DSC cavity. The cavity pressure should be reduced in steps of approximately 100 mm Hg, 50 mm Hg, 25 mm Hg, 15 mm Hg, 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less as specified in Technical Specification *3.1.1.*

AMD 11

19. Open the valve to the vent port and allow the helium to flow into the DSC cavity.

20. Pressurize the DSC with helium to about 24 psia not to exceed 34 psia.

21. Helium leak test the inner top cover plate weld for a leak rate of 1×10^{-4} atm cm^3/sec. This test is optional. AMD 11

22. If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.

23. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

24. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored. When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less in accordance with Technical Specification *3.1.1* limits. AMD 11

25. Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC to 2.5 psig ± 1.0 psig in accordance with Technical Specification *3.1.2.b* limits. AMD 11

26. Close the valves on the helium source.

27. Decontaminate as necessary, and store.

P.8.1.4 <u>DSC Sealing Operations</u>

CAUTION: During performance of steps listed in Section P.8.1.4, monitor the *c*ask/DSC annulus water level and replenish as necessary to maintain cooling. AMD 11

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports, inject helium into blind space just prior to completing welding, and perform a dye penetrant weld examination in accordance with the Technical Specification *5.2.4.b* requirements. *Use of an optional test head is acceptable to perform the helium leak test of the inner top cover plate and vent/siphon port welds in accordance with Technical Specification 5.2.4.c. If an optional test head is not used, proceed to Step 2.* AMD 11 & 72.48

2. Temporary shielding may be installed as necessary to minimize personnel exposure. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Verify proper fit up of the outer top cover plate with the DSC shell.

3. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

4. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification *5.2.4.c* limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.4]. Alternatively this can be done with a test head in P.8.1.4 *step 1.* AMD 11 AMD 11

5. If a leak is found, remove the outer cover plate root pass, the vent and siphon port plugs and repair the inner cover plate welds. Repeat procedure steps from P.8.1.3 Step 18.

6. Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification *5.2.4.b* requirements. AMD 11

7. Seal weld the prefabricated plug over the outer cover plate test port and perform dye penetrant weld examinations.

8. Remove the automatic welding machine from the DSC.

8a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 9 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

9. Open the cask drain port valve and drain the water from the cask/DSC annulus.

10. Rig the cask top cover plate and lower the cover plate onto the TC.

11. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

CAUTION: Monitor the applicable time limits of Technical Specification *3.1.3* until the completion of DSC transfer Step 6 of Section P.8.1.6. AMD 11

12. *Verify that the TC radial dose rates measured at the surface of the Transfer Cask are compliant with limits specified in Technical Specification 5.2.4.e. The configuration for determining the TC radial surface dose rates shall be in accordance with Technical Specification 5.2.4.e.* AMD 11

P.8.1.5 <u>TC Downending and Transfer to ISFSI</u>

1. If loading with OS197/OS197H/OS197FC TC, drain the TC neutron shield to an acceptable location as required to meet the plant lifting crane capacity limit.

CAUTION: *The radiation dose rates around the surface of the transfer cask without water in the neutron shield (through step P.8.1.5.10) are expected to be high. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.*

2. Re-attach the TC lifting yoke to the crane hook, as necessary. Ready the transfer trailer and cask support skid for service.

3. Move the scaffolding away from the cask as necessary. Engage the lifting yoke and lift the cask over the cask support skid on the transfer trailer.

4. The transfer trailer should be positioned so that cask support skid is accessible to the crane with the trailer supported on the vertical jacks.

5. Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.

6. Move the crane forward while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.

7. Inspect the positioning of the cask to insure that the cask and trunnion pillow blocks are properly aligned.

8. Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.

9. Inspect the trunnions to ensure that they are properly seated onto the skid. Install the trunnion tower closure plates (optional for the OS197 TC and the OS197H TC).

10. Fill the neutron shield, if it was drained in P.8.1.5.1 *step 1, and verify that the NS is filled, in accordance with Technical Specification 5.2.4.a.*

11. Remove the bottom ram access cover plate from the cask. Install the two-piece temporary neutron/gamma shield plug to cover the bottom ram access. Install the ram trunnion support frame on the bottom of the TC. (The temporary shield plug and ram trunnion support frame are not required with integral ram/trailer).

P.8.1.6 DSC Transfer to the HSM

1. Prior to transferring the cask to the ISFSI, remove the HSM door, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.

   Caution: The insides of empty modules have the potential for high does rates due to adjacent loaded modules. Proper ALARA practices should be followed for operations inside these modules and in the areas outside these modules whenever the door from the empty HSM has been removed.

2. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.

   CAUTION: Verify that the requirements of Technical Specification *5.3.1.B*, "TC/DSC Transfer Operations at High Ambient Temperatures" are met prior to next step.

3. Using a suitable vehicle, transfer the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

4. Once at the ISFSI, position the transfer trailer to within a few feet of the HSM.

5. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

All changes on this page are AMD 11

6. Unbolt and remove the cask top cover plate.

   CAUTION: Verify that the applicable time limits of Technical Specification *3.1.3* are met.

7. Back the cask to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Extend the transfer trailer vertical jacks.

8. Connect the skid positioning system hydraulic power unit to the positioning system via the hose connector panel on the trailer, and power it up. Remove the skid tie-down bracket fasteners and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.

9. Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.

10. Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.

11. After the cask is docked with the HSM, verify the alignment of the TC using the optical survey equipment.

12. Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

14. Recheck all alignment marks in accordance with the Technical Specification *5.3.3* limits and ready all systems for DSC transfer.

15. Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening.

19. Install the DSC drop-in retainer through the HSM door opening.

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20. The trailer may be moved as necessary to install the HSM door. Install the HSM door and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Technical Specification*s 5.4.1 and 5.4.2.*

21. Replace the TC top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

22. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

23. Close and lock the ISFSI access gate and activate the ISFSI security measures.

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P.8.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform a daily visual surveillance of the HSM air inlets and outlets to insure that no debris is obstructing the HSM vents in accordance with Technical Specification *5.2.5.a* requirements OR perform a temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification *5.2.5.b* requirements.

essentially identical to those of DSC loading through the DSC weld removal (beginning of preparation to placement of the cask in the fuel pool). Prior to opening the DSC, the following operations are to be performed.

CAUTION: Verify that the applicable time limits of Technical Specification *3.1.3* are met until the completion of Step P.8.2.2.14. AMD 11

1. The TC may now be transferred to the cask handling area inside the plant's fuel/reactor building.

2. Position and ready the trailer for access by the crane and install the ram access penetration cover plate.

3. Attach the lifting yoke to the crane hook.

4. Engage the lifting yoke with the trunnions of the TC.

5. Visually inspect the yoke lifting hooks to insure that they are properly aligned and engaged onto the TC trunnions.

6. If unloading with OS197/OS197H/OS197FC TC, drain the TC water from the neutron shield to an acceptable location as required to meet the plant lifting crane capacity limit.

   CAUTION: The radiation dose rates around the surface of the transfer cask without water in the neutron shield (through step P.8.2.2.9) are expected to be high. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

7. Lift the TC approximately one inch off the trunnion supports. Visually inspect the yoke lifting hooks to insure that they are properly positioned on the trunnions.

8. Move the crane backward in a horizontal motion while simultaneously raising the crane hook vertically and lift the TC off the trailer. Move the TC to the cask decon area.

9. Lower the TC into the cask decon area in the vertical position. Fill the neutron shield with water if it was drained in Step P.8.2.2.6, *and verify that the NS is filled, in accordance with Technical Specification 5.2.4.a.* AMD 11

10. Wash the TC to remove any dirt which may have accumulated on the TC during the DSC loading and transfer operations.

11. Place scaffolding around the TC so that any point on the surface of the TC is easily accessible to handling personnel.

12. Unbolt the TC top cover plate.

13. Connect the rigging cables to the TC top cover plate and lift the cover plate from the TC. Set the TC cover plate aside and disconnect the lid lifting cables.

14. Install temporary shielding to reduce personnel exposure as required. Fill the TC/DSC annulus with clean demineralized water and seal the annulus.

The process of DSC unloading is similar to that used for DSC loading. DSC opening operations described below are to be carefully controlled in accordance with plant procedures. This operation is to be performed under the site's standard health physics guidelines for welding, grinding, and handling of potentially highly contaminated equipment. These are to include the use of prudent housekeeping measures and monitoring of airborne particulates. Procedures may require personnel to perform the work using respirators or supplied air.

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If fuel needs to be removed from the DSC, either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of damaged or oxidized fuel and to prevent radiological exposure to personnel during this operation. A sampling of the atmosphere within the DSC will be taken prior to inspection or removal of fuel.

If the work is performed outside the fuel/reactor building, a tent may be constructed over the work area, which may be kept under a negative pressure to control airborne particulates. Any radioactive gas release will be Kr-85, which is not readily captured. Whether the krypton is vented through the plant stack or allowed to be released directly depends on the plant operating requirements.

Following opening of the DSC, the cask and DSC are filled with water prior to lowering the top of cask below the surface of the fuel pool to prevent a sudden inrush of pool water. Cask placement into the pool is performed in the usual manner. Fuel unloading procedures will be governed by the plant operating license under 10CFR50. The generic procedures for these operations are as follows:

15. Locate the DSC siphon and vent port using the indications on the top cover plate. Place a portable drill press on the top of the DSC. Position the drill with the siphon port.

16. Place an exhaust hood or tent over the DSC, if necessary. The exhaust should be filtered or routed to the site radwaste system.

17. Drill a hole through the DSC top cover plate to expose the siphon port quick connect.

18. Drill a second hole through the top cover plate to expose the vent port quick connect.

19. Obtain a sample of the DSC atmosphere, if necessary (e.g., at the end of service life). Fill the DSC with water from the fuel pool through the siphon port with the vent port open and routed to the plant's off-gas system.

    CAUTION:

    (a) The water fill rate must be regulated during this reflooding operation to ensure that the DSC vent pressure does not exceed 20.0 psig.

    (b) In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 19(c) is initiated and continually monitored

during the first five minutes of the draining evolution to ensure the NS remains filled.

(c) Provide for continuous hydrogen monitoring of the DSC cavity atmosphere during all subsequent cutting operations to ensure that a safety limit of 2.4% is not exceeded [8.2 and 8.3] *and in compliance with Technical Specification 5.2.6.* Drain appropriate amount of water from the DSC cavity before cutting operations to ensure that sufficient free volume exists in the DSC cavity for H_2 concentration limit. Purge with 2-3 psig helium (or any other inert medium) as necessary to maintain the hydrogen concentration safely below this limit.

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20. Place welding blankets around the cask and scaffolding.

21. Using plasma arc-gouging, a mechanical cutting system or other suitable means, remove the seal weld from the outer top cover plate and DSC shell. A fire watch should be placed on the scaffolding with the welder, as appropriate. The exhaust system should be operating at all times.

22. The material or waste from the cutting or grinding process should be treated and handled in accordance with the plant's low level waste procedures unless determined otherwise.

23. Remove the top of the tent, if necessary.

24. Remove the exhaust hood, if necessary.

25. Remove the DSC outer top cover plate.

26. Reinstall tent and temporary shielding, as required. Remove the seal weld from the inner top cover plate to the DSC shell in the same manner as the top cover plate. Remove the inner top cover plate. Remove any remaining excess material on the inside shell surface by grinding.

27. Clean the cask surface of dirt and any debris which may be on the cask surface as a result of the weld removal operation. Any other procedures which are required for the operation of the cask should take place at this point as necessary.

28. Engage the yoke onto the trunnions, install eyebolts into the top shield plug and connect the rigging cables to the eyebolts.

29. Visually inspect the lifting hooks or the yoke to insure that they are properly positioned on the trunnions.

29a. If the neutron shield is to remain filled during Step 30, in accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 30 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.

30. If unloading 24PTH-S DSC or 24PTH-L DSC, drain a minimum of 750 gallons of water from the DSC. The neutron shield water from the TC may also need to be drained as required, to meet plant crane limits.

    CAUTION: The radiation dose rates around the surface of the transfer cask without water in the neutron shield and/or in the DSC cavity (through step P.8.2.2.37) are expected to be high. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

31. The cask should be lifted just far enough to allow the weight of the TC to be distributed onto the yoke lifting hooks. Inspect the lifting hooks to insure that they are properly positioned on the trunnions.

32. Install suitable protective material onto the bottom of the TC to minimize cask contamination. Move the cask to the fuel pool.

33. Prior to lowering the cask into the pool, adjust the pool water level, if necessary, to accommodate the volume of water which will be displaced by the cask during the operation.

34. Lower the cask into the fuel pool leaving the top surface of the cask approximately one foot above the surface of the pool water.

35. Fill the DSC with appropriate amount pool water.

P.8.3 Identification of Subjects for Safety Analysis

No Change *to Section 5.1.3.*

P.8.4 Fuel Handling Systems

No Change *to Section 5.2.*

P.8.5 Other Operating Systems

No Change *to Section 5.3.*

P.8.6 Operation Support System

No Change *to Section 5.4.*

P.8.7 Control Room and/or Control Areas

No Change *to Section 5.5.*

P.8.8 Analytical Sampling

No Change *to Section 5.6.*

P.9 Acceptance Tests and Maintenance Program

P.9.1 Acceptance Tests

The acceptance requirements for the NUHOMS®-24PTH system are given in the *U*FSAR except as described in the following sections. The NUHOMS®-24PTH DSC has been enhanced to provide leaktight confinement and the basket includes an updated poison plate design. The requirements for the poison plate material acceptance tests *and the NUHOMS®-24PTH DSC welds for the 24PTH system are described.*

All changes on this page are AMD 11.

P.9.1.1 Visual Inspection

Visual examinations are performed at the fabricator's facility to ensure that the NUHOMS®-24PTH system components conform to the fabrication specifications and drawings.

P.9.1.2 Structural Tests

The NUHOMS®-24PTH DSC confinement welds are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Section III, Subsection NB [9.1] with exceptions as listed in Section P.3.1. The following requirements are unique to the NUHOMS®-24PTH DSC:

- The inner bottom cover weld is inspected in accordance with Article NB-5231 when the weld joint design is per Figure NB-4243-1,
- The outer bottom cover weld is penetrant tested, and
- The outer top cover plate weld root and cover are penetrant tested.

The NUHOMS®-24PTH DSC basket is designed, fabricated, and inspected in accordance with ASME B&PV Code Section III, Subsection NG [9.1] with exceptions as listed in Section P.3.1.

P.9.1.3 Leak Tests

The NUHOMS®-24PTH DSC confinement boundary is leak tested to verify that it is leaktight in accordance with the criteria of ANSI N14.5 [9.2]. The personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.8].

The leak tests are typically performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved.

P.9.1.4 Component Tests

The NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. The gaskets in the transfer cask do not require acceptance testing other than the leak testing cited above. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

P.9.1.5 Shielding Integrity Tests

The transfer cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6"x 6" grid, the detector will encompass a 6"x 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

All changes on this page are AMD 11

P.9.1.6 Thermal Acceptance Tests

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section P.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section P.9.1.7.6.

P.9.1.7 Poison Acceptance

CAUTION
Sections P.9.1.7.1 through P.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated *a*luminum

(b) Boron carbide/*a*luminum metal matrix composite (MMC)

(c) BORAL®

The 24PTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials is given in Table P.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to *BORAL®*, which is described later in this section.

All changes on this page are AMD 11

P.9.1.7.1 Borated Aluminum

See the Caution in Section P.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur)*. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section P.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

P.9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMC)

See the Caution in Section P.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 24PTH DSC, MMCs shall pass the qualification testing specified in Section P.9.1.7.*8*, and shall subsequently be subject to the process controls specified in Section P.9.1.7.*9*.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section P.9.1.7.*7*. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

All changes on this page are AMD 11

P.9.1.7.3 **BORAL®**

See the Caution in Section P.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. *Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns.* The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of BORAL®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken *from* the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

P.9.1.7.4 *Visual Inspections of Neutron Absorbers*

See the Caution in Section P.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Material that does not meet the following acceptance criteria shall be reworked, repaired, or scrapped. Blisters shall be treated as non-conforming. Inspection of MMCs with an integral aluminum cladding shall also include verification that the matrix is not exposed through the faces of the aluminum cladding and that solid aluminum is not present at the edges. For BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet.

P.9.1.7.5 *Other Visual Inspections Criteria (non-Technical Specifications)*

All changes on this page are AMD 11

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

P.9.1.7.6 *Thermal Conductivity Testing of Poison Plates*

Testing shall conform to ASTM E1225[1]*, ASTM E1461*[2]*, or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Previous testing of borated aluminum and metal matrix composite shows that thermal conductivity increases slightly with temperature. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After 25 tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C*,* TiB_2*, or* AlB_2*, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.*

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section P.4.3.

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section P.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

1 *ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"*

2 *ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"*

P.9.1.7.7 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

All changes on this page are AMD 11

> ***CAUTION***
>
> *Portions of Section P.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

a) **Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.**

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to *1.1 inch* diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. *Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.*

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be *no more than 0.75 sq. inch*.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. *If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the*

one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications.

The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor for a normal distribution with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than *the statistically derived minimum thickness from P.9.1.7.7 a)* or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. *Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.*

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

P.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

CAUTION
Section P.9.1.7.8.3.1, Section P.9.1.7.8.4 and Section P.9.1.7.8.5 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

P.9.1.7.8.1 Applicability and Scope

Metal matrix composites (MMCs) *acceptable for use in the 24PTH DSC are described in Section P.9.1.7.2.*

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section P.9.1.7.*9* so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

All changes on this page are AMD 11

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the *C*ertificate *H*older.

All changes on this page are AMD 11

P.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in SectionP.9.1.7.*8*.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section P.9.1.7.*8*.5.

P.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

*T*hermal damage and corrosion (hydrogen generation) testing shall be *performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.*

Thermal damage testing is not required for *unclad* MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for full density MMCs *(clad or unclad)* consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

P.9.1.7.8.3.1 ***Delamination Testing of Clad MMC***

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage.

P.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from *approximately* the two ends and middle of the *qualification* material run shall be subject to:

a) room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:

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[3] Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

[4] Boralyn testing submitted to the NRC under docket 71-1027, 1998.

[5] ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture,

b) Testing to verify more than 98% of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a means to be approved by the Certificate Holder. The maximum interconnected porosity is 0.5 volume %,

and for at least one sample,

c) For MMCs with an integral aluminum cladding, thermal durability testing demonstrating that after a minimum 24 hour soak in either pure or borated water, then insertion into a preheated oven at approximately 825°F for a minimum of 24 hours, the specimens are free of blisters and delamination and pass the mechanical testing requirements described in test 'a' of this section.

All changes on this page are AMD 11

P.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section P.9.1.7.7, or by chemical analysis for boron carbide content in the composite.

P.9.1.7.8.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

6 *ASTM E290, Standard Methods for Bend Testing of Materials for Ductility*
7 ASTM E94, Recommended Practice for Radiographic Testing
8 ASTM E142, Controlling Quality of Radiographic Testing
9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

P.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION *Sections P.9.1.7.9.1 and P.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

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P.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section P.9.1.7.*8* is required, depending on the characteristics of the material that could be affected by the process change.

P.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, *reduce corrosion resistance,* reduce the mechanical strength or ductility of the MMC.

P.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section P.9.1.7.*9*.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that *are* established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a *magnesium-alloyed* aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, *and*

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

In no case shall process changes be accepted if they result in a product outside the limits in Sections *9.5.3.1* and *9.5.3.4*.

All changes on this page are AMD 11

P.9.2 Maintenance Program

NUHOMS®-24PTH system is a totally passive system and therefore requires little, if any, maintenance over the lifetime of the ISFSI. Typical NUHOMS®-24PTH system maintenance tasks are performed in accordance with the *U*FSAR.

All changes on this page are AMD 11

P.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition including 2000 addenda.

9.2 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.3 *Deleted.*

9.4 *Deleted.*

9.5 "Aluminum Standards and Data, 2003" The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 *Deleted.*

9.8 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.9 *Deleted.*

9.10 *Deleted.*

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Table P.9-1
B10 Specification for the NUHOMS®-24PTH Poison Plates

Poison Type	24PTH Basket Type	Minimum Poison Loading (B10 mg/cm^2)	% Credit Used in Criticality Analysis
Borated Aluminum /MMC	1A or 2A	7	90
	1B or 2B	15	
	1C or 2C	32	
BORAL®	1A or 2A	9	75
	1B or 2B	19	
	1C or 2C	40	

All changes on this page are AMD 11

All changes on this page are AMD 11

P.11.1.2 Extreme Temperatures

No change. The off-normal maximum ambient temperature of 125°F is used in Section 8.1.2.2. For the NUHOMS®-24PTH system, a maximum ambient temperature of 117°F is used. Therefore, the analyses in Section 8.1.2.2 bound TCs and HSM Model 102 used in the NUHOMS®-24PTH system.

P.11.1.2.1 Postulated Cause of Event

No change. See Section 8.1.2.2.

P.11.1.2.2 Detection of Event

No change. See Section 8.1.2.2.

P.11.1.2.3 Analysis of Effects and Consequences

The thermal evaluation of the NUHOMS®-24PTH system for off-normal conditions is presented in Section P.4. The 100°F normal condition with insolation bounds the 117°F case without insolation for the DSC in the TC. Therefore the normal condition maximum temperatures are bounding. The 117°F case with the DSC in the HSM-H is not bounded by the normal conditions and therefore evaluated in Section P.4.

The NUHOMS® standardized TC and HSM Model 102 were evaluated for a maximum heat load of 24 kW and maximum off-normal ambient temperature of 125°F. The maximum heat load of the 24PTH-S-LC DSC in standardized TC or HSM Model 102 is limited to 24 kW. Therefore the evaluation presented in Section 8.1.2.2 is bounding for these components.

The structural evaluation of the 24PTH DSC for off-normal temperature conditions is presented in Section P.3.6.2.2. The structural evaluation of the basket due to off-normal thermal conditions is presented in Section P.3.6.1.3. The structural evaluation of HSM-H and OS197FC Transfer Cask for off-normal conditions with 24PTH DSC are presented in Section P.3.6.

P.11.1.2.4 Corrective Actions

Restrictions for onsite handling of the TC with a loaded DSC under extreme temperature conditions are presented in Technical Specifications *5.3.1.A* and *5.3.1.B*. There is no change to this requirement as a result of addition of the NUHOMS®-24PTH DSC.

P.11.1.3 Off-Normal Releases of Radionuclides

The NUHOMS®-24PTH DSC is designed and tested to the leak tight criteria of ANSI N14.5 [11.2]. Therefore the estimated quantity of radionuclides expected to be released annually to the environment due to normal or off-normal events is zero.

Decay heat is rejected from the DSC to the HSM Model 152 air space by convection and then is removed from the HSM by a natural circulation air flow. Heat is also radiated from the DSC surface to the heat shield and HSM walls where again the natural convection air flow and conduction through the walls removes the heat. Figure R.1-3 shows the ventilation flow paths for the DSC and the HSM Model 152. The passive cooling system for the HSM Model 152 is designed to assure that peak cladding temperatures during long term storage remain below acceptable limits to ensure fuel cladding integrity.

The NUHOMS® HSM Model 152 provides an independent, passive system with substantial structural capacity to ensure the safe dry storage of spent fuel assemblies. To this end, the HSMs are designed to ensure that normal transfer operations and postulated accidents or natural phenomena do not impair the DSC or pose a hazard to plant personnel.

The HSM Model 152s are constructed on a non-safety load bearing foundation, which consists of a reinforced concrete basemat on compacted engineered fill. The HSMs are located in a fenced, secured location with controlled access.

R.1.2.2 Operational Features

R.1.2.2.1 General Features

The HSM Model 152 is designed to safely store a DSC with a maximum weight up to 105 kips which includes the 24P, 52B, 61BT, 24PT2, 32PT, and 24PHB DSCs. The HSM Model 152 protects the DSC from the potentially adverse effects of natural phenomena hazards, such as earthquake, tornado, tornado missiles, flood, and elevated temperatures. In addition, the HSM Model 152 dissipates decay heat from the spent fuel by a combination of radiation, conduction, and convection. Natural convection air flow enters the bottom front wall of the HSM, circulates around the DSC, and exits through the flow channel at the rear of the HSM roof slab. The cross-sectional areas of the air inlet and outlet openings, and the interior flow paths are designed to optimize ventilation air flow in the Model 152 for decay heat removal including worst-case extreme summer ambient conditions. Furthermore, like the HSM Model 80 and Model 102, a thermal radiation heat shield is used in the Model 152 to reduce the HSM concrete temperatures to acceptable limits for all thermal conditions.

R.1.2.2.2 Sequence of Operations

The sequence of operations to be performed in loading a DSC containing spent nuclear fuel into the NUHOMS® HSM Model 152 is presented in Chapter R.8.

R.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

R.1.2.2.3.1 Criticality Prevention

No change *to Section 5.1.3.1.*

R.1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the NUHOMS® HSM Model 152 System.

R.1.2.2.3.3 Operation Shutdown Modes

The NUHOMS® HSM Model 152 is a totally passive system so that consideration of operation shutdown modes is unnecessary.

R.1.2.2.3.4 Instrumentation

No change *to Section 5.1.3.4.*

R.1.2.2.3.5 Maintenance Techniques

No change *to Section 5.1.3.5.*

R.1.2.3 Cask Contents

No change *to Section 1.2.3.*

R.1.3 Identification of Agents and Contractors

Transnuclear, Inc. (TN) provides the design, analysis, licensing support and quality assurance for the NUHOMS® HSM Model 152 System. Fabrication of the NUHOMS® HSM Model 152 is done by one or more qualified fabricators under TN's quality assurance program described in Chapter R.13. This program is written to satisfy the requirements of Subpart G of 10CFR72, [1.2] and covers control of design, procurement, fabrication, inspection, testing, operations and corrective action. Experienced TN operations personnel will assist in the preparation of generic operating procedures and provide training to utility personnel prior to their first use of the NUHOMS® HSM Model 152 System.

Managerial and administrative controls, which are used to ensure safe operation of the casks, will be provided by the host utility. NUHOMS® HSM Model 152 System operations and maintenance will be performed by utility personnel. Decommissioning activities will be performed by utility personnel in accordance with site procedures.

TN provides specialized services for the nuclear fuel cycle that support transportation, storage and handling of spent nuclear fuel, radioactive waste and other radioactive materials. TN is the holder of NUHOMS® CoC 1004 [1.3].

R.1.4 Generic Cask Arrays

No change *to Section 1.2.1.*

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
2	REVISED PER FCN 721004-1218 AND EDITORIAL CORRECTIONS ON SHEET 8	01/09/14
1	REVISED PER FCN 721004-562 AND 721004-688 REV 2 , UPDATED TITLE BLOCK	01/28/10

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
GENERAL LICENSE
NUHOMS® HSM MODEL 152
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH-03-6400-SAR	NONE	1 OF 9

NUH-03-6400-SAR 1 OF 9

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-6400-SAR	5 OF 9	2

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-6400-SAR	6 OF 9	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-6400-SAR	7 OF 9	2

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

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PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-6400-SAR	9 OF 9	2

Table R.1-1
Comparison of Key Parameters of NUHOMS® HSM Model 152 Versus HSM Model 80 and Model 102

Characteristic	HSM Model 80	HSM Model 102	HSM Model 152
Overall Length (without Shield Walls)	19'-10"	19'-10"	19'-7"
Overall Width (without Shield Walls)	9'-8"	9'-8"	8'-5"
Overall Height	15'-0"	15'-0"	17'-10"
Roof Thickness	3'-0"	3'-0"	5'-8"
End Shield Wall Thickness	2'-0"	2'-0"	3'-0"
Rear Shield Wall Thickness	2'-0"	2'-0"	3'-0"
Side Wall Thickness	1'-6"	1'-6"	1'-0
Back Wall Thickness	1'-0"	1'-0"	1'-0"
Front Wall Thickness	2'-6"	2'-6"	2'-6"
Floor Thickness	1'-0"	1'-0"	N/A
Door Construction	~ 8" thick consisting of concrete core (~ 6") encased by stainless steel (2")	24" thick consisting of reinforced concrete	24" thick consisting of reinforced concrete
Inlet Vent Configuration	4 along lower side walls	4 along lower side walls	1 along bottom front wall
Inlet Vent Area	1200 in^2	1200 in^2	792 in^2
Outlet Vent Configuration	4 along upper side walls	4 along upper side walls	1 along interface of roof and rear wall
Outlet Vent Area	1680 in^2	1680 in^2	608 in^2
Gap Between Adjacent Modules Placed Side-By-Side	6"	6"	0"
Bird Screen Type	Wire Cloth 3/4" mesh x *0.120*" wire	Wire Cloth 3/4" mesh x *0.120*" wire	Wire Cloth 3/4" mesh x 0.080" wire
Weight – Base Unit (including HSM support steel)	164,403 (1)	167,267 (1)	175,678
Weight – Roof	80,970 (1)	82,486 (1)	134,043
Weight Door	6,556	11,200	12,599
DSC Support Steel Configuration	Structural steel frame with rails installed to permit sliding of DSC	Structural steel frame with rails installed to permit sliding of DSC	Guide rails bolted to concrete to permit sliding of DSC
Heat Shield Thickness	12 Gauge (0.1054") Galvanized Steel	12 Gauge (0.1054") Galvanized Steel	12 Gauge (0.1054") Stainless Steel

Note: (1) Based on BWR dimensions and weights which envelope the PWR dimensions and weights.

R.2 Principal Design Criteria

This section provides the principal design criteria for the NUHOMS® HSM Model 152 System. The principal design criteria for the NUHOMS® HSM Model 152 are the same as the NUHOMS® HSM Model 80 and Model 102 as described in Chapter 3. Section R.2.1 presents a general description of the spent fuel to be stored. Section R.2.2 provides the design criteria for environmental conditions and natural phenomena. Section R.2.3 provides a description of the systems which have been designated as important to safety. Section R.2.4 discusses decommissioning considerations. Section R.2.5 summarizes the NUHOMS® HSM Model 152 design criteria.

R.2.1 Spent Fuel To Be Stored

The NUHOMS® DSCs are designed to store a total of 24 or 32 PWR fuel assemblies and 52 or 61 BWR fuel assemblies with the same characteristics as those described, respectively, in Chapter 3 and Appendices Chapters K.2, L.2, M.2, and N.2.

R.2.1.1 General Operating Functions

No change *to Section 3.1.2.*

R.2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The NUHOMS® HSM Model 152 is handled and utilized in the same manner as the existing NUHOMS® HSM Model 80 and Model 102 Systems. The environmental conditions, natural phenomena and design criteria are the same as described for the NUHOMS® HSM Model 80 and Model 102 in Chapter 3. Design criteria for the NUHOMS® DSC and TC remain unchanged.

R.2.2.1 Tornado Wind and Tornado Missiles

No change *to Section 3.2.1.*

R.2.2.2 Water Level (Flood) Design

No change *to Section 3.2.2.*

R.2.2.3 Seismic Design

No change *to Section 3.2.3.*

R.2.2.4 Snow and Ice Loading

No change *to Section 3.2.4.*

R.2.2.5 Combined Load Criteria

No change *to Section 3.2.5.*

R.2.3 Safety Protection Systems

R.2.3.1 General

No change *to Section 3.3.1.*

R.2.3.2 Protection By Multiple Confinement Barriers and Systems

No change *to Section 3.3.2.*

R.2.3.3 Protection By Equipment and Instrumentation Selection

No change *to Section 3.3.3.*

R.2.3.4 Nuclear Criticality Safety

R.2.3.4.1 Control Methods for Prevention of Criticality

No change *to Section 3.3.4.*

R.2.3.4.2 Error Contingency Criteria

No change *to Section 3.3.4.*

R.2.3.4.3 Verification Analysis-Benchmarking

No change *to Section 3.3.4.*

R.2.3.5 Radiological Protection

No change *to Section 3.3.5.*

R.2.3.6 Fire and Explosion Protection

No change *to Section 3.3.6.*

R.2.4 Decommissioning Considerations

No change *to Section 3.5.*

R.2.5 Summary of NUHOMS® HSM Model 152 Design Criteria

The principal design criteria for the NUHOMS® HSM Model 152 are the same as those presented for the NUHOMS® HSM Model 80 and Model 102 in Chapter 3. The NUHOMS® HSM Model 152 is designed to store a DSC loaded with PWR or BWR fuel assemblies identical to those stored in a NUHOMS® HSM Model 80 or Model 102 as described in Chapter 3 and Appendices Chapters K.2, L.2, M.2, and N.2.

The HSM Model 152 has a single air inlet vent located at the lower front wall of the base unit and a single air outlet vent located at the rear of the roof slab. A roof vent shield cap is installed above the outlet vent to provide additional shielding.

For thermal protection of the HSM Model 152 concrete, stainless steel or carbon steel heat shields or metal-coated carbon steel heat shields are installed on the sidewalls of the base unit. Heat shields are also installed under the roof. The heat shields guide cooling ventilation airflow through the HSM Model 152.

The HSM Model 152 front door is a two-foot thick composite door (i.e., reinforced concrete and steel) which provides missile protection and shielding for the DSC. The door is rectangular on the outside of the opening and it is circular at the rear where it fits into the opening in the base unit. In addition, a ½ to 3/4-inch thick circular steel plate is attached to the rear of the door.

During DSC insertion/retrieval operations, the TC is docked with the HSM Model 152 docking surface and mechanically secured to the HSM Model 152 cask restraint embedments provided in the lower front wall of the HSM Model 152 base unit. These embedments are equally spaced on either side of the HSM Model 152 access opening and serve to restrain the transfer trailer skid during insertion/retrieval of the DSC.

R.3.1.2 Design Criteria

The design criteria for the HSM Model 152 are provided in Section R.2.2. The design criteria for the DSC and TC are not changed.

R.3.2 Weights

Table R.3-1 shows the weights of the various components of the NUHOMS® HSM Model 152 system. The dead weights of the components are determined based on the nominal dimensions.

R.3.3 Mechanical Properties of Materials

The material and section properties used for different components of the HSM Model 152 and the internal DSC support structure are provided in Table 8.1-3. The temperature-dependent material properties for concrete, ASTM A36 carbon steel, and ASME SA240, Type 304 and SA479 stainless steel are also provided in Table 8.1-3.

R.3.4 General Standards for Casks

No change to *Sections 3.4.1, K.3.4, M.3.4, P.3.4.1.*

R.3.5 Fuel Rods

No change to *Section 3.1.1.*

R.3.6 Structural Analysis (Normal and Off-Normal Operations)

In accordance with NRC Regulatory Guide 3.48 [3.1] the design events identified by ANSI/ANS 57.9-1984, [3.2] form the basis for the accident analyses performed for the standardized NUHOMS® system. Four categories of design events are defined. Design event Types I and II cover normal and off-normal events and are addressed in Section 8.1. Design event Types III and IV cover a range of postulated accident events and are addressed in Section 8.2. The purpose of this section of the Appendix is to present the structural analyses for normal and off-normal operating conditions for the NUHOMS® HSM Model 152 system using a format similar to the one used in Section 8.1 for analyzing the NUHOMS® HSM Model 80 and Model 102 systems.

R.3.6.1 Normal Operation Structural Analysis

Table 8.1-1 shows the normal operating loads for which the NUHOMS® safety-related components are designed. The table also lists the individual NUHOMS® components which are affected by each loading. The magnitude and characteristics of each load are described in Section R.3.6.1.1.

The method of analysis and the analytical results for each load are described in Section R.3.6.1.2.

R.3.6.1.1 Normal Operating Loads

The normal operating loads for the NUHOMS® system components are:

1. Dead Weight Loads
2. Design Basis Internal and External Pressure Loads
3. Design Basis Thermal Loads
4. Operational Handling Loads
5. Design Basis Live Loads

These loads are described in detail in the following paragraphs.

A. Dead Weight Loads

Table R.3-1 shows the weights of various components of the NUHOMS® HSM Model 152 system. The deadweight of the component materials is determined based on nominal component dimensions.

B. Design Basis Internal and External Pressure

No change *to Section 8.1.1.1B.*

The factor of safety (F.S.) against overturning for a single, freestanding HSM Model 152 with the shield walls due to the postulated design basis flood water velocity is given by:

$$F.S. = 23,161 / 17,542 = 1.32$$

The required minimum factor of safety against overturning is 1.1 [3.2]. Therefore, the overturning factor of safety is within the allowable value.

R.3.7.4.3 HSM Model 152 Flooding Sliding Analysis

The factor of safety against sliding of a freestanding HSM Model 152 due to the maximum postulated flood water velocity of 15 fps is calculated using methods similar to those described above. The effective weight of the HSM Model 152 including the DSC and end shield wall acting vertically downward, less the effects of buoyancy acting vertically upward is 329.5 K. The friction force resisting sliding of the HSM Model 152 is equal to the product of the net weight of the HSM Model 152 and DSC and the coefficient of friction for concrete placed against another concrete surface such as that between the HSM Model 152 and basemat, which is 0.6 [3.7]. Therefore, the force resisting sliding of the HSM Model 152 is 0.6 x 329.5 or 197.7 kips. The drag force acting on a HSM Model 152 is 8.07 kips/ft x 19.583 = 158.1 kips total acting on the side wall of a single HSM Model 152, due to a flood velocity of 15 fps. The resulting factor of safety against sliding of a free standing HSM Model 152 due to the design basis flood water velocity is 1.25. The required minimum factor of safety against sliding = 1.1 [3.2]. Therefore, the sliding factor of safety is within the allowable value.

R.3.7.5 Lightning

No change *to Section 8.2.6.*

R.3.7.6 Blockage of HSM Model 152 Air Inlet and Outlet Openings

This accident conservatively postulates the complete blockage of the HSM Model 152 ventilation air inlet and outlet openings on the HSM Model 152. Since the NUHOMS® HSM Model 152s are located outdoors, there is a remote probability that the ventilation air inlet and outlet openings could become blocked by debris. The NUHOMS® design features such as the perimeter security fence, the above ground location of the air inlet opening and protected location of the outlet vent opening and the vent screens reduces the probability of occurrence of such an accident. Nevertheless, for this conservative generic analysis, such an accident is postulated to occur and is analyzed.

The structural consequences due to the weight of the debris blocking the air inlet and outlet vent openings are negligible and are bounded by the HSM Model 152 loads induced for a postulated tornado (Section 8.2.2) or earthquake (Section 8.2.3).

The thermal effects of this accident for various NUHOMS® DSCs with a 24 kW heat load are described in Sections R.4 and R.11.

The canister stop plates are loaded by the normal and off-normal handling loads and seismic loads. The normal handling load during the insertion of the DSC is 60 kips on both of the rails. The maximum off-normal handling load is 80 kips on one rail. The seismic load considering a conservative factor of 1.5 is 95.625 kips acting on each plate. Stresses in the canister stop plates, rail-to-canister stop end plate weld, and canister stop end plate-to-stiffener plate welds are all determined to be less than the specified allowables.

R.3.7.8.8 Thermal Cycling of the HSM Model 152

No change *to Section 8.2.10.5.*

R.3.7.8.9 Evaluation of HSM Model 152 Concrete Components with Temperature Exceeding Code Limits

The maximum concrete temperature under normal and off-normal condition for the HSM Model 152 are 221°F and 231/234°F (for 117°F and 125°F ambient conditions), respectively. These temperatures exceed 200°F in normal condition and 225°F in off-normal condition, but do not exceed 300°F. Therefore, as specified in CoC 1004 SER [3.9], no tests or reduction in concrete strength are required to demonstrate the capability of the concrete to adequately handle the elevated temperatures provided Type II cement is used and special aggregates are selected which are acceptable for concrete in this temperature range. This approach is consistent with standardized HSM design, for which special aggregates for the roof concrete mix are provided.

The maximum concrete temperature for a 40-hour blocked vent condition is 394/397°F (for 117°F and 125°F ambient conditions), which exceeds the 350°F limit specified in CoC 1004 SER [3.9]. As noted in the CoC 1004 SER [3.9], use of any Portland cement concrete where accident temperature exceeds 350°F will require testing be performed on the exact concrete mix. Elevated temperature testing of the exact concrete mix (cement type, additives, water-cement ratio, aggregates, proportions) is to be performed for the HSM Model 152. The use of high temperature concrete testing is explicitly accepted by the NRC, as documented in the NRC's SER, Section 3.0, Page 3-5. The testing shall demonstrate the level of strength reduction is less than that which was applied (10% in the calculation), and show that the increased temperatures do not cause deterioration of the concrete.

DSC stored in HSM Model 152 does not exceed 100°F for 5-year old or greater cooled fuel when the DSC is fully loaded with 24 KW heat load.

All changes on this page are AMD 11

R.4.4.5 References

[4.1] "Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," Transnuclear West Report NUH-003, Revision 8, File Number NUH003.0103, June 2004.

[4.2] Certificate of Compliance No. 1004 for Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Docket No. 72-1004, Amendment No. 7, Effective Date 3/2/04.

[4.3] Safety Evaluation Report of Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, U.S. Nuclear Regulatory Commission, December 1994.

R.9 Acceptance Tests and Maintenance Program

R.9.1 Acceptance Tests

The addition of the HSM Model 152 to the standardized NUHOMS® system does not result in any change to the Pre-Operational Tests described in Section 9.2 since the transfer cask involved is not changed and the HSM Model 152 is very similar to the HSM Model 102 from an operations perspective.

Prior to operation of the ISFSI for a particular plant, the licensee should perform functional tests of the in-plant operations, the on-site transfer operations, and DSC insertion and retrieval (operations at the ISFSI). These tests are intended to verify that the storage system components (e.g., DSC, HSM, transfer cask, transfer equipment, etc.) operate safely and effectively. Such a program has been successfully completed for the NUHOMS® ISFSIs at Duke Power Company's Oconee Nuclear Station, Baltimore Gas and Electric Company's Calvert Cliffs Nuclear Power Plant, Toledo Edison's Davis Besse Nuclear Station and Pennsylvania Power and Light's Susquehanna Nuclear Station.

R.9.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility to ensure that the DSC and the HSM conform to the drawings and specifications. The visual inspections include verifying dimensions and the application of specified coatings and that the DSC is clean and free of defects. Visual inspections are performed in accordance with the requirements and acceptance criteria specified by the codes applicable to the associated components.

Upon arrival at the site, the DSCs and HSMs are again inspected to ensure that they have not been damaged during shipment. Conditions which are not in conformance with the drawings and specifications will be repaired or evaluated, in accordance with 10CFR 72.48, for the effect of the condition on the safety function of the components.

R.9.1.2 Structural Tests

No change *to Section 9.2* associated with the addition of the HSM Model 152.

R.9.1.3 Leak Tests and Pressure Tests

No change *to Section 9.2* associated with the addition of the HSM Model 152.

R.9.1.4 Component Tests

No change *to Section 9.2* associated with the addition of HSM Model 152.

R.9.1.5 Shielding Integrity Tests

No change *to Section 9.2* associated with the addition of the HSM Model 152.

R.9.1.6 Thermal Acceptance Tests

No change associated with the addition of the HSM Model 152. AMD 11

R.9.1.7 Neutron Absorber Tests

No change associated with the addition of the HSM Model 152.

R.9.2 Maintenance Program

The NUHOMS® HSM Model 152 system is designed to be totally passive and require minimal maintenance. The DSC does not require any maintenance once it is loaded into the HSM Model 152.

R.9.3 Training Program

No change *to Section 9.3.*

LIST OF TABLES

T.1.2.2 Operational Features

T.1.2.2.1 General Features

The NUHOMS®-61BTH DSC is designed to safely store 61 intact BWR fuel assemblies or up to 16 damaged and remaining intact fuel assemblies. The NUHOMS®-61BTH DSC is designed to maintain the fuel cladding temperature below allowable limits during normal storage, short-term accident conditions, short-term off-normal conditions and fuel loading/transfer operations.

The criticality control features of the NUHOMS®-61BTH 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.

T.1.2.2.2 Sequence of Operations

The sequence of operations to be performed in loading fuel into the NUHOMS®-61BTH DSCs is presented in Chapter T.8.

T.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

T.1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry and by utilizing fixed neutron poison material in the fuel basket. 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®-61BTH DSC and the assurance that no water can enter the DSC cavity during storage.

T.1.2.2.3.2 Chemical Safety

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

T.1.2.2.3.3 Operation Shutdown Modes

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

T.1.2.2.3.4 Instrumentation

No change *to Section 5.1.3.4.*

T.1.2.2.3.5 Maintenance Techniques

No change *to Section 5.1.3.5.*

T.1.4 Generic Cask Arrays

No change *to Section 1.2.1.*

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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2	REVISED PER FCNs 721004-1005, -1063, -1208, -1087 AND -1228	01/08/14
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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

NUHOMS® 61BTH DSC TYPE 1
MAIN ASSEMBLY

DRAWING NO. NUH61BTH-1000-SAR | SCALE NONE | SHEET 1 OF 5

DRAWING NO. NUH61BTH-1000-SAR SHEET 1 OF 5

NUH-61BTH-1000-SAR 2 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-1000-SAR	2 OF 5	2

DRAWING NO. NUH-61BTH-1000-SAR SHEET 3 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-1000-SAR	3 OF 5	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-61BTH-1000-SAR SHEET 4 OF 5

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-1000-SAR	5 OF 5	2

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1	REVISED PER FCNs 721004-665, 721004-797 AND 721004-908 REV 1	01/26/12
0	INITIAL ISSUE PER FCN 721004-557 (SHOWN BY CLOUDS), 721004-655 (SHOWN BY CLOUDS), 721004-756 (FCN 721004-756 INCORPORATES AND 10)	01/28/10
REVISION	DESCRIPTION	DATE

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

NUHOMS® 61BTH DSC TYPE 2 MAIN ASSEMBLY

DRAWING NO. NUH61BTH-2000-SAR | SCALE NONE | SHEET 1 OF 4

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2000-SAR	2 OF 4	2

NUH-61BTH-2000-SAR 3 OF 4

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-61BTH-2000-SAR SHEET 3 OF 4 REVISION 2

DRAWING NO.	SHEET
NUH-61BTH-2000-SAR	4 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2000-SAR	4 OF 4	2

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0	INITIAL ISSUE PER FCN 721004-557 (SHOWN BY CLOUDS) AND 721004-756 (FCN 721004-756 INCORPORATES AMD 10)	01/28/10

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

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SAFETY ANALYSIS REPORT

NUHOMS® 61BTH DSC TYPE 2 SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH61BTH-2001-SAR	NONE	1 OF 2

NUH-61BTH-2001-SAR 2 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2001-SAR	2 OF 2	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
2	REVISED PER FCN 721004-1063	01/08/14
1	REVISED PER FCNs 721004-665, 721004-797 AND 721004-1036	01/26/12
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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
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SAFETY ANALYSIS REPORT

NUHOMS® 61BTH DSC TYPE 2
BASKET ASSEMBLY

DRAWING NO. NUH61BTH-2002-SAR | SCALE NONE | SHEET 1 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2002-SAR	2 OF 6	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-61BTH-2002-SAR SHEET 3 OF 6

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2002-SAR	4 OF 6	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2002-SAR	5 OF 6	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2002-SAR	6 OF 6	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	INITIAL ISSUE PER FCNs 721004-1030 AND 721004-1063	01/08/14
0	INITIAL ISSUE PER FCNs 721004-665 AND 721004-822	01/26/12
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
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SAFETY ANALYSIS REPORT

NUHOMS® 61BTH DSC TYPE 2 TOP GRID ASSEMBLY ALTERNATE 3

DRAWING NO. NUH61BTH-2006-SAR | SCALE NONE | SHEET 1 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-61BTH-2006-SAR	2 OF 2	1

NUH-03-8007-SAR 1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
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ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
GENERAL LICENSE NUHOMS® ISFSI
OS197FC-B ONSITE TRANSFER CASK
MAIN ASSEMBLY

DRAWING NO. NUH-03-8007-SAR | SCALE NONE | SHEET 1 OF 1

DSC are 15, 20, and 120 psig for normal, off-normal and accident conditions, respectively during storage and transfer operations.

T.2.1.1 General Operating Functions

No change *to Section 3.1.2.*

T.2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The NUHOMS®-61BTH DSC is handled and stored in the same manner as the existing NUHOMS®-61BT System. The environmental conditions and natural phenomena are the same as those described in the existing UFSAR Appendix K. Updated criteria are given in the applicable section. Table T.2-14 summarizes the design criteria for the 61BTH DSC. This table also summarizes the applicable codes and standards utilized for design. Design criteria for the Standardized HSM Model 80, 102, 152 and 202 remain the same as shown in Section 3.2.5 of the UFSAR. Design criteria for the HSM-H is the same as described in Appendix P. The OS197FC TC described in the UFSAR, provided with a modified top lid, is designated as OS197FC-B TC. The design criteria for OS197FC-B TC remain the same as shown in Section 3.2.5 of the UFSAR.

T.2.2.1 Tornado Wind and Tornado Missiles

No change to Section P.2.2.1 for HSM-H or to Section 3.2.5 for the standardized HSM.

The evaluation of tornado-generated missile loads on the transfer cask summarized in Section 8.2 of the UFSAR remains unchanged.

T.2.2.2 Water Level (Flood) Design

No change *to Section 3.2.2.*

T.2.2.3 Seismic Design

The seismic design criteria for the HSM-H, the 61BTH DSC and the OS197FC-B TC are consistent with the criteria set forth in Section 3.2.3, with the exception that the NRC Regulatory Guide 1.60 (R.G. 1.60) [2.6] response spectra is anchored to a maximum ground acceleration of 0.30g (instead of 0.25g) for the horizontal components and 0.20g (instead of 0.17g) for the vertical component. The results of the frequency analysis of the HSM-H structure (which includes a simplified model of the DSC) yield a lowest frequency of 23.2 Hz in the transverse direction and 28.4 Hz in the longitudinal direction. The lowest vertical frequency exceeds 33 Hz. Thus, based on the R.G. 1.60 response spectra amplifications, the corresponding seismic accelerations used for the design of the HSM-H are 0.37g and 0.33g in the transverse and longitudinal directions, respectively, and 0.20g in the vertical direction. The corresponding accelerations applicable to the DSC are 0.41g and 0.36g in the transverse and longitudinal directions, respectively, and 0.20g in the vertical direction. The seismic analysis of the HSM-H and 61BTH DSC are further discussed in Section T.3.7.

The seismic design criteria for the HSM Model 80, 102, 152 or 202 do not change from that documented in Section 8.2 of the UFSAR for Models 80/102 or the applicable appendix for Models 152/202. Similarly, the seismic design criteria for OS197 TC or OS197H TC remain unchanged from that documented in Section 8.2, except for seismic which is the same as HSM-H.

T.2.2.4 Snow and Ice Loading

No change *to Section 3.2.4.*

T.2.2.5 Combined Load Criteria

The NUHOMS®-61BTH system is subjected to the same types of loads as the existing NUHOMS®-61BT system. The load combination criteria for the OS197FC-B TC for transfer are the same as those shown in the UFSAR Table 3.2-7. Similarly, the load combination criteria for the HSM Model 80, 102, 152, or 202 do not change from that documented in the UFSAR. The criteria applicable to the NUHOMS®-61BTH DSC and HSM-H are discussed in the following subsections.

T.2.2.5.1 NUHOMS®–61BTH DSC Structural Design Criteria

The NUHOMS®-61BTH DSC is designed using the ASME Boiler and Pressure Vessel Code [2.2] criteria given in the existing UFSAR, Appendix K for the 61BT, except as noted in the following sections. A summary of the NUHOMS®-61BTH DSC load combinations is presented in Table T.2-11.

T.2.2.5.1.1 NUHOMS®–61BTH DSC Shell Stress Criteria

The 61BTH DSC is designed utilizing linear elastic and nonlinear elastic-plastic analytical methods. The stress limits for the NUHOMS®-61BTH 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 NB-3225, Appendix F for accident condition loads (Level D). The stress limits for Level B and Level C are taken from ASME, Section III, Subsection NB, Paragraph NB-3223 and 3224. The 61BTH DSC shell stress limits are summarized in Table T.2-12.

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 are used.

The allowable stress intensity value, S_m, as defined by the Code is based on the temperature calculated for each service load condition or a bounding temperature.

T.2.2.5.1.2 NUHOMS®-61BTH DSC Shell Assembly Stability Criteria

Stability of the 61BTH DSC shell assembly is addressed for those load conditions in which the 61BTH DSC is under external hydrostatic pressure (e.g., vacuum drying and external flood load cases) and/or axial compression, (e.g., loading the shell due to the shield plug's deadweight). Stability criteria are from ASME Section III, NB-3133.3 and NB-3133.6.

T.2.3 Safety Protection Systems

T.2.3.1 General

The NUHOMS®-61BTH system is designed to provide storage of spent fuel for at least 40 years. The DSC cavity is inerted and backfilled with helium and the internal pressure is always above atmospheric during the storage period as a precaution against 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 DSC cavity gas cannot escape.

Only those features that are not addressed in the existing UFSAR, 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. The quality category classification for the various NUHOMS® System Components is described in Section 3.4. The quality categories for the 61BTH system are summarized in Table T.2.15. The detailed quality category of components of the NUHOMS®-61BTH DSC and OS197FC-B TC that are "Important to Safety" and "Not Important to Safety" are also shown on the drawings listed in Section T.1.5.

T.2.3.2 Protection By Multiple Confinement Barriers and Systems

The NUHOMS®-61BTH DSC provides a leak tight confinement of the spent fuel. Similar to the existing NUHOMS®-61BT DSC, sealing of the NUHOMS®-61BTH DSC involves leak testing to the criteria of ANSI N14.5 [2.3] after loading and sealing the canister, as described in Chapter T.7.

T.2.3.3 Protection By Equipment and Instrumentation Selection

No change *to Section 3.3.3.*

T.2.3.4 Nuclear Criticality Safety

T.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 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 neutron absorber 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 criteria of the neutron absorber materials is described in Chapter T.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 pool water. The method of criticality control is in accordance with the requirements of 10CFR72.124 [2.5].

The criticality analyses performed for the 61BTH system are described in Chapter T.6.

T.2.3.4.2 Error Contingency Criteria

Provision for error contingency is built into the criterion used in Section T.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.

T.2.3.4.3 Verification Analysis-Benchmarking

The verification analysis benchmarking used in the criticality safety analysis is described in Chapter T.6.

T.2.3.5 Radiological Protection

No change *to Section 3.3.5.*

T.2.3.6 Fire and Explosion Protection

No change *to Section 3.3.6.*

T.2.4 Decommissioning Considerations

No change *to Section 3.5.*

All changes on this page are AMD 11

Table T.2-1
BWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-61BTH DSC

PHYSICAL PARAMETERS:	
Fuel Class	*Intact or damaged 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or reload fuel manufactured by other vendors that are enveloped by the fuel assembly design characteristics listed in Table T.2-2. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.*
Fuel Damage	*Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly will still be able to be handled by normal means and retrievability is assured following normal and off-normal conditions. Missing fuel rods are allowed.*
RECONSTITUTED FUEL ASSEMBLIES:	
• *Maximum Number of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods*	*4*
• *Maximum Number of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly*	*10*
• *Maximum Number of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods*	*61*
Number of Intact Assemblies	*≤61*
Number and Location of Damaged Assemblies	*Up to 16 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for storage in 61BTH DSC.* *Damaged fuel assemblies may only be stored in the 2x2 compartments as shown in Figure T.2-9. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.*
Channels	*Fuel may be stored with or without channels, channel fasteners, or finger springs.*
Maximum Initial Uranium Content	*198 kg/assembly*
Maximum Assembly Weight with Channels	*705 lbs*

Table T.2-1
BWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-61BTH DSC

(Concluded)

THERMAL/RADIOLOGICAL PARAMETERS:	
Allowable Heat Load Zoning Configurations for each Type 1 61BTH DSC	*Per Figure T.2-1 or Figure T.2-2 or Figure T.2-3 or Figure T.2-4.*
Allowable Heat Load Zoning Configurations for each Type 2 61BTH DSC:	*Per Figure T.2-1 or Figure T.2-2 or Figure T.2-3 or Figure T.2-4 or Figure T.2-5 or Figure T.2-6 or Figure T.2-7 or Figure T.2-8.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 1	*Per Table T.2-7 for Zone 3 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 2	*Per Table T.2-6 for Zone 2 fuel, Table T.2-8 for Zone 4 fuel, and Table T.2-9 for Zone 5 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 3	*Per Table T.2-6 for Zone 2 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 4	*Per Table T.2-5 for Zone 1 fuel, Table T.2-6 for Zone 2 fuel, Table T.2-8 for Zone 4 fuel, and Table T.2-9 for Zone 5 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 5	*Per Table T.2-6 for Zone 2 fuel and Table T.2-9 for Zone 5 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 6	*Per Table T.2-5 for Zone 1 fuel, Table T.2-8 for Zone 4 fuel, Table T.2-9 for Zone 5 fuel, and Table T.2-10 for Zone 6 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 7	*Per Table T.2-8 for Zone 4 fuel and Table T.2-9 for Zone 5 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Heat Load Zoning Configuration 8	*Per Table T.2-6 for Zone 2 fuel, Table T.2-7 for Zone 3 fuel, Table T.2-8 for Zone 4 fuel, and Table T.2-9 for Zone 5 fuel.*
Maximum Lattice Average Initial Enrichment	*Per Table T.2-3 or Table T.2-4*
Maximum Pellet Enrichment	*5.0 wt. % U-235*
Maximum Decay Heat Limits for Zones 1, 2, 3, 4, 5 and 6 Fuel	*Per Figure T.2-1 or Figure T.2-2 or Figure T.2-3 or Figure T.2-4 or Figure T.2-5 or Figure T.2-6 or Figure T.2-7 or Figure T.2-8*
Decay Heat per DSC	*≤ 22.0 kW for Type 1 DSC*
	≤ 31.2 kW for Type 2 DSC
Minimum B10 Content in Poison Plates	*Per Table T.2-3 or Table T.2-4*

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Table T.2-2
BWR Fuel Assembly Design Characteristics[(1)] for the NUHOMS®-61BTH DSC

Transnuclear ID	**7x7 49/0**	**8x8 63/1**	**8x8 62/2**	**8x8 60/4**	**8x8 60/1**	**9x9 74/2**	**10x10 92/2**	**7x7 49/0**	**7x7 48/1Z**	**8x8 60/4Z**	**8x8 62/2**	**9x9 79/2**	**SiemensQ FA**	**10x10 91/1**
Initial Design or Reload Fuel Designation	GE1 GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12 GE14	ENC-IIIA	ENC-III[(2)]	ENC Va ENC Vb	FANP 8x8-2	FANP9 9x9-2	9x9	ATRIUM-10
Maximum Length (in) (Unirradiated)	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.51	176.2	176.51	176.51
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum Number of Fuel Rods	49	63	62	60	60	74	92	49	48	60	62	79	72	91

(1) *Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs.*
(2) *Includes ENC-IIIE and ENC-IIIF.*

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Table T.2-3
Maximum Fuel Assembly Lattice Average Initial Enrichment v/s Minimum B10 Requirements for the NUHOMS®-61BTH DSC Poison Plates (Intact Fuel)

61BTH DSC Type	*Basket Type*	*Maximum Lattice Average Enrichment (wt. % U-235)*	*Minimum B10 Areal Density, (grams/cm²)*	
			Borated Aluminum/MMC	*Boral®*
1	A	3.7	0.021	0.025
	B	4.1	0.032	0.038
	C	4.4	0.040	0.048
	D	4.6	0.048	0.058
	E	4.8	0.055	0.066
	F	5.0	0.062	0.075
2	A	3.7	0.022	0.027
	B	4.1	0.032	0.038
	C	4.4	0.042	0.050
	D	4.6	0.048	0.058
	E	4.8	0.055	0.066
	F	5.0	0.062	0.075

All changes on this page are AMD 11

Table T.2-4

Maximum Fuel Assembly Lattice Average Initial Enrichment v/s Minimum B10 Requirements for the NUHOMS®-61BTH DSC Poison Plates (Damaged Fuel)

61BTH DSC Type	*Basket Type*	*Maximum Lattice Average Enrichment (wt. % U-235)*		*Minimum B10 Areal Density, (grams/cm²)*	
		Up to 4 Damaged Assemblies⁽¹⁾	*Five or More Damaged Assemblies⁽¹⁾ (16 Maximum)*	*Borated Aluminum/MMC*	*Boral®*
1	*A*	*3.7*	*2.80*	*0.021*	*0.025*
	B	*4.1*	*3.10*	*0.032*	*0.038*
	C	*4.4*	*3.20*	*0.040*	*0.048*
	D	*4.6*	*3.40*	*0.048*	*0.058*
	E	*4.8*	*3.50*	*0.055*	*0.066*
	F	*5.0*	*3.60*	*0.062*	*0.075*
2	*A*	*3.7*	*2.80*	*0.022*	*0.027*
	B	*4.1*	*3.10*	*0.032*	*0.038*
	C	*4.4*	*3.20*	*0.042*	*0.050*
	D	*4.6*	*3.40*	*0.048*	*0.058*
	E	*4.8*	*3.50*	*0.055*	*0.066*
	F	*5.0*	*3.60*	*0.062*	*0.075*

Note 1: See Figure T.2-9 for the location of damaged fuel assemblies within the 61BTH DSC.

Table T.2-5

BWR Fuel Qualification Table for Zone 1 Fuel with 0.22 kW per Assembly for the NUHOMS®-61BTH DSC

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

BU GWd/ MTU	*Assembly Average Initial Enrichment (wt. % U-235)*																																
	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
20	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
23	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
25	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
28	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
30	10.5	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5
32				11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0
34				14.0	14.0	14.0	14.0	14.0	14.0	14.0	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5
36				16.5	16.0	16.0	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	14.5	14.5	14.5	14.5
38				19.5	19.0	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
39				21.0	21.0	20.5	20.5	20.5	20.5	20.5	20.5	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0
40										22.0	21.5	21.5	21.5	21.5	21.5	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5
41										23.5	23.5	23.0	23.0	23.0	23.0	23.0	22.5	22.5	22.5	22.5	22.5	22.5	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	21.5
42										24.5	24.5	24.5	24.5	24.5	24.5	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5
43										26.0	26.0	26.0	26.0	26.0	25.5	25.5	25.5	25.5	25.5	25.5	25.5	25.5	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.5
44										27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	26.5	26.5	26.5	26.5	26.5	26.0	26.5	26.0	26.0	26.0
45										29.0	29.0	29.0	29.0	29.0	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.0	28.0	28.0	28.0	28.0	27.5	27.5	27.5	27.5	27.5	27.5
46										30.5	30.5	30.5	30.5	30.0	30.0	30.0	30.0	30.0	30.0	30.0	30.0	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.0	29.0	29.0
47										31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	30.5	30.5	30.5	30.5	30.5	30.5
48										33.0	33.0	33.0	33.0	33.0	33.0	33.0	33.0	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.0	32.0	32.0	32.0	32.0	32.0	32.0
49										34.5	34.5	34.5	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.0	33.0	33.0	33.0
50										36.0	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.0	35.0	35.0	35.0	35.0	35.0	35.0	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5
51										37.0	37.0	37.0	37.0	37.0	37.0	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0
52										38.5	38.0	38.0	38.0	38.0	38.0	38.0	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.0
53										39.5	39.5	39.5	39.5	39.5	39.5	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.5	39.0	38.5	38.5	38.5
54										41.0	41.0	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	39.5
55										41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.0	41.0	41.0	41.0	41.0
56										43.0	43.0	43.0	43.0	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5
57										44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5
58										45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	44.5	44.5	44.5	44.5
59										46.0	46.0	46.0	46.0	46.0	46.5	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.5	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.0	45.5
60										47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0
61										48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.5	48.0	48.0	48.0	48.0
62										49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5

Not Analyzed

If 10 irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional 5.0 years of cooling time.

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

All changes on this page are AMD 11

Table T.2-6

BWR Fuel Qualification Table for Zone 2 Fuel with 0.35 kW per Assembly for the NUHOMS®-61BTH DSC

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

BU GWd/ MTU	*Assembly Average Initial Enrichment (wt. % U-235)*																																
	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
25	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
28	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
30	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
32			6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
34				6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
36				6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
38				7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
39				7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
40										7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
41										8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
42										8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
43										9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0
44										9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5
45										10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0
46										11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5
47										12.0	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0
48										12.5	12.5	12.5	12.5	12.5	12.5	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.0	10.5	10.5	10.5
49										13.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5
50										14.5	14.5	14.5	14.5	14.5	14.5	14.5	14.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0
51										15.5	15.0	15.0	15.0	15.0	15.0	15.0	15.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	13.0	13.0	13.0	13.0	13.0	13.0
52										16.5	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0
53										17.5	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	15.0	15.0	15.0	15.0
54										18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	16.0	16.0	16.0	16.0	16.0	16.0
55										20.5	20.5	19.0	19.0	19.0	19.0	19.0	19.0	19.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.0	17.0	17.0	17.0	17.0	17.0
56										21.5	21.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0
57										22.5	22.5	22.5	21.5	21.5	21.5	21.5	21.5	21.5	21.5	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	19.0	19.0	19.0	19.0	19.0	19.0
58										22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.0	20.0	20.0	20.0
59										23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	21.0	21.0	21.0	21.0
60										24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0
61										26.5	26.5	26.5	25.5	25.5	25.5	25.5	25.5	25.5	25.5	25.5	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	23.0	23.0	23.0	23.0
62										27.5	27.5	27.5	27.5	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.0	24.0

Not Analyzed

If 10 irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional 5.0 years of cooling time.

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

All changes on this page are AMD 11

Table T.2-7

BWR Fuel Qualification Table for Zone 3 Fuel with 0.393 kW per Assembly for the NUHOMS®-61BTH DSC

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																																
	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
28	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
30	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
32				5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
34				5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5
36				6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
38				6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
39				6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
40										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
41										7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0
42										7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
43										7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5
44										8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
45										8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0
46										8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5
47										9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
48										10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5
49										10.5	10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0
50										11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0
51										11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5
52										12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5
53										13.5	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0
54										14.0	14.0	14.0	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5
55										15.0	15.0	14.5	14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0
56										16.0	16.0	15.5	15.5	15.5	15.0	15.0	15.0	15.0	15.0	14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0
57										17.0	16.5	16.5	16.5	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.0	15.0	14.5	14.5	14.5	14.5	14.5	14.5	14.0	14.0	14.0	14.0	14.0
58										18.0	17.5	17.5	17.5	17.5	17.0	17.0	16.5	16.5	16.5	16.5	16.5	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.0	15.0	14.5	14.5
59										19.5	18.5	18.5	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.0	17.0	17.0	17.0	16.5	16.5	16.5	16.0	16.0	16.0	16.0	16.0	16.0	15.5	15.5
60										20.0	19.5	19.5	19.5	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	18.0	17.5	17.5	17.5	17.0	17.0	17.0	17.0	17.0	16.5	16.5	16.5	16.5
61										20.5	20.5	20.5	20.5	20.5	20.0	19.5	19.5	19.5	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.0	18.0	18.0	18.0	17.5	17.5	17.5	17.5
62										21.5	21.5	21.0	21.0	21.0	21.0	20.5	20.5	20.5	20.0	20.0	20.0	19.5	19.5	19.5	19.0	19.0	19.0	19.0	19.0	19.0	18.5	18.5	18.5	18.0

Not Analyzed

If 10 irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional 5.0 years of cooling time.

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

All changes on this page are AMD 11

Table T.2-8

BWR Fuel Qualification Table for Zone 4 Fuel with 0.48 kW per Assembly for the NUHOMS®-61BTH DSC

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

BU GWd/ MTU	*Assembly Average Initial Enrichment (wt. % U-235)*																																
	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
30	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
32				4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5
34				4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
36				5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
38				5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
39				5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
40										5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
41										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
42										6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
43										6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
44										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
45										6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5
46										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
47										7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
48										7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0
49										7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
50										7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5
51										8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
52										8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0
53										9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
54										9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5
55										10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0
56										10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5
57										11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0
58										11.5	11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0
59										12.0	12.0	12.0	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5
60										13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0
61										13.5	13.5	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	10.5
62										14.0	14.0	14.0	14.0	13.5	13.5	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.0	11.0

Not Analyzed

If 10 irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional 5.0 years of cooling time.

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

All changes on this page are AMD 11

Table T.2-9

BWR Fuel Qualification Table for Zone 5 Fuel with 0.54 kW per Assembly for the NUHOMS®-61BTH DSC

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																																
	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
32				4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
34				4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
36				4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
38				4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
39				5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
40										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
41										5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
42										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
43	Not Analyzed									5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5
44										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
45										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
46	*If 10 irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional 5.0 years of cooling time*									6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
47										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0
48										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
49										6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5
50										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5
51										7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
52										7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0
53										7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
54										8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5
55										8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5
56										8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0
57										9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
58										9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5
59										9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
60										10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0
61										10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5
62										11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0

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

All changes on this page are AMD 11

Table T.2-10

BWR Fuel Qualification Table for Zone 6 Fuel with 0.7 kW per Assembly for the NUHOMS®-61BTH DSC

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

BU GWd/ MTU	*Assembly Average Initial Enrichment (wt. % U-235)*																																
	0.9	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0
32				4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
34				4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
36				4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
38				4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
39				4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0
40										4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
41										5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
42										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
43										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
44										5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5
45										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
46										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
47										6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0
48										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
49										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
50										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5
51										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
52										7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0
53										7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
54										7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5
55										8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5
56										8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0
57										8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0
58										9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
59										9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5
60										10.0	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0
61										10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5
62										10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5

Not Analyzed

If 10 irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional 5.0 years of cooling time.

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

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Notes: Tables T.2-5 through Table T.2-10:

- BU = Assembly Average burnup.
- Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with *an* lattice average initial enrichment less than 0.9 (or less than the minimum provided above for each burnup) or greater than 5.0 wt. % U-235 is unacceptable for storage.
- Fuel with a burnup greater than 62 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 3-years cooling.
- See Figure T.2-1 through Figure T.2-8 for a description of the zones.
- For reconstituted fuel assemblies with UO_2 *rods* and/or Zr rods or Zr pellets and/or stainless steel rods, use the lattice average equivalent enrichment to determine the minimum cooling time.
- The cooling times for damaged and intact assemblies are identical.
- *Example*: An *INTACT FUEL ASSEMBLY*, with a decay heat load of 0.22 kW or less, an initial enrichment of 3.65 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a 24 year cooling time as defined by 3.6 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) in Table T.2-5.
- *Using the data specified in Table T.*2-10 results in a maximum decay heat load of 0.55 kW per assembly. Using these data for Zone 6 is conservative since it reduces the heat load from 0.7 kW per assembly to 0.55 kW per assembly.

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Table T.3.1-2
ASME Code Alternatives for the NUHOMS®-61BTH DSC Confinement Boundary
(Part 1 of 2)

Reference ASME Code Section/Article	Code Requirement	*Alternatives*, Justification & Compensatory Measures
NCA	All	Not compliant with NCA. *Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section T.2 may be used for construction, but in no case earlier than 3 years before that specified in Section T.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section T.2 may be used, so as long the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	Requirements for Code Stamping of Components, *Code reports and certificates, etc.*	Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-1132	Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.	Bottom shield plug *and* outer bottom cover plate are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.
NB-2130	Material must be supplied by ASME approved material suppliers.	Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.
NB-4121	Material Certification by Certificate Holder	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT.	The shell to the outer top cover weld, the shell to the inner top cover/weld, the siphon/vent cover welds and the vent and siphon block welds to the shell are all partial penetration welds. As an alternative to the NDE requirements of NB-5230 for Category C welds, all of these closure welds will be multi-layer welds and receive a root and final PT examination, except for the shell to the outer top cover weld. The shell to the outer top cover weld will be a multi-layer weld and receive multi-level PT examination in accordance with the guidance provided in ISG-15 for NDE. The multi-level PT Examination provides reasonable assurance that flaws of interest will be identified. The PT examination is done by qualified personnel, in accordance with Section V and the acceptance standards of Section III, Subsection NB-5000. All of these welds will be designed to meet the guidance provided in ISG-15 for stress reduction factor.

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Table T.3.1-2
ASME Code Alternatives for the NUHOMS®-61BTH DSC Confinement Boundary
(Part 2 of 2)

Reference ASME Code Section/Article	Code Requirement	*Alternatives*, Justification & Compensatory Measures
NB-6100 and 6200	All completed pressure retaining systems shall be pressure tested	The 61BTH is not a complete or "installed" pressure vessel until the top closure is welded following placement of Fuel Assemblies with the DSC. Due to the inaccessibility of the shell and lower end closure welds following fuel loading and top closure welding, as an alternative, the pressure testing of the DSC is performed in two parts. The DSC shell (including all longitudinal and circumferential welds) is pressure tested and examined at the fabrication facility. The shell to the inner top cover closure weld are pressure tested and examined for leakage in accordance with NB-6300 in the field. The siphon/vent cover welds are not pressure tested; these welds and the shell to the inner top cover closure weld are helium leak tested after the pressure test. Per NB-6324 the examination for leakage shall be done at a pressure equal to the greater of the design pressure or three-fourths of the test pressure. As an alternative, if the examination for leakage of these field welds, following the pressure test, is performed using helium leak detection techniques, the examination pressure may be reduced to ≥ 1.5 psig. This is acceptable given the significantly greater sensitivity of the helium leak detection method.
NB-7000	Overpressure Protection	No overpressure protection is provided for the NUHOMS® DSC*s*. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA *d*ata packages are prepared in accordance with the requirements of TN's approved QA program.
NB-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of the Recommended Practice SNT-TC-1A to include up to the most recent 2011 edition.*

Table T.3.1-3
ASME Code Alternatives for the NUHOMS®-61BTH DSC Basket

Reference ASME Code Section/Article	Code Requirement	*Alternatives*, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section T.2 may be used for construction, but in no case earlier than 3 years before that specified in Section T.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section T.2 may be used, so as long the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG/NF-1100	Requirements for Code Stamping of Components, *Code reports and certificates, etc.*	Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME *"N"* or *"NPT"* stamp or *to* be ASME Certified.
NG/*NF*-2000	Use of ASME Material	*Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness.* Material properties in the ASME Code for Type 6061 aluminum are limited to 400°F to preclude the potential for annealing out the hardening properties. Annealed properties (as published by the Aluminum Association and the American Society of Metals) are conservatively assumed for the aluminum transition rails for use above the Code temperature limits.
NG/NF-2130	Materials must be supplied by ASME approved material suppliers	Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NG/NF-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.
NG/NF-4121	Material Certification by Certificate Holder.	
NG-3352	Table NG 3352-1 lists the permissible welded joints and quality factors.	The fuel compartment tubes may be fabricated from sheet with full penetration seam weldments. Per Table NG-3352-1 a joint efficiency (quality) factor of 0.5 is to be used for full penetration weldments examined in accordance with ASME Section V visual examination (VT). A joint efficiency (quality) factor of 1.0 is utilized for the fuel compartment longitudinal seam welds (if present) with VT examination. This is justified because the compartment seam weld is thin and the weldment is made in one pass; and both surfaces of the weldment (inside and outside) receive 100% VT examination. The 0.5 quality factor, applicable to each surface of the weldment, results is a quality factor of 1.0 since both surfaces are 100% examined. In addition, the fuel compartments have no pressure retaining function and the stainless steel material that comprises the fuel compartment tubes is very ductile.
NG/*NF*-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA Data packages are prepared in accordance with the requirements of TN's approved QA program.
NG/NF-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of the Recommended Practice SNT-TC-1A to include up to the most recent 2011 edition.*

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Table T.3.2-1
Summary of the NUHOMS®-61BTH System Component Weights(4)

Component Description	Calculated Weight (kips)
Type 1 DSC Shell Assembly	13.43
Type 1 DSC Top Shield Plug and Top Cover Plates	8.90
Type 1 DSC Internal Basket Assembly	23.37
Type 1 DSC Total Empty Weight	**45.70**
Type 2 DSC Shell Assembly	13.43
Type 2 DSC Top Cover Plates and Shield Plug	8.90
Type 2 DSC Internal Basket Assembly	27.79
Type 2 DSC Total Empty Weight	**50.12**
61 BWR Spent Fuel Assemblies	≤ 43.0
Total Loaded Type 1 DSC Weight (Dry)	**88.70**
Total Loaded Type 2 DSC Weight (Dry)	**93.12**
Water in Loaded Type 1 DSC	13.91
Water in Loaded Type 2 DSC	12.08
Total Loaded Type 1 DSC Weight (Wet)(1)	**93.71**
Total Loaded Type 2 DSC Weight (Wet)(1)	**96.30**
Transfer Cask Empty Weight (with Neutron Shield/Top Lid)	111.25
Total Type 1 Loaded Transfer Cask Weight (Dry/Wet)	**199.95/204.96**
Total Type 2 Loaded Transfer Cask Weight (Dry/Wet)	**204.37/207.55**
HSM Single Module Weight, Model 80/102 (Empty)	263.0
HSM Single Module Weight, Model 152 (Empty)	318.3
HSM-H/202 Single Module Weight (Empty)	306.1
HSM Single Module Weight, Model 80/102 (Loaded)	351.7(2)/356.1(3)
HSM Single Module Weight, Model 152 (Loaded)	407.0(2)/411.4(3)
HSM-H/202 Single Module Weight (Loaded)	394.8(2)/399.2(3)

Notes:
(1) Without top shield plug and top cover plates
(2) Loaded with bounding weight of Type 1 61BTH DSC
(3) Loaded with Type 2 61BTH DSC
(4) Weights are based on nominal dimensions.

K.4. The emissivity of the fuel compartment is 0.2, which is typical for non-polished stainless steel surfaces. If the stainless steel is oxidized, this value would increase, improving heat transfer. The fuel rod emissivity value used is 0.74, which is a typical value for oxidized Zircaloy. Therefore, the passivation reactions would not reduce the thermal properties of the component cask materials or the fuel cladding.

There are no reactions that would cause binding of the mechanical surfaces or the fuel to basket compartment boxes due to galvanic or chemical reactions.

There is no significant degradation of any safety components caused directly by the effects of the reactions or by the effects of the reactions combined with the effects of long term exposure of the materials to neutron or gamma radiation, high temperatures, or other possible conditions.

T.3.4.2 Positive Closure

Positive closure is provided by the OS197/OS197H/OS197FC-B transfer cask. No change *to Section 3.3.2.*

T.3.4.3 Lifting Devices

As described in Section 8.1.19(B), the evaluations for the OS197 and OS197H TC trunnions are based on critical lift weights (with water in the DSC) of 208,500 lbs and 250,000 lbs, respectively. These lifted weights capacities are not changed for the OS197FC-B since the only design features that are different between the OS197/OS197H and the OS197FC-B are the introduction of vent passages around the circumference of the top lid (similar to those in the OS197FC) and the addition of wedge shaped plates at the TC bottom to distribute the incoming air to the TC/DSC annular space. The maximum critical lift weight with a NUHOMS®-61BTH DSC is approximately 204,400 lbs (dry) or 207,600 lbs (wet). Therefore, an OS197FC-B TC that is based on the OS197H design is acceptable with any NUHOMS®-61BTH DSC. An OS197FC-B TC that is based on the OS197 design is limited to a total critical lift weight of 208,500 lbs.

T.3.4.4 Heat and Cold

T.3.4.4.1 Summary of Pressures and Temperatures

Temperatures and pressures for the 61BTH DSC and basket are calculated in Section T.4. Section T.4.4 provides the thermal evaluation of the HSM and HSM-H loaded with a 61BTH DSC. Section T.4.5 provides the thermal evaluation of the OS197/OS197H/OS197FC-B transfer casks loaded with a 61BTH DSC. Section T.4.6 provides the thermal evaluation of the 61BTH DSC. Section T.4.7 provides the thermal evaluation for fuel loading/unloading conditions, including during vacuum drying operations. Tables T.4-12, T.4-17, and T.4-21 summarize the maximum fuel cladding temperatures for normal, off-normal and accident conditions. Tables T.4-13, T.4-14, T.4-18, T.4-19, T.4-22, and T.4-23 summarize the 61BTH DSC maximum component temperatures for normal, off-normal and accident conditions. Tables T.4-16, T.4-20, and T.4-24 summarize the maximum DSC cavity pressures for normal, off-normal and accident conditions. Tables T.4-25, T.4-26, and T.4-27 summarize fuel cladding and basket component temperatures for vacuum drying conditions.

Proprietary Information Withheld Pursuant to 10 CFR 2.390

T.3.6.1.5 HSM and HSM-H Design Analysis

The HSM and HSM-H are qualified for a maximum DSC weight of 102 kips in Appendix M and 110 kips in Appendix P, respectively. Therefore, the HSM and HSM-H (loaded with the 61BTH Type 1 and the Type 2 DSC) results are bounded by the results presented in Appendix M and Appendix P.

The HSM Models 80/102/152/202 are qualified for a maximum heat load of 24 kW in Section 8.0 (for HSM Models 80/102) or the applicable appendix (for HSM Models 152/202) which bounds the maximum heat load up to 22 kW for the Type 1 DSC. The HSM-H is qualified for maximum heat load up to 40.8 kW which bounds the results of up to 31.2 kW for the 61BTH Type 1 and the Type 2 DSC.

T.3.6.1.6 HSM and HSM-H Door Analyses

HSM Door Analysis

No change *to Section 8.1.1.5.*

HSM-H Standard Door Analysis

To accommodate the length of 61BTH DSC, the concrete thickness of the HSM-H door is reduced by 4". The evaluation presented in Appendix P does not take credit for the concrete portion of the door. Thus, the Appendix P evaluation remains unchanged.

HSM-H Optional Door Analysis

The optional shield door for the HSM-H consists of 3" square or round thick steel plate attached to the front wall concrete by four 1" bolts for square plate or four clamps for round plate. At the rear of the 3" thick steel plate, a stepped circular reinforced concrete block is provided.

The optional door is conservatively evaluated for a bounding pressure of 10 psi to bound normal condition loading. Due to this pressure, the maximum moment and shear in the door are equal to 29.2 kip-in/ft and 2.1 kips/ft, respectively. The allowable bending moment and shear forces in the door (without taking credit for the concrete) are 486 kip-in/ft and 518 kips/ft, respectively. Therefore, the door is qualified to meet the design requirements of the code.

T.3.6.1.7 HSM and HSM-H Heat Shield Analysis

No change *to Section 8.1.1.7.*

T.3.6.1.8 HSM Axial Retainer for DSC

The HSM axial retainer is qualified for a maximum DSC weight of 102 kips in Appendix M. The maximum DSC weight is 93 kips for the 61BTH Type 2 DSC. Therefore, the axial retainer results presented in Appendix M bounds these results for 61BTH Type 1 and Type 2 DSC.

The weight of the NUHOMS®-61BTH Type 2 DSC is 93,120 lbs compared to the 80,000 lbs used for the NUHOMS®-52B DSC. The minimum margin of safety for the NUHOMS®-52B DSC analysis for this accident has been scaled by a factor of [80,000/93,120 = 0.859] to establish the minimum factor of safety applicable to the NUHOMS®-61BTH DSC. See Section T.3.7.12.3.

T.3.7.5 Loss of Neutron Shield

No change *to Section 8.2.5.3.*

T.3.7.6 Lightning

No change *to Section 8.2.6.*

T.3.7.7 Blockage of Air Inlet and Outlet Openings

This accident conservatively postulates the complete blockage of the HSM-H ventilation air inlet and outlet openings on the HSM-H side walls.

Since the NUHOMS® HSM-Hs are located outdoors, there is a remote probability that the ventilation air inlet and outlet openings could become blocked by debris from such unlikely events as floods and tornadoes. The NUHOMS® design features such as the perimeter security fence and the redundant protected location of the air inlet and outlet openings reduces the probability of occurrence of such an accident. Nevertheless, for this conservative generic analysis, such an accident is postulated to occur and is analyzed.

The structural consequences due to the weight of the debris blocking the air inlet and outlet openings are negligible and are bounded by the HSM-H loads induced for a postulated tornado (Section 8.2.2) or earthquake (Section 8.2.3).

The thermal effects for this accident for NUHOMS®-61BTH DSC are described in Section T.4 and T.11. The blocked vent accident condition stress evaluation is described in Section T.3.7.12.5.

T.3.7.8 DSC Leakage

The 61BTH DSC is leak tested to meet the leaktight criteria (1×10^{-7} std. cm^3/sec) of ANSI N14.5 [3.37]. The analysis of the 61BTH demonstrate that the pressure boundary is not breached since its meets the applicable stress limits for normal, off-normal and postulated accident conditions.

T.3.7.9 Accident Pressurization of DSC

The NUHOMS® 61BTH is evaluated and designed for DSC internal pressure which bounds the maximum accident pressure calculated in Chapter T.4. The pressure boundary stresses due to this pressure load are bounded by the results presented in Table T.3.7-16 and Table T.3.7-17. Therefore, the 61BTH DSC is acceptable for this postulated accident condition.

load combination has been updated to reflect the increased deadweight of 93,120 lbs for the NUHOMS®-61BTH DSC. This updated limiting factor of safety is conservatively established as 1.22. Hence, the resulting stresses for the OS197 TC when handling the NUHOMS®-61BTH DSC remain well below the code allowables

T.3.7.12.4 TC Fatigue Evaluation

No change *to Section 8.2.10.3.*

T.3.7.12.5 HSM-H Load Combination Evaluation

The HSM-H evaluations in P.3.7.11.5 are bounding. The evaluated loads for the HSM-H bound those associated with the 61BTH DSC.

T.3.7.12.6 Thermal Cycling of the HSM

No change *to Section 8.2.10.5.*

T.3.7.12.7 DSC Support Structure Load Combination Evaluation

See Section T.3.7.12.5 above.

Table T.3.7-5
Stress Summary of the Type 1 Basket Due to Side Drop Loads – 75g

Drop Orientation	Component[1]	Stress Category	Max. Stress (ksi)[4]	Allowable Stress (ksi)[2]	Reference Figures
45° Side Drop	Basket	P_m	*15.72*	44.31	Figure T.3.7-5
		$P_m + P_b$	*30.43*	56.97	Figure T.3.7-6
	Rails	P_m	*17.70*	44.31	Figure T.3.7-7
		$P_m + P_b$	*29.40*	56.97	Figure T.3.7-8
	Canister	P_m	*2.20*	44.31	Figure T.3.7-9
		$P_m + P_b$	*20.80*	56.97	Figure T.3.7-10
60° Side Drop	Basket	P_m	*15.70*	44.31	Figure T.3.7-11
		$P_m + P_b$	*29.08*	56.97	Figure T.3.7-12
	Rails	P_m	*22.51*	44.31	Figure T.3.7-13
		$P_m + P_b$	*41.00*	56.97	Figure T.3.7-14
	Canister[3]	P_m	*2.67*	44.31	Figure T.3.7-15
		$P_m + P_b$	*21.55*	56.97	Figure T.3.7-16
90° Side Drop	Basket	P_m	*19.42*	44.31	Figure T.3.7-17
		$P_m + P_b$	*24.58*	56.97	Figure T.3.7-18
	Rails	P_m	*30.12*	44.31	Figure T.3.7-19
		$P_m + P_b$	*34.35*	56.97	Figure T.3.7-20
	Canister[3]	P_m	*3.43*	44.31	Figure T.3.7-21
		$P_m + P_b$	*18.19*	56.97	Figure T.3.7-22
161.5° Side Drop Impact on one Transfer cask Support rail	Basket	P_m	*15.31*	44.31	Figure T.3.7-23
		$P_m + P_b$	*28.75*	56.97	Figure T.3.7-24
	Rails	P_m	*21.16*	44.31	Figure T.3.7-25
		$P_m + P_b$	*48.43*	56.97	Figure T.3.7-26
	Canister[3]	P_m	*3.57*	44.31	Figure T.3.7-27
		$P_m + P_b$	*24.87*	56.97	Figure T.3.7-28
180° Side Drop Impact on two Transfer cask Support rails	Basket	P_m	*17.56*	44.31	Figure T.3.7-29
		$P_m + P_b$	*26.24*	56.97	Figure T.3.7-30
	Rails	P_m	*29.22*	44.31	Figure T.3.7-31
		$P_m + P_b$	*37.32*	56.97	Figure T.3.7-32
	Canister[3]	P_m	*5.10*	44.31	Figure T.3.7-33
		$P_m + P_b$	*28.17*	56.97	Figure T.3.7-34

Notes:
1. Reported rails are stainless steel rails only.
2. Based on elastic/plastic analyses and allowable at 750°F.
   $P_m \leq \max(0.7\ S_U,\ S_Y + 1/3\ (S_U - S_Y))$
   $P_m + P_b \leq 0.9\ S_U$
3. Canister stresses excluded pressure.
4. *ANSYS results are side drop stresses resulting from a 76g load.*

- Maximum off-normal ambient temperature of 117°F with insolation, and
- Minimum off-normal ambient temperature of -40°F without insolation.

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The HSM-H thermal model described in Section T.4.4.4 above provides the surface temperatures that are applied to the DSC shell, basket and payload model. The results are presented in Table T.4-28.

T.4.6.7.3 Off-Normal Ambient Temperatures during Transfer

The thermal performance of the NUHOMS®-61BTH DSC during transfer under the minimum ambient temperature of 0°F with no insolation and 117°F with maximum insolation, for Type 1 and Type 2 DSCs are examined.

Note that a solar shield is used for transfer operations when the ambient temperature exceeds 100°F up to 117°F. This is done according to the NUHOMS® CoC 1004, Technical Specification *5.3.1.B*, "TC/DSC Transfer Operations at High Ambient Temperatures" for transfer operations, when ambient temperatures exceed 100°F, which requires that a solar shield shall be used to provide protection against direct solar radiation.

T.4.6.7.4 Boundary Conditions, Off-Normal Transfer

The off-normal conditions of transfer analyses are performed for the following ambient conditions:

- Maximum off-normal ambient temperature of 117°F without insolation.

The 61BTH DSC temperature profiles calculated using the OS197FC-B thermal model as described in Section T.4.5 are applied to the corresponding surfaces of the DSC thermal analysis finite element model described in Section T.4.6.2.

T.4.6.7.5 61BTH DSC Thermal Analyses Results for Off-Normal Conditions of Storage and Transfer

Fuel Cladding Temperatures

The maximum fuel cladding temperatures during off-normal conditions of storage and transfer are evaluated for both Type 1 and Type 2 DSCs and compared with the corresponding fuel cladding temperature limits. The results are reported in Table T.4-17.

DSC Basket Component Temperatures

The maximum temperatures of the DSC components for off-normal conditions of storage and transfer are listed in Table T.4-18 and Table T.4-19 for Type 1 and Type 2 DSCs, respectively.

T.4.6.7.6 Off-Normal 61BTH DSC Maximum Internal Pressure during Storage/Transfer

T.8.1 Procedures for Loading the Cask

T.8.1.1 Preparation of the Transfer Cask and DSC

1. Prior to placement in dry storage, the candidate intact and damaged fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification *2.1.* AMD 11

2. Prior to being placed in service, the transfer cask is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification *5.2.4.d.* AMD 11

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

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

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

6. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed. Record the DSC serial number which is located on the grapple ring. Verify the correct DSC type, basket type, and poison material types against the DSC serial number. Verify that the DSC is appropriate for the specific fuel loading campaign per Technical Specification *2.1.*

   CAUTION: If loading fuel assemblies through the basket hold down ring (HDR) or top grid assembly (TGA), verify that the lifting grapple will be able to release fuel assemblies while inside the HDR/TGA. AMD 11 & 72.48

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

8. If damaged fuel assemblies are to be included in a specific loading campaign, place the required number of bottom end caps provided (up to a maximum of 16) into the bottom of the appropriate 2x2 compartments of the basket, as shown in Figure 1-25 of Technical Specification *2.1. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.* AMD 11 & 72.48

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

10. Fill the DSC cavity with water from the fuel pool or an equivalent source.

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

T.8.1.2 DSC Fuel Loading

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

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

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

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

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

   - A TC/DSC loading plan is developed to verify that the damaged and/or intact fuel assemblies meet the burnup, enrichment and cooling time parameters of Technical Specification *2.1*. AMD 11
   - The loading plan is independently verified and approved before the fuel load.
   - A fuel movement schedule is then written, verified and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance with the fuel movement schedule.
   - If loading damaged fuel assemblies, verify that the required number of bottom end caps are installed in appropriate fuel compartment tube locations before fuel load.

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

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Steps *6 and* 7 for each SFA loaded into the DSC. A maximum of 16 damaged fuel assemblies may be loaded into the appropriate 2x2 compartments of the 61BTH DSC basket per Technical Specification *2.1*. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly in the DSC. If loading damaged fuel assemblies, place top end caps over each damaged fuel assembly placed into the basket. AMD 11 & 72.48

8. a. After all the SFAs have been placed into the DSC and their identities verified, place the hold down ring or optional top grid assembly as applicable. Visually verify that the hold down ring is properly seated. *If using the hold down ring or top grid not integral to the basket, they may be placed on the basket before loading the SFAs.*

b. Position the lifting yoke and the top shield plug and lower the shield plug into the DSC. Note that separate rigging may be used to install the shield plug prior to engaging the trunnions with the lifting yoke.

CAUTION: Verify that all the lifting height restrictions as a function of temperature specified in Technical Specification *5.3.1.A* can be met in the following steps which involve lifting of the transfer cask. AMD 11

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

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

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

12. Inspect the top shield plug to verify that it is properly seated within the DSC. If not, lower the cask and reposition the top shield plug and/or remove the shield plug and reposition the hold down ring. Repeat Steps 8 through 12 as necessary. AMD 11

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

14. Drain any excess water from the top of the DSC shield plug back to the fuel pool. Check the radiation levels at the center of top shield plug and around the perimeter of the cask. Disconnect the top shield plug rigging.

15. Drain a minimum of 50 gallons of water. Optionally up to approximately 1100 gallons of water (as indicated on the flow meter) may be drained from the DSC back into the fuel pool or other suitable location to meet the weight limit on the crane. Use 1-3 psig of helium to backfill the DSC with an inert gas per ISG-22 [8.2] guidance as water is being removed from the DSC.

16. Lift the cask from the fuel pool. As the cask is raised from the pool, continue to spray the cask with water and decon as directed. Provisions shall be made to assure that air will not enter the DSC cavity. One way to achieve this is by replenishing the helium in the DSC cavity during cask movement from the fuel pool to the decon area in case of malfunction of equipment used for cask movement.

17. Move the cask with loaded DSC to the cask decon area.

17A. Replace the water removed from the DSC cavity in Step 15 with water from the fuel pool or an equivalent source.

18. Install cask seismic restraints if required by Technical Specification *4.3.3 Step 7* (required only on plant specific basis). AMD 11

T.8.1.3 DSC Drying and Backfilling

CAUTION: During performance of steps listed in Section T.8.1.3, monitor the TC/DSC annulus water level and replenish as necessary *to maintain cooling.*

1. Check the radiation levels along the perimeter of the cask. The cask exterior surface should be decontaminated as necessary. Temporary shielding may be installed as necessary to minimize personnel exposure.
2. Place scaffolding around the cask so that any point on the surface of the cask is easily accessible to personnel.
3. Disengage the rigging cables from the top shield plug and remove the eyebolts. Disengage the lifting yoke from the trunnions and position it clear of the cask.
4. Decontaminate the exposed surfaces of the DSC shell perimeter and remove the inflatable TC/DSC annulus seal.

4a. In accordance with Technical Specification 5.2.4.a, verify that the neutron shield (NS) is filled before the draining operation in Step 5 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

5. Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately *12* inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with the Technical Specification *5.2.4.d* limits. AMD 11 AMD 11

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

5a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 6 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

6. Drain approximately 1100 gallons of water (as indicated on a flowmeter) from the DSC back into the fuel pool or other suitable location if not drained in *T.8.1.2 Step 15*. Consistent with ISG-22 [8.2] guidance, helium at 1-3 psig is used to backfill the DSC with an inert gas (helium) as water is being removed from the DSC. AMD 11
7. *Not used.*
8. Install the automatic welding machine onto the inner top cover plate and place the inner top cover plate with the automatic welding machine onto the DSC. Optionally, the inner top cover plate and the automatic welding machine can be placed separately. Verify proper fit-up of the inner top cover plate with the DSC shell.
9. Check radiation levels along the surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure. AMD 11
10. Insert approximately ¼ inch tubing of sufficient length and adequate temperature resistance through the vent port such that it terminates just below the DSC top shield plug. Connect the tubing to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner top cover plate, in compliance with Technical Specification *5.2.6*. AMD 11
11. Cover the TC/DSC annulus to prevent debris and weld splatter from entering the annulus.

12. Ready the automatic welding machine and tack weld the inner top cover plate to the DSC shell. Install the inner top cover plate weldment and remove the automatic welding machine.

    CAUTION: Continuously monitor the hydrogen concentration in the DSC cavity using the arrangement or other alternate methods described in step 10 during the inner top cover plate cutting/welding operations. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [8.3 and 8.4]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with 2-3 psig helium via the tubing to reduce the hydrogen concentration safely below the 2.4% limit.

13. Perform dye penetrant weld examination of the inner top cover plate weld in accordance with the Technical Specification *5.2.4.b* requirements. AMD 11

14. If loading a Type 2 61BTH DSC or if using a suction pump rather than blowdown to remove water, skip to step 16; otherwise, place the strongback so that it sits on the inner top cover plate and is oriented such that:

    - The DSC siphon and vent ports are accessible
    - The strongback stud holes line up with the TC lid bolt holes

15. Lubricate the studs and, using a crossing pattern, adjust the strongback studs to snug tight ensuring approximately even pressure on the cover plate.

16. Remove purge lines and connect the VDS to the DSC siphon and vent ports.

17. Install temporary shielding to minimize personnel exposure throughout the subsequent welding operations as required.

17a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 18 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

18. a. If using blowdown method to remove water, engage helium supply (up to 10 psig for Type 1 DSC or 15 psig for Type 2 DSC) and open the valve on the vent port and allow helium to force the water from the DSC cavity through the siphon port.

    b. Alternatively a suction pump may be used to remove water from DSC.

19. Once the water stops flowing from the DSC, close the DSC siphon port and disengage the gas source or turn off the suction pump, as applicable.

19a. Verify that the TC axial surface dose rates are compliant with limits specified in Technical Specification 5.2.4.e. The configuration for determining the TC axial dose rates shall be in accordance with Technical Specification 5.2.4.e. AMD 11

20. Connect the hose from the vent port and the siphon port to the intake of the vacuum pump. Connect a hose from the discharge side of the VDS to the plant's radioactive waste system or spent fuel pool. Connect the VDS to a helium source.

NOTE: Proceed cautiously when evacuating the DSC to avoid freezing consequences.

CAUTION: During the vacuum drying evolution, personnel should be in the area of loading operations, or in nearby low dose areas in order to take proper action in the event of a malfunction.

21. Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the DSC cavity. The cavity pressure should be reduced in steps of approximately 100 mm Hg, 50 mm Hg, 25 mm Hg, 15 mm Hg, 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level (these levels are optional), the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less as specified in Technical Specification *3.1.1*. AMD 11

Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC. AMD 11

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

22. Open the valve to the vent port and allow the helium to flow into the DSC cavity.

23. Pressurize the DSC with helium (up to 0 to 10 psig for Type 1 DSC or 0 to 15 psig for Type 2 DSC).

24. Helium leak test the inner top cover plate weld for a leak rate of 1×10^{-4} atm-cm^3/sec. This test is optional. AMD 11

25. If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.

26. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

27. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored (these levels are optional). When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less in accordance with Technical Specification *3.1.1* limits. AMD 11

Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC. AMD 11

28. If loading a Type 1 61BTH DSC, and if the strongback was not installed in Step 14, install the strongback at this time. Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC between 14.5 to 16.0 psig for 61BTH Type 1 and 18.5 to 20.0 psig for 61BTH Type 2 psig and hold for 10 minutes. Depressurize the DSC cavity by releasing the helium through the VDS to the plant spent fuel pool or radioactive waste system to about 2.5 psig in accordance with Technical Specification *3.1.2.b* limits. AMD 11

    CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

29. Close the valves on the helium source.

30. Remove the strongback, if installed in step 14 or step 28 above, decontaminate as necessary, and store.

T.8.1.4 DSC Sealing Operations

CAUTION: During performance of steps listed in Section T.8.1.4, monitor the cask/DSC annulus water level and replenish as necessary to maintain cooling.

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports. Inject helium into blind space just prior to completing welding and perform a dye penetrant weld examination in accordance with the Technical Specification *5.2.4.b* requirements. *Use of an optional test head is acceptable to perform the helium leak test of the inner top cover plate and vent/siphon port welds in accordance with Technical Specification 5.2.4.c. If an optional test head is not used, proceed to Step 2.* AMD 11 & 72.48

2. Temporary shielding may be installed as necessary to minimize personnel exposure. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Optionally, outer top cover plate may be installed separately from the welding machine. Verify proper fit up of the outer top cover plate with the DSC shell.

3. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

4. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification *5.2.4.c* limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.5]. Alternatively, this can be done with a test head in step 1 of Section T.8.1.4. AMD 11

5. If a leak is found, remove the outer cover plate root pass (if not using test head), the vent and siphon port plugs and repair the inner cover plate welds. Then install the strongback (if used) and repeat procedure steps from T.8.1.3 step 21.

6. Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification *5.2.4.b* requirements.

7. Install and seal weld the prefabricated plug, if applicable, over the outer cover plate test port and perform dye penetrant weld examinations in accordance with Technical Specification *5.2.4.b* requirements.

All changes on this page are AMD 11

8. Remove the automatic welding machine from the DSC.

8a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 9 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.

9. Open the cask drain port valve and drain the water from the cask/DSC annulus.

10. Rig the cask top cover plate and lower the cover plate onto the transfer cask.

11. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

    CAUTION: Monitor the applicable time limits of Technical Specification *3.1.3* until the completion of DSC transfer step 6 of Section T.8.1.6, if loading Type 2 61BTH DSC.

12. Verify that the TC *radial* dose rates *measured at the surface of the Transfer Cask* are compliant with limits specified in Technical Specification *5.2.4.e. The configuration for determining the TC radial surface dose rates shall be in accordance with Technical Specification 5.2.4.e.*

T.8.1.5 Transfer Cask Downending and Transfer to ISFSI

NOTE:

Alternate Procedure for Downending of Transfer Cask: Some plants have limited floor hatch openings above the cask/trailer/skid, which limit crane travel (within the hatch opening) that would be needed in order to downend the TC with the trailer/skid in a stationary position. For these situations, alternate procedures are to be developed on a plant-specific basis, with detailed steps for downending.

1. Re-attach the transfer cask lifting yoke to the crane hook, as necessary. Ready the transfer trailer and cask support skid for service.

2. Move the scaffolding away from the cask as necessary. Engage the lifting yoke and lift the cask over the cask support skid on the transfer trailer.

3. The transfer trailer should be positioned so that the cask support skid is accessible to the crane with the trailer supported on the vertical jacks.

4. Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.

5. Move the crane forward while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.

6. Inspect the positioning of the cask to insure that the cask and trunnion pillow blocks are properly aligned.

7. Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.

8. Inspect the trunnions to insure that they are properly seated onto the skid and install the trunnion tower closure plates if required.

9. Remove the bottom ram access cover plate from the cask if integral ram/trailer is not used. Install the two-piece temporary neutron/gamma shield plug to cover the bottom ram access. Install the ram trunnion support frame on the bottom of the transfer cask. (The temporary shield plug and ram trunnion support frame are not required with the integral ram/trailer.)

All changes on this page are AMD 11

T.8.1.6 DSC Transfer to the HSM

1. Prior to transferring the cask to the ISFSI or prior to positioning the transfer cask at the HSM designated for storage, remove the HSM door using a porta-crane, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.

   CAUTION: The insides of empty modules have the potential for high dose rates due to adjacent loaded modules. Proper ALARA practices should be followed for operations inside these modules and in the areas outside these modules whenever the door from the empty HSM has been removed.

2. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.

   CAUTION: Verify that the requirements of Technical Specification *5.3.1.B*, "TC/DSC Transfer Operations at High Ambient Temperatures," are met prior to the next step.

3. Using a suitable vehicle, transfer the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

4. Once at the ISFSI, position the transfer trailer to within a few feet of the HSM.

5. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

6. Using a crane, unbolt and remove the cask top cover plate.

CAUTION: Verify that the applicable time limits of Technical Specification *3.1.3* are met if loading Type 2 61BTH DSC.

All changes on this page are AMD 11

7. Back the transfer trailer to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer. Extend the transfer trailer vertical jacks.

8. Remove the skid tie-down bolts and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.

9. Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.

10. Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.

11. After the cask is docked with the HSM, verify the alignment of the transfer cask using the optical survey equipment.

12. Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug if installed. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

14. Recheck all alignment marks in accordance with the Technical Specification *5.3.3* limits and ready all systems for DSC transfer.

15. Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening. Insert the DSC axial retainer.

19. Install the HSM door using a portable crane and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Technical Specifications *5.4.1 and 5.4.2*.

20. Replace the transfer cask top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

21. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

22. Close and lock the ISFSI access gate and activate the ISFSI security measures.

23. Ensure the *HSM or* HSM-H maximum air exit temperature requirements of Technical Specification *3.1.4* are met. AMD 11 & 72.48

T.8.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform one of the two alternate daily surveillance activities listed below:

   a. A daily visual surveillance of the HSM air inlets and outlets to insure that no debris is obstructing the HSM vents in accordance with Technical Specification *5.2.5.a* requirements. AMD 11

   b. A temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification *5.2.5.b* requirements. AMD 11

T.8.3 Identification of Subjects for Safety Analysis

No change *to Section 5.1.3.*

T.8.4 Fuel Handling Systems

No change *to Section 5.2.*

T.8.5 Other Operating Systems

No change *to Section 5.3.*

T.8.6 Operation Support System

No change *to Section 5.4.*

T.8.7 Control Room and/or Control Areas

No change *to Section 5.5.*

T.8.8 Analytical Sampling

No change *to Section 5.6.*

All changes on this page are AMD 11

T.9.1.3 Leak Tests

DSC confinement welds in the DSC shell and bottom are leak tested at the fabricator's shop to an acceptance criterion of 1×10^{-7} ref cm^3/s, i.e., "leaktight" as defined in ANSI N14.5 [9.4]. Personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.2].

The weld between the DSC shell and inner top cover and *the* siphon/vent cover welds are also leak tested to an acceptance criteria of 1×10^{-7} ref cm^3/s at the field after the fuel assemblies are loaded in the canister.

T.9.1.4 Components

The *Standardized* NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. The gaskets in the Transfer Cask do not require acceptance testing other than the leak testing cited above. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

T.9.1.5 Shielding Integrity

The Transfer Cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6" × 6" grid, the detector will encompass a 6" × 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

T.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section T.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section T.9.1.7.*6*.

T.9.1.7 Poison Acceptance

> ***CAUTION***
>
> *Sections T.9.1.7.1 through T.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.*

All changes on this page are AMD 11

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated *a*luminum

(b) Boron carbide / *a*luminum metal matrix composite (MMC)

(c) *BORAL®*

The 61BTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these three types of materials is given in Table T.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to *BORAL®*, which is described later in this section.

T.9.1.7.1 **Borated Aluminum**

See the Caution in Section T.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur)*. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section T.9.1.7.7. The specified acceptance testing assures that

at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

All changes on this page are AMD 11

T.9.1.7.2 **Boron Carbide / Aluminum Metal Matrix Composites (MMC)**

See the Caution in Section T.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 61BTH DSC, MMCs shall pass the qualification testing specified in Section T.9.1.7.*8*, and shall subsequently be subject to the process controls specified in Section T.9.1.7.*9*.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section T.9.1.7.*7*. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

T.9.1.7.3 ***BORAL®***

See the Caution in Section T.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. *Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns.*. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of *BORAL*®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken *from* the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

All changes on this page are AMD 11

T.9.1.7.4 *Visual Inspections of Neutron Absorbers*

See the Caution in Section T.9.1.7.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Material that does not meet the following acceptance criteria shall be reworked, repaired, or scrapped. Blisters shall be treated as non-conforming. Inspection of MMCs with an integral aluminum cladding shall also include verification that the matrix is not exposed through the faces of the aluminum cladding and that solid aluminum is not present at the edges. For BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet.

T.9.1.7.5 *Other Visual Inspections Criteria (non-Technical Specifications)*

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

T.9.1.7.6 Thermal Conductivity Testing

Testing shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Previous testing of borated aluminum and metal matrix composite shows that thermal conductivity increases slightly with temperature. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.

If a thermal conductivity test result is below the specified minimum, *at least four* additional tests *shall* be performed on the material from that lot. If the mean value of those tests, *including the original test,* falls below the specified minimum, the associated lot shall be rejected.

1 *ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"*
2 *ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"*

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the *same primary* boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as specified in Section T.4.3.

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section T.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

All changes on this page are AMD 11

T.9.1.7.7 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

CAUTION

Portions of T.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

a) **Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.**

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to *1.1 inch* diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. ***Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.***

All changes on this page are AMD 11

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be ***no more than 0.75 sq. inch.***

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. ***If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.***

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than ***the statistically derived minimum thickness from T.9.1.7.7 a)*** **or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness.** ***Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.***

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

All changes on this page are AMD 11

T.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

> ***CAUTION***
>
> *Section T.9.1.7.8.3.1, Section T.9.1.7.8.4 and Section T.9.1.7.8.5 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

T.9.1.7.8.1 Applicability and Scope

Metal matrix composites (MMCs) *acceptable for use in the 61BTH DSC are described in Section T.9.1.7.2..*

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section T.9.1.7.*9* so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the *C*ertificate *H*older.

T.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section T.9.1.7.*8*.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section T.9.1.7.*8*.5.

T.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

*T*hermal damage and corrosion (hydrogen generation) testing shall be *performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.*

Thermal damage testing is not required for *unclad* MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below

842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for MMCs *(clad or unclad)* consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

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T.9.1.7.8.3.1 ***Delamination Testing of Clad MMC***

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage.

T.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from *approximately* the two ends and middle of the *qualification* material run shall be subject to:

a) room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture,

b)* Testing to verify more than 98% *of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs.* Testing or examination for interconnected porosity *on the faces and edges of unclad MMC, and on the edges of clad MMC* shall be performed by a means to be approved by the Certificate Holder. *The maximum interconnected porosity is 0.5 volume %,

and for at least one sample,

c) For MMCs with an integral aluminum cladding, thermal durability testing demonstrating that after a minimum 24 hour soak in either pure or borated water, then insertion into a preheated oven at approximately 825°F for a minimum of 24 hours, the

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3 Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

4 Boralyn testing submitted to the NRC under docket 71-1027, 1998.

5 ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

6 *ASTM E290, Standard Methods for Bend Testing of Materials for Ductility*

specimens are free of blisters and delamination and pass the mechanical testing requirements described in test 'a' of this section.

T.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section 9.1.7.7, or by chemical analysis for boron carbide content in the composite.

T.9.1.7.8.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

T.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION
Sections T.9.1.7.9.1 and T.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

T.9.1.7.9.1 **Applicability and Scope**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section T.9.1.7.8 is required, depending on the characteristics of the material that could be affected by the process change.

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[7] ASTM E94, Recommended Practice for Radiographic Testing

[8] ASTM E142, Controlling Quality of Radiographic Testing

[9] ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

All changes on this page are AMD 11

T.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, *reduce corrosion resistance,* reduce the mechanical strength or ductility of the MMC.

T.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section T.9.1.7.*9*.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that *are* established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a *magnesium-alloyed* aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, *and*

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

In no case shall process changes be accepted if they result in a product outside the limits in Sections *9.5.3.1* and *9.5.3.4*.

T.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, *2004* Edition with *2006* Addenda.

9.2 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.3 *Deleted.*

9.4 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.5 "Aluminum Standards and Data, 2003," The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 *Deleted.*

9.8 *Deleted.*

9.9 *Deleted.*

9.10 *Deleted.*

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Table T.9-1
B10 Specification for the NUHOMS® 61BTH Poison Plates

Basket Type	*Specified Minimum B10 Areal Density for Borated Aluminum/MMC for 90% Credit (g/cm^2)*	*Specified Minimum B10 Areal Density for BORAL® for 75% Credit (g/cm^2)*
	Type 1 DSC	
A	*0.021*	*0.025*
B	*0.032*	*0.038*
C	*0.040*	*0.048*
D	*0.048*	*0.058*
E	*0.055*	*0.066*
F	*0.062*	*0.075*
	Type 2 DSC	
A	*0.022*	*0.027*
B	*0.032*	*0.038*
C	*0.042*	*0.050*
D	*0.048*	*0.058*
E	*0.055*	*0.066*
F	*0.062*	*0.075*

Table T.9-2
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Table T.9-3
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T.11.1 Off-Normal Operations

Off-normal operations are design events of the second type (Design Event II) as defined in ANSI/ANS 57.9 [11.1]. Off-normal conditions consist of that set of events that, although not occurring regularly, can be expected to occur with moderate frequency or on the order of once during a calendar year of ISFSI operation.

The off-normal conditions considered for the NUHOMS®-61BTH DSC are off-normal transfer loads, extreme temperatures and a postulated release of radionuclides.

The term "Standardized HSM" used in this Chapter, is applicable to HSM Model 80, Model 102 and Model 152 or Model 202.

T.11.1.1 Off-Normal Transfer Loads

No change *to Section 8.1.2.1.*

T.11.1.1.1 Postulated Cause of Event

Same as Section 8.1.2. The probability of a jammed DSC does not increase with the NUHOMS®-61BTH DSC, since the interfacing design features and dimensions of the transfer cask top end and HSM access opening are not changed. The 61BTH DSC is provided with similar beveled lead-ins as the 61BT/52B. The maximum allowed misalignment of the sliding surfaces has not changed nor have any of the HSM insertion/retrieval procedures.

T.11.1.1.2 Detection of Event

No change *to Section 8.1.2.1.*

T.11.1.1.3 Analysis of Effects and Consequences

A detailed evaluation of this event is presented in Section T.3.6.2 and is summarized below. The NUHOMS®-61BTH DSC has a 0.5 inch shell wall thickness, while the NUHOMS®-24P and NUHOMS®-52B have a 0.62 inch shell. Therefore the stresses in the canister shell are increased. The DSC shell stress due to the 2,690 in-kip moment due to axial sticking of the DSC is S_{mx} = 1.55 ksi. This magnitude of stress is negligible when compared to the allowable membrane stress of 17.2 ksi.

The DSC shell stress due to the 1,400 pound axial load during the binding of the DSC is 15.7 ksi. This stress is well within the ASME Code Service Level C allowable of 21.7 ksi for an off-normal jammed DSC event.

The evaluation of the basket due to normal and off-normal handling and transfer loads is presented in Section T.3.6.1.3.3.

T.11.1.1.4 Corrective Actions

No change *to Section 8.1.2.1.*

T.11.1.2 Extreme Temperatures

No change. The off-normal maximum ambient temperature of 125°F is used in Section 8.1.2.2. For the NUHOMS®-61BTH system, a maximum ambient temperature of 117°F is used. Therefore, the analyses in Section 8.1.2.2 bound TCs and Standardized HSM used in the NUHOMS®-61BTH system.

T.11.1.2.1 Postulated Cause of Event

No change *to Section 8.1.2.2.*

T.11.1.2.2 Detection of Event

No change *to Section 8.1.2.2.*

T.11.1.2.3 Analysis of Effects and Consequences

The thermal evaluation of the NUHOMS®-61BTH system for off-normal conditions is presented in Chapter T.4. The 100°F normal condition with insolation bounds the 117°F case without insolation for the DSC in the TC. Therefore the normal condition maximum temperatures are bounding. The 117°F case with the DSC in the HSM-H is not bounded by the normal conditions and therefore evaluated in Chapter T.4.

The NUHOMS® TC and the Standardized HSM were evaluated for a maximum heat load of 24 kW and maximum off-normal ambient temperature of 125°F. The maximum heat load of the 61BTH Type 1 DSC in the TC or the Standardized HSM is limited to 22 kW. Therefore the evaluations presented in Section 8.1.2.2 (for HSM Models 80/102) or in their respective appendices (for Models 152/202) are bounding for these components.

The structural evaluation of the 61BTH Type 2 DSC in HSM-H off-normal temperature conditions is presented in Section T.3.6.2.2. The structural evaluation of the basket due to off-normal thermal conditions is presented in Section T.3.6.1.3. The structural evaluation of HSM-H and OS197FC-B Transfer Cask for off-normal conditions with 61BTH DSC are presented in Section T.3.6.

T.11.1.2.4 Corrective Actions

Restrictions for onsite handling of the transfer cask with a loaded DSC under extreme temperature conditions are presented in Technical Specifications *5.3.1.A* and *5.3.1.B*. There is no change to this requirement as a result of addition of the NUHOMS®-61BT DSC. AMD 11

T.11.1.3 Off-Normal Releases of Radionuclides

The NUHOMS®-61BTH DSC is designed and tested to the leak tight criteria of ANSI N14.5 [11.2]. Therefore the estimated quantity of radionuclides expected to be released annually to the environment due to normal or off-normal events is zero.

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

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

T.11.2.3.3 Accident Dose Calculations

The increase in the dose rates at the localized impact location following the missile impact accident is expected to be bounded by the dose rates at the HSM-H vents, calculated to be 600 mrem/hour in Table T.5-1, since the structural analysis results demonstrate that there is no full penetration. This represents an increase in the peak roof dose rates by a factor greater than 20 and is conservative.

For the purpose of this calculation, it is conservatively assumed that the affected area is twice the area of impact ~ 1.6 ft^2. The approximate surface areas at the HSM-H front is 140 ft^2, at the HSM-H roof is 200 ft^2 and that at the HSM-H side is 280 ft^2. The impact area, therefore, represents approximately 0.6% to 1.2% of the surface area of the HSM-H. This will result in an increase of not more than 24% in the average dose rates at the front or roof or the side of the HSM-H. This increase does not significantly affect the ISFSI site dose rates and the results from Section T.10.2 (specifically Table T.10-17) can be utilized to determine the exposure. The dose received by a person located 100 meters away from the ISFSI for the assumed 8 hour duration would be less than 5 mrem (8*Model 80 dose rates at 100m, 1.06E-01 mrem/hour) with a 2x10 array of HSMs. The increased dose to an offsite person located 500 meters away for the assumed 8 hour duration would be less than 0.01 mrem (8*Model 80 dose rates at 500m, 5.25E-04 mrem/hour) with a 2x10 array of HSMs.

T.11.2.3.4 Corrective Actions

After excessive high winds or a tornado, the HSMs and TCs would be inspected for damage. Any debris would be removed. Any damage resulting from impact with a missile would be evaluated to determine if the system was still within the licensed design basis.

T.11.2.4 Flood

This event is described in Section 8.2.4.

T.11.2.4.1 Cause of Accident

No change *to Section 8.2.4.1.*

T.11.2.4.2 Accident Analysis

No change to the Standardized HSM analysis presented in UFSAR Section 8.2.4.2 for Models 80/102 and in their respective appendices for Models 152/202. The HSM-H and DSCs are evaluated for flooding in Section T.3.7.3.

T.11.2.5.3 Accident Dose Calculations for Loss of Neutron Shield

The postulated accident condition for the onsite TC assumes that after a drop event, the water in the neutron shield is lost. The loss of neutron shield is modeled using the normal operation models described in Section T.5.4 by replacing the neutron shield with air. Also, damaged fuel is modeled as fuel rubble that falls to the bottom of the cask. The dose rates due to the fuel rubble model are bounded by the results from assuming intact fuel in damaged fuel locations at far distances. The accident condition dose rates from Chapter T.5, are summarized in Table T.11.-2 for the bounding 61BTH DSC Type 1 loaded with design basis fuel.

Table T.11–2 shows the accident condition dose rates at 1,100 and 500 meters from the TC (OS197 or OS197H or OS197FC-B). The source terms utilized for the accident analysis models are based on normal condition models and result in neutron dose rates that are not maximized. In order to obtain bounding neutron dose rates for accident evaluations, the calculated neutron dose rates are scaled conservatively by a ratio of the maximum allowable total peripheral source terms (calculated from the FQT for the outer 36 fuel assemblies to be 2.4 E+10 neutrons/sec) to the total peripheral neutron source terms utilized in the accident calculations documented in Section T.5.4.8 (4.10 E+09 neutrons). The scaling factor for neutron dose rates is calculated to be 5.85. The gamma dose rates remain unchanged for further conservatism. The scaled dose rates are also shown in Table T.11-2. The dose received by a person located 100 meters away from the NUHOMS® 61BTH system installation for an assumed 8 hour duration would be less than 18 mrem with the OS197FC-B. The increased dose to an offsite person located 500 meters away for the assumed 8 hour duration would be less than 0.02 mrem with both the OS197FC-B TC with NUHOMS® 61BTH DSC. These exposures are well within the limits of 10CFR72.106 for an accident condition.

T.11.2.5.4 Corrective Action

No change. See Section 8.2.5.4.

T.11.2.6 Lightning

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

T.11.2.7 Blockage of Air Inlet and Outlet Openings

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

T.11.2.7.1 Cause of Accident

No change *to Section 8.2.7.1.*

T.11.2.7.2 Accident Analysis

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

The thermal evaluation of this event is presented in Chapter T.4 for HSM-H and a 61BTH DSC. The temperatures determined in Chapter T.4 are used in the structural evaluation of this event, which is presented in Sections T.3.7.7 and T.3.4.4.3 for HSM-H and 61BTH DSC.

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

T.11.2.7.3 Accident Dose Calculations

There are no off-site dose consequences as a result of this accident. The only significant dose increase is that related to the recovery operation. Based on the results presented in Chapter T.5, Table T.5-1 and Table T.5-2, the bounding average dose on HSM front or roof is 19.5 mrem/hr and 58.4 mrem/hr for the HSM-H and Standardized HSM, respectively.

It is conservatively estimated that the on-site workers will receive an additional dose of no more than 467 mrem during an estimated eight hour period that may be required for removal of debris from the inlet and outlet vent openings. These exposures are well within the limits of 10CFR72.106 for an accident condition.

T.11.2.7.4 Corrective Action

No change *to Section 8.2.7.4.*

T.11.2.8 DSC Leakage

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

T.11.2.9 Accident Pressurization of DSC

T.11.2.9.1 Cause of Accident

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

LIST OF TABLES

U.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

U.1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry, soluble boron in the spent fuel pool and by utilizing fixed neutron absorber material in the fuel basket. 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®-32PTH1 DSC and the assurance that no water can enter the DSC cavity during storage.

U.1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the NUHOMS®-32PTH1 system.

U.1.2.2.3.3 Operation Shutdown Modes

The NUHOMS®-32PTH1 DSC system is a totally passive system so that consideration of operation shutdown modes is unnecessary.

U.1.2.2.3.4 Instrumentation

No change *to Section 5.1.3.4.*

U.1.2.2.3.5 Maintenance Techniques

No change *to Section 5.1.3.5.*

U.1.2.3 Cask Contents

The NUHOMS®-32PTH1 DSC system is designed to store 32 intact (or up to 16 damaged and remaining intact) PWR fuel assemblies with or without control components. The fuel that may be stored in the NUHOMS®-32PTH1 DSC is presented in Chapter U.2.

Chapter U.3 provides the structural analysis. Chapter U.4 includes the thermal analysis. Chapter U.5 provides the shielding analysis. Chapter U.6 covers the criticality safety of the NUHOMS®-32PTH1 DSC system and its contents, listing material densities, moderator ratios, and geometric configurations.

U.1.4 Generic Cask Arrays

No change for the HSM-H arrays. *See Section 1.2.1.*

The high seismic HSM-HS design requires that three adjacent HSM-H modules be connected via the roof-to-roof and base-to-base connections provided. The ISFSI pad is designed to allow 10 feet of space for sliding and to facilitate retrievability.

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1
TRANSPORTABLE CANISTER
FOR PWR FUEL MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH32PTH1-1001-SAR	NONE	1 OF 3

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1001-SAR	2 OF 3	2

DRAWING NO. NUH32PTH1-1001-SAR SHEET 2 OF 3

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH32PTH1-1001-SAR | SHEET 3 OF 3 | REVISION 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

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SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1
TRANSPORTABLE CANISTER
PWR FUEL SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH32PTH1-1002-SAR	NONE	1 OF 2

NUH32PTH1-1002-SAR 1 OF 2

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SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390**

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1002-SAR	2 OF 2	1

NUH32PTH1-1002-SAR 2 OF 2

8 7 6 5 4 3 2 1

NUH32PTH1-1003-SAR 1 OF 5

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PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

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SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1
TRANSPORTABLE CANISTER
PWR FUEL BASKET ASSEMBLY

DRAWING NO. NUH32PTH1-1003-SAR | SCALE NONE | SHEET 1 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

NUH32PTH1-1003-SAR 4 OF 5

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH32PTH1-1003-SAR | SHEET 4 OF 5 | REVISION 2

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1003-SAR	5 OF 5	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
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SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1
TRANSPORTABLE CANISTER
PWR FUEL TRANSITION RAILS

DRAWING NO.	SCALE	SHEET
NUH32PTH1-1004-SAR	NONE	1 OF 4

NUH32PTH1-1004-SAR 2 OF 4

PROPRIETARY AND
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WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND
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WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1004-SAR	3 OF 4	1

NUH32PTH1-1004-SAR | 4 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1004-SAR	4 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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TRANSNUCLEAR
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SAFETY ANALYSIS REPORT
NUHOMS® 32PTH1
TRANSPORTABLE CANISTER FOR PWR FUEL ALTERNATE TOP CLOSURE

DRAWING NO.	SCALE	SHEET
NUH32PTH1-1005-SAR	NONE	1 OF 5

DRAWING NO. NUH32PTH1-1005-SAR SHEET 1 OF 5

8 7 6 5 4 3 2 1

NUH32PTH1-1005-SAR 2 OF 5

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PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1005-SAR	2 OF 5	1

8 7 6 5 4 3 2 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1005-SAR	3 OF 5	1

NUH32PTH1-1005-SAR 4 OF 5

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-1005-SAR	4 OF 5	1

8 7 6 5 4 3 2 1

DRAWING NO. NUH32PTH1-1005-SAR SHEET 5 OF 5

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PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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DRAWING NO.	SHEET	REVISION
NUH32PTH1-1005-SAR	5 OF 5	1

8 7 6 5 4 3 2 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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

TRANSNUCLEAR
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SAFETY ANALYSIS REPORT
STANDARDIZED NUHOMS® ISFSI
HSM-HS
MAIN ASSEMBLY

DRAWING NO. NUH-03-7003-SAR | SCALE NONE | SHEET 1 OF 10

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH-03-7003-SAR 3 OF 10

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-03-7003-SAR SHEET 3 OF 10 REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7003-SAR	4 OF 10	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7003-SAR	5 OF 10	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7003-SAR	6 OF 10	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7003-SAR	7 OF 10	1

DRAWING NO. NUH-03-7003-SAR SHEET 8 OF 10

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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DRAWING NO. NUH-03-7003-SAR | SHEET 9 OF 10

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7003-SAR	9 OF 10	1

8 7 6 5 4 3 2 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-7003-SAR	10 OF 10	1

NUH-03-7004-SAR | 1 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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SAFETY ANALYSIS REPORT
STANDARDIZED NUHOMS® ISFSI
HSM-H/HSM-HS
DOSE REDUCTION HARDWARE

DRAWING NO.	SCALE	SHEET
NUH-03-7004-SAR	NONE	1 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-03-7004-SAR | SHEET 2 OF 2 | REVISION 0

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SAFETY ANALYSIS REPORT
NUHOMS®- OS200
ONSITE TRANSFER CASK
INNER & OUTER SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH-08-8002-SAR	NONE	1 OF 3

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-08-8002-SAR	2 OF 3	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH-08-8003-SAR 1 OF 3

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
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SAFETY ANALYSIS REPORT
NUHOMS® - OS200
ONSITE TRANSFER CASK
MAIN ASSEMBLY

DRAWING NO. NUH-08-8003-SAR | SCALE NONE | SHEET 1 OF 3

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-08-8003-SAR	2 OF 3	2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

The NUHOMS®-32PTH1 DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with CCs that can be accommodated in the 32PTH1-L, 32PTH1-M, and 32PTH1-S is 1,715 lbs, 1,625 lbs, and 1,665 lbs, respectively.

The maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-32PTH1 DSC per Interim Staff Guidance (ISG) No. 11, Revision 3 [2.5]. In addition, ISG-11 does not permit repeated thermal cycling of the fuel cladding (limited to less than 10 cycles) with cladding temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal storage thermal transients [2.5].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, thermal and confinement. These evaluations are performed in Chapter U.5, U.6, U.4 and U.7 respectively. The fuel assembly classes considered are listed in Table U.2-3. It was determined that the B&W 15x15 is the enveloping fuel design for the shielding source term calculation because of its total assembly weight and highest initial heavy metal loading. For criticality safety, the B&W 15x15 assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analysis for all other fuel assembly classes is performed to determine the maximum enrichment allowed as a function of the soluble boron concentration and fixed poison plate loading. For thermal analysis, the WE 14x14 fuel assembly is limiting for the 32PTH1 DSCs, since it results in the lowest effective fuel thermal conductivity. The confinement analysis is based on B&W 15x15 fuel assembly, since it results in a smaller free volume inside the DSC cavity as compared to a 14x14 fuel assembly.

For calculating the maximum internal pressure in the NUHOMS®-32PTH1 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 [2.1]. A minimum of 100% of the fill gas and 30% of the fission gases within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

The maximum internal pressures used in the structural analysis for the NUHOMS®-32PTH1 DSC are 15, 20, and 140 psig for normal, off-normal and accident conditions, respectively, during storage and transfer operations for the 32PTH1 DSCs.

U.2.1.1 General Operating Functions

No change *to Section 3.1.2.*

U.2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The NUHOMS®-32PTH1 DSC is handled and stored in the same manner as the existing NUHOMS®-24P System. The environmental conditions and natural phenomena are the same as those described in the existing UFSAR, Chapter 3 with a few specific differences which are discussed in detail in this section.

Table U.2-16 summarizes the design criteria for the 32PTH1 DSC and HSM-H. This table also summarizes the applicable codes and standards utilized for design. The design criteria for HSM-H are the same as those described in Chapter P.2

The OS200 TC used for the transfer of 32PTH1 DSC is a modified version of the NUHOMS® OS197 TC described in the UFSAR Section 1.3.2, with an extended total length of 211". The design criteria for the OS200 TC is the same as those described in UFSAR Chapter 3. An alternate option, designed with a modified top lid, is designated as OS200FC TC. The design criteria for OS200/OS200 FC TC are the same as those as described in Section P.3.2.5 of the UFSAR with a few specific differences which are detailed in this section.

U.2.2.1 Tornado Wind and Tornado Missiles

No change to Section P.2.2.1 for HSM-H.

The evaluation of tornado-generated missile loads on the transfer cask summarized in Section 8.2 of the UFSAR remains unchanged.

U.2.2.2 Water Level (Flood) Design

No change *to Section 3.2.2.*

U.2.2.3 Seismic Design

The seismic design criteria for the HSM-H is consistent with the criteria set forth in Section 3.2.3, with the exception that the NRC Regulatory Guide 1.60 (R.G. 1.60) [2.11] response spectra is anchored to a maximum ground acceleration of 0.30g (instead of 0.25g) for the horizontal components and 0.25g (instead of 0.17g) for the vertical component (i.e., the site ZPA is 0.3g horizontal and 0.25g vertical). The seismic analysis of the HSM-H and 32PTH1 DSC are further discussed in Section U.3.7.

An alternate high seismic design option of the HSM-H has also been provided. The seismic criteria for the high seismic HSM-H (HSM-HS) is an "enhanced" NRC Regulatory Guide 1.60 response spectra anchored at 1.0g maximum horizontal acceleration and 1.0g maximum vertical acceleration shown in Figure U.2-4.

U.2.2.4 Snow and Ice Loading

No change *to Section 3.2.4.*

Normal Conditions

Normal Condition Stress Criteria for Steel Elements

As summarized in Table U.2-15, the normal condition stress criteria for the fuel compartment tubes and the transition rails, is based on Subsection NG of the ASME Code, Section III [2.2].

Normal Condition Stress Criteria for Aluminum Transition Rails

The aluminum transition rail bodies perform their function (support of the fuel compartment tubes) by remaining in place. The loads on the rail bodies are primarily bearing from the fuel compartment tubes. "Failure" of the transition rail would require that the rail no longer provide support to the fuel compartment tubes. Since the aluminum rail bodies are constrained between the DSC shell and the fuel support compartment tubes, this cannot occur.

Therefore, for deadweight and handling condition loads, stress in the aluminum bodies will be compared to the allowable bearing stress, equal to S_y, from NG-3227.1(a). Values of S_y are taken from Table U.3.3-5 for annealed 6061 aluminum material at temperature (as described in Section U.3.3, these yield stresses are lower bound values).

Accident Conditions

Accident Condition Stress Criteria for Steel Elements

As summarized in Table U.2-15 the accident condition (Level D) stress criteria for the fuel support structure and the welded steel transition rails is based on Appendix F of the ASME Code, Section III. Criteria are provided for both linear elastic and elastic-plastic stress analyses.

Accident Condition Criteria for Aluminum Transition Rails

For accident condition loading (i.e., the postulated drops), the aluminum transition rail bodies must support the fuel tubes such that stresses and displacements in the fuel compartment tubes are acceptable. Since, the rail bodies are captured between the fuel compartment tube and the DSC shell, large displacements of the rails are prevented. Thus, no additional checks (of the aluminum) are required for accident/drop loading. Qualification of the fuel tubes demonstrates that the rails perform their intended function.

U.2.2.5.2 NUHOMS® HSM-H Structural Design Criteria

There are no changes to the HSM-H structural design criteria presented in Appendix P.2, except for the modified earthquake loads (EQ) as discussed in Section U.2.2.3 above.

U.2.2.5.3 NUHOMS® OS200 TC Structural Design Criteria

There are no changes to the design criteria presented in Table 3.2-1 Section 3 of the UFSAR for the OS197/OS197H TCs, except for the modified earthquake loads (EQ) of 0.3g horizontal and 0.25g vertical ZPA accelerations as discussed in Section U.2.2.3 and the maximum decay heat load of 40.8 kW for the OS200 TC (similar to the OS197FC TC described in Appendix P) *and the edition year of the ASME Code.*

For the high seismic accident scenario with maximum site accelerations of 1.0g horizontal and 1.0g vertical, the 75g accident drop evaluation criteria is considered bounding. *The OS200 TC is designed and fabricated in accordance with the 1998 edition with addenda through 2000 of the ASME Code Section III, Subsection NC and the alternative provisions to the ASME Code as described in Table U.3.1-3.*

All changes on this page are AMD 11

U.2.3 Safety Protection Systems

U.2.3.1 General

The NUHOMS®-32PTH1 DSC is designed to provide storage of spent fuel for at least 40 years. The DSC cavity is inerted and backfilled with helium and the internal pressure is always above atmospheric during the storage period as a precaution against 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 DSC cavity gas cannot escape.

Only those features that are not addressed in the existing UFSAR, 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. The quality category classification for the various NUHOMS®system components is described in Section 3.4. The quality categories for the 32PTH1 system are summarized in Table U.2-19. The detailed quality category of components of the NUHOMS®-32PTH1 DSC and the OS200TC that are "Important to Safety" and "Not Important to Safety" are also shown on the drawings listed in Section U.1.5. For the HSM-HS, the components that are "Important to Safety" are unchanged from those listed for the Horizontal Storage Module in Table 3.3-1 and discussed in Section 3.3, "Safety Protection System."

U.2.3.2 Protection By Multiple Confinement Barriers and Systems

The NUHOMS®-32PTH1 DSC provides a leak tight confinement of the spent fuel. Although similar to the existing NUHOMS®-24P DSC, sealing of the NUHOMS®-32PTH1 DSC involves leak testing to the criteria of ANSI N14.5 [2.4] after loading and sealing the canister, as described in Chapter U.7.

U.2.3.3 Protection By Equipment and Instrumentation Selection

No change *to Section 3.3.3.*

U.2.3.4 Nuclear Criticality Safety

U.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 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, soluble boron in the pool 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 criteria of the plates is described in Chapter U.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 borated pool water. The method of criticality control is in accordance with the requirements of 10CFR72.124 [2.10].

The criticality analyses performed for the 32PTH1 system are described in Chapter U.6.

U.2.3.4.2 Error Contingency Criteria

Provision for error contingency is built into the criterion used in Section U.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.

U.2.3.4.3 Verification Analysis-Benchmarking

The verification analysis benchmarking used in the criticality safety analysis is described in Chapter U.6.

U.2.3.5 Radiological Protection

No change *to Section 3.3.5.*

U.2.3.6 Fire and Explosion Protection

No change *to Section 3.3.6.*

U.2.4 Decommissioning Considerations

No change *to Section 3.5.*

U.2.5 Summary of NUHOMS®-32PTH1 DSC and HSM-H Design Criteria

All changes on this page are AMD 11

U.2.5.1 32PTH1 DSC Design Criteria

The NUHOMS®-32PTH1 DSC is designed to store intact and/or damaged PWR fuel assemblies with or without Control Components with assembly average burnup, initial enrichment and cooling time as described in Table U.2-1 and Table U.2-3. The maximum decay heat load of the stored fuel is limited to 1.5 kW per fuel assembly for Type 1 DSC and 0.98 kW for a Type 2 DSC. The maximum heat load per canister is limited to 40.8 kW for a Type 1 DSC and 31.2 kW for a Type 2 DSC in order to keep the maximum fuel cladding temperature below the limit [2.5] necessary to ensure cladding integrity. The fuel cladding integrity is assured by the NUHOMS®-32PTH1 DSC and basket design which limit fuel cladding temperature and maintains a nonoxidizing environment in the DSC cavity as described in Section U.4.

The NUHOMS®-32PTH1 DSC (shell and closure) is designed and fabricated as a Class 1 component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [2.2], and the alternative provisions to the ASME Code as described in Table U.3.1-1.

The NUHOMS®-32PTH1 DSC is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. A combination of fixed neutron absorbers, soluble boron in the pool and favorable geometry are employed to maintain the upper subcritical limit of 0.9411. The fixed neutron absorbers are in the form of plates made from either borated aluminum alloy or MMC or Boral®. The basket is designed and fabricated in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200 [2.2] and the alternative provisions to the ASME Code as described in Table U.3.1-2.

The NUHOMS®-32PTH1 DSC is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornadoes, lightning and floods. Chapter U.11 describes the NUHOMS®-32PTH1 DSC behavior under these accident conditions.

The NUHOMS®-32PTH1 DSC design, fabrication and testing are covered by Transnuclear's Quality Assurance Program, which conforms to the criteria in Subpart G of 10CFR72.

U.2.5.2 HSM-H Design Criteria

There is no change to the HSM-H design criteria presented in Appendix P, Chapter P.2 except for the modified seismic loads as discussed in Section U.2.2.

U.2.5.3 OS200 TC Design Criteria

Same as the OS197/OS197H/OS197FC TC described in Section 3 and Appendix P of the UFSAR with modified seismic loads as described in U.2.2. *The OS200 TC is designed and fabricated in accordance with the 1998 edition with addenda through 2000 of the ASME Code, section III, Subsection NC and the alternatives to the ASME Code as described in Table U.3.1-3.*

All changes on this page are AMD 11

Table U.2-1
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-32PTH1 DSC

PHYSICAL PARAMETERS:	
Fuel Class	*Intact or damaged unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, WE 14x14 and CE 16x16 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table U.2-3. Reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table U.2-3 is also acceptable. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.*
Fuel Damage	*Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly will still be able to be handled by normal means and retrievability is assured following normal and off-normal conditions.*
Reconstituted Fuel Assemblies: • *Maximum Number of Reconstituted Assemblies per DSC With Irradiated Stainless Steel Rods* • *Maximum Number of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly* • *Maximum Number of Reconstituted Assemblies per DSC with unlimited number of low enriched UO_2 rods, or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods*	 *4* *10* *32*
Control Components (CCs)	• *Up to 32 CCs are authorized for storage in 32PTH1-S, 32PTH1-M and 32PTH1-L DSCs.* • *Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies ((CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources.* • *Design basis thermal and radiological characteristics for the CCs are listed in Table U.2-2.*
Number of Intact Assemblies	*≤32*
Number and Location of Damaged Assemblies	*Up to 16 damaged fuel assemblies with balance intact fuel assemblies, or dummy assemblies are authorized for storage in 32PTH1 DSC.* *Damaged fuel assemblies are to be placed in the center 16 locations as shown in Figures U.2-1 through U.2-3. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.*
Maximum Assembly plus CC Weight	*1715 lbs*

Table U.2-1
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-32PTH1 DSC (Concluded)

THERMAL/RADIOLOGICAL PARAMETERS:	
Allowable Heat Load Zoning Configurations for each 32PTH1 DSC	*Per Figure U.2-1 or Figure U.2-2 or Figure U.2-3.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 1	*Per Table U.2-6 for Zone 1 fuel, Per Table U.2-9 and Table U.2-11 for Zone 5 fuel, and Per Table U.2-10 for Zone 6 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 2	*Per Table U.2-8 for Zone 4 and Zone 3 fuel.*
Burnup, Enrichment, and Minimum Cooling Time for Configuration 3	*Per Table U.2-7 for Zone 2 fuel.*
Maximum Planar Average Initial Fuel Enrichment	*Per Table U.2-4 or Table U.2-5.*
Maximum Decay Heat Limits for Zones 1, 2, 3, 4, 5 and 6 Fuel	*Per Figure U.2-1 or Figure U.2-2 or Figure U.2-3.*
Decay Heat per DSC	≤□ *40.8 kW for 32PTH1-S, 32PTH1-M and 32PTH1-L DSCs (Type 1 Basket).*
	≤ *31.2 kW for 32PTH1-S, 32PTH1-M and 32PTH1-L DSCs (Type 2 Basket).*
Minimum Boron Loading	*Per Table U.2-4 or Table U.2-5.*

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Table U.2-2

Thermal and Radiological Characteristics for Control Components Stored in the NUHOMS®-32PT and NUHOMS®-32PTH1 DSCs

Parameter	*BPRAs, NSAs, CRAs, RCCAs, VSIs, Neutron Sources, and APSRAs*	*TPAs and ORAs*
Maximum Gamma Source (γ/sec/Assembly)	*3.91E+13*	*4.1E+12*
Decay Heat (Watts/Assembly)	*8*	*8*

Note: NSAs and Neutron Sources shall only be stored in the interior compartments of the basket. Interior compartments are those that are completely surrounded by other compartments, including the corners. There are twelve interior compartments in the 32PT and 32PTH1 DSCs.

All changes on this page are AMD 11

Table U.2-3

PWR Fuel Assembly Design Characteristics for the NUHOMS®-32PTH1 DSC

Assembly Class		*B&W 15x15*	*WE 17x17*	*CE 15x15*	*WE 15x15*	*CE 14x14*	*WE 14x14*	*CE 16x16*
Maximum Unirradiated Length (in)[(1)]	*32PTH1-S*	*162.6*	*162.6*	*162.6*	*162.6*	*162.6*	*162.6*	*162.6*
	32PTH1-M	*170.0*	*170.0*	*170.0*	*170.0*	*170.0*	*170.0*	*170.0*
	32PTH1-L	*178.3*	*178.3*	*178.3*	*178.3*	*178.3*	*178.3*	*178.3*
Fissile Material		UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/Assembly[(2)]		*0.49*	*0.49*	*0.49*	*0.49*	*0.49*	*0.49*	*0.49*
Maximum Number of Fuel Rods		*208*	*264*	*216*	*204*	*176*	*179*	*236*
Maximum Number of Guide/ Instrument Tubes		*17*	*25*	*9*	*21*	*5*	*17*	*5*

Notes:

(1) Maximum Assembly + Control Component Length (unirradiated).

(2) The maximum MTU/assembly is based on the shielding analysis. The listed value is higher than the actual.

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Table U.2-4

Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for 32PTH1 DSC (Intact Fuel)

Fuel Assembly Class	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)*					
	Minimum Soluble Boron (ppm)	*Basket Type[1]*				
		1A or 2A	*1B or 2B*	*1C or 2C*	*1D or 2D*	*1E or 2E*
WE 17x17 Assembly Class[4]	*2000*	*3.40*	*3.80*	*3.90*	*4.10*	*4.30*
	2300	*3.70*	*4.00*	*4.20*	*4.40*	*4.70*
	2400	*3.70*	*4.10*	*4.30*	*4.50*	*4.80*
	2500	*3.80*	*4.20*	*4.40*	*4.60*	*4.90*
	2800	*4.00*	*4.50*	*4.70*	*5.00*	*5.00*
	3000	*4.20*	*4.60*	*4.80*	*5.00*	*5.00*
CE 16x16 Assembly Class[5]	*2000*	*3.90*	*4.30*	*4.50*	*4.80*	*5.00*
	2300	*4.10*	*4.60*	*4.80*	*5.00*	*5.00*
	2400	*4.20*	*4.70*	*4.90*	*5.00*	*5.00*
	2500	*4.30*	*4.80*	*5.00*	*5.00*	*5.00*
	2800	*4.60*	*5.00*	*5.00*	*5.00*	*5.00*
	3000	*4.70*	*5.00*	*5.00*	*5.00*	*5.00*
BW 15x15 Assembly Class[5]	*2000*	*3.30*	*3.60*	*3.80*	*4.00*	*4.20*
	2300	*3.50*	*3.90*	*4.10*	*4.30*	*4.60*
	2400	*3.60*	*4.00*	*4.20*	*4.40*	*4.70*
	2500	*3.70*	*4.10*	*4.30*	*4.50*	*4.80*
	2800	*3.90*	*4.30*	*4.50*	*4.80*	*5.00*
	3000	*4.10*	*4.50*	*4.70*	*5.00*	*5.00*
CE 15x15 Assembly Class[5]	*2000*	*3.50*	*3.90*	*4.00*	*4.20*	*4.40*
	2300	*3.80*	*4.10*	*4.30*	*4.60*	*4.80*
	2400	*3.90*	*4.30*	*4.40*	*4.70*	*4.90*
	2500	*3.90*	*4.35*	*4.50*	*4.80*	*5.00*
	2800	*4.20*	*4.60*	*4.80*	*5.00*	*5.00*
	3000	*4.30*	*4.80*	*5.00*	*5.00*	*5.00*

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Table U.2-4
Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for 32PTH1 DSC (Intact Fuel)
(Concluded)

Fuel Assembly Class	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)*					
	Minimum Soluble Boron (ppm)	*Basket Type[1]*				
		1A or 2A	*1B or 2B*	*1C or 2C*	*1D or 2D*	*1E or 2E*
WE 15x15 Assembly Class[5]	2000	3.50	3.80	3.90	4.20	4.40
	2300	3.70	4.10	4.20	4.50	4.80
	2400	3.80	4.20	4.40	4.60	4.90
	2500	3.90	4.30	4.50	4.70	5.00
	2800	4.10	4.50	4.70	5.00	5.00
	3000	4.20	4.70	4.90	5.00	5.00
CE 14x14 Assembly Class[6]	2000	3.90	4.40	4.60	4.90	5.00
	2300	4.20	4.70	5.00	5.00	5.00
	2400	4.30	4.80	5.00	5.00	5.00
	2500	4.40	5.00	5.00	5.00	5.00
	2800	4.60	5.00	5.00	5.00	5.00
	3000	4.80	5.00	5.00	5.00	5.00
WE 14x14 Assembly Class[7]	2000	4.20	4.70	4.90	5.00	5.00
	2300	4.50	5.00	5.00	5.00	5.00
	2400	4.60	5.00	5.00	5.00	5.00
	2500	4.70	5.00	5.00	5.00	5.00
	2800	5.00	5.00	5.00	5.00	5.00
	3000	5.00	5.00	5.00	5.00	5.00

Notes:

(1) The fixed poison loading requirements as a function of Basket Type are specified in Table U.2-17.
(2) Not used.
(3) Not used.
(4) Reduce Maximum Planar Average Initial Enrichment by 0.05 wt. % U-235 for assemblies with CCs that extend into the active fuel region.
(5) Reduce Maximum Planar Average Initial Enrichment by 0.10 wt. % U-235 for assemblies with CCs that extend into the active fuel region.
(6) Reduce Maximum Planar Average Initial Enrichment by 0.25 wt. % U-235 for assemblies with CCs that extend into the active fuel region.
(7) No reduction in Maximum Planar Average Initial Enrichment required for assemblies with CCs that extend into the active fuel region.

All changes on this page are AMD 11

Table U.2-5

Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for 32PTH1 DSC (Damaged Fuel)

Fuel Assembly Class	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)*					
	Minimum Soluble Boron (ppm)	*Basket Type(1)*				
		1A or 2A	*1B or 2B*	*1C or 2C*	*1D or 2D*	*1E or 2E*
WE 17x17 Assembly Class (without CCs)	2000	3.40	3.70	3.80	4.05	4.25
	2300	3.60	3.95	4.10	4.35	4.65
	2400	3.70	4.05	4.20	4.45	4.75
	2500	3.75	4.15	4.30	4.55	4.85
	2800	4.00	4.40	4.60	4.85	5.00
	3000	4.15	4.55	4.75	5.00	5.00
WE 17x17 Assembly Class (with CCs)	2000	3.35	3.65	3.75	4.00	4.20
	2300	3.55	3.90	4.05	4.30	4.55
	2400	3.65	4.00	4.15	4.40	4.70
	2500	3.70	4.10	4.25	4.50	4.75
	2800	3.95	4.35	4.55	4.80	5.00
	3000	4.10	4.50	4.70	5.00	5.00
CE 16x16 Assembly Class (without CCs)	2000	3.65	4.05	4.20	4.50	4.75
	2300	3.90	4.30	4.50	4.80	5.00
	2400	4.00	4.40	4.60	4.90	5.00
	2500	4.05	4.50	4.70	5.00	5.00
	2800	4.30	4.80	5.00	5.00	5.00
	3000	4.50	4.95	5.00	5.00	5.00
CE 16x16 Assembly Class (with CCs)	2000	3.60	3.95	4.10	4.40	4.65
	2300	3.80	4.20	4.40	4.70	4.90
	2400	3.90	4.30	4.50	4.80	5.00
	2500	4.00	4.40	4.60	4.80	5.00
	2800	4.20	4.70	4.90	5.00	5.00
	3000	4.40	4.85	5.00	5.00	5.00

Table U.2-5
Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for 32PTH1 DSC (Damaged Fuel)
(Continued)

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Fuel Assembly Class	*Minimum Soluble Boron (ppm)*	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)* Basket Type[1] 1A or 2A	1B or 2B	1C or 2C	1D or 2D	1E or 2E
BW 15x15 Assembly Class (without CCs)	2000	3.30	3.60	3.75	3.95	4.20
	2300	3.50	3.90	4.05	4.30	4.50
	2400	3.60	4.00	4.15	4.40	4.65
	2500	3.65	4.05	4.20	4.50	4.75
	2800	3.90	4.30	4.50	4.75	5.00
	3000	4.05	4.45	4.65	5.00	5.00
BW 15x15 Assembly Class (with CCs)	2000	3.20	3.50	3.65	3.90	4.10
	2300	3.40	3.80	3.95	4.20	4.40
	2400	3.50	3.90	4.05	4.30	4.55
	2500	3.60	4.00	4.15	4.40	4.65
	2800	3.80	4.20	4.40	4.65	4.90
	3000	3.95	4.40	4.55	4.90	5.00
CE 15x15 Assembly Class (without CCs)	2000	3.35	3.70	3.80	4.05	4.25
	2300	3.60	3.95	4.10	4.30	4.60
	2400	3.65	4.05	4.20	4.45	4.70
	2500	3.75	4.15	4.30	4.55	4.80
	2800	4.00	4.40	4.60	4.85	5.00
	3000	4.15	4.55	4.75	5.00	5.00
CE 15x15 Assembly Class (with CCs)	2000	3.30	3.65	3.80	4.00	4.20
	2300	3.55	3.90	4.05	4.30	4.55
	2400	3.65	4.00	4.15	4.45	4.65
	2500	3.70	4.10	4.25	4.50	4.80
	2800	3.95	4.35	4.55	4.80	5.00
	3000	4.10	4.55	4.70	5.00	5.00
WE 15x15 Assembly Class (without CCs)	2000	3.40	3.75	3.90	4.15	4.30
	2300	3.65	4.00	4.20	4.45	4.70
	2400	3.75	4.10	4.30	4.55	4.80
	2500	3.80	4.20	4.40	4.65	4.90
	2800	4.05	4.45	4.60	4.90	5.00
	3000	4.20	4.60	4.80	5.00	5.00

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Table U.2-5
Maximum Planar Average Initial Enrichment v/s Neutron Poison Requirements for 32PTH1 DSC (Damaged Fuel)
(Concluded)

Fuel Assembly Class	*Maximum Planar Average Initial Enrichment (wt. % U-235) as a Function of Soluble Boron Concentration and Basket Type (Fixed Poison Loading)*					
	Minimum Soluble Boron (ppm)	*Basket Type[1]*				
		1A or 2A	*1B or 2B*	*1C or 2C*	*1D or 2D*	*1E or 2E*
WE 15x15 Assembly Class (with CCs)	*2000*	*3.35*	*3.65*	*3.80*	*4.00*	*4.20*
	2300	*3.55*	*3.90*	*4.10*	*4.35*	*4.60*
	2400	*3.65*	*4.00*	*4.20*	*4.45*	*4.70*
	2500	*3.70*	*4.10*	*4.30*	*4.55*	*4.80*
	2800	*3.95*	*4.35*	*4.50*	*4.80*	*5.00*
	3000	*4.10*	*4.50*	*4.70*	*5.00*	*5.00*
CE 14x14 Assembly Class (without CCs)	*2000*	*3.70*	*4.10*	*4.30*	*4.60*	*4.85*
	2300	*3.95*	*4.40*	*4.60*	*4.95*	*5.00*
	2400	*4.05*	*4.50*	*4.70*	*5.00*	*5.00*
	2500	*4.15*	*4.60*	*4.80*	*5.00*	*5.00*
	2800	*4.40*	*4.90*	*5.00*	*5.00*	*5.00*
	3000	*4.55*	*5.00*	*5.00*	*5.00*	*5.00*
CE 14x14 Assembly Class (with CCs)	*2000*	*3.55*	*3.95*	*4.10*	*4.35*	*4.60*
	2300	*3.80*	*4.20*	*4.40*	*4.70*	*4.90*
	2400	*3.9*	*4.30*	*4.50*	*4.80*	*5.00*
	2500	*4.00*	*4.40*	*4.60*	*4.90*	*5.00*
	2800	*4.20*	*4.65*	*4.90*	*5.00*	*5.00*
	3000	*4.35*	*4.85*	*5.00*	*5.00*	*5.00*
WE 14x14 Assembly Class (without CCs)	*2000*	*3.75*	*4.15*	*4.30*	*4.60*	*4.85*
	2300	*3.95*	*4.45*	*4.65*	*5.00*	*5.00*
	2400	*4.05*	*4.55*	*4.75*	*5.00*	*5.00*
	2500	*4.15*	*4.65*	*4.85*	*5.00*	*5.00*
	2800	*4.40*	*4.90*	*5.00*	*5.00*	*5.00*
	3000	*4.60*	*5.00*	*5.00*	*5.00*	*5.00*
WE 14x14 Assembly Class (with CCs)	*2000*	*3.70*	*4.10*	*4.20*	*4.50*	*4.75*
	2300	*3.90*	*4.40*	*4.60*	*4.90*	*5.00*
	2400	*4.00*	*4.50*	*4.65*	*5.00*	*5.00*
	2500	*4.10*	*4.55*	*4.80*	*5.00*	*5.00*
	2800	*4.30*	*4.80*	*5.00*	*5.00*	*5.00*
	3000	*4.50*	*5.00*	*5.00*	*5.00*	*5.00*

Note:
(1) The fixed poison loading requirements as a function of Basket Type are specified in Table U.2-17.

Table U.2-6

PWR Fuel Qualification Table for Zone 1 Fuel with 0.6 kW per Assembly for the NUHOMS®-32PTH1 DSC (Fuel without CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																																											
	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
25									6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0
28									7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0
30									8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
32									10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0
34									12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5
36									14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5
38									17.5	17.0	17.0	17.0	17.0	17.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	15.0	14.5	14.5	14.5	14.5	14.5	14.5	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	13.5
39									18.5	18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	17.0	17.0	17.0	16.5	16.5	16.5	16.5	16.5	16.5	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.0	15.0	15.0
40									20.0	20.0	20.0	20.0	20.0	20.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	17.5	17.5	17.5	17.5	17.5	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	16.0	16.0
41									22.0	21.5	21.5	21.0	21.0	20.5	20.5	20.0	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0
42																					21.0	21.0	20.5	20.5	20.5	20.5	20.0	20.0	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5
43																					22.5	22.5	22.0	22.0	22.0	21.5	21.5	21.5	21.5	21.5	21.5	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.0
44																					24.0	23.5	23.5	23.5	23.0	23.0	23.0	23.0	23.0	23.0	22.5	22.5	22.5	22.5	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0
45																					25.5	25.0	25.0	25.0	24.5	24.5	24.5	24.5	24.5	24.0	24.0	24.0	24.0	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5
46																					27.0	26.5	26.5	26.0	26.0	26.0	26.0	26.0	26.0	25.5	25.5	25.5	25.5	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.5
47																					28.0	28.0	28.0	27.5	27.5	27.5	27.5	27.0	27.0	27.0	27.0	27.0	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5
48																					29.5	29.5	29.5	29.0	29.0	29.0	28.5	28.5	28.5	28.5	28.5	28.5	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0
49																								30.5	30.5	30.0	30.0	30.0	30.0	30.0	30.0	29.5	29.5	29.5	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0
50																								31.5	31.5	31.5	31.5	31.0	31.0	31.0	31.0	31.0	31.0	31.0	30.5	30.5	30.5	30.5	30.0	30.0	30.0	30.0	30.0	30.0
51																								33.0	33.0	32.5	32.5	32.5	32.5	32.5	32.5	32.0	32.0	32.0	32.0	32.0	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5
52																									34.5	34.0	34.0	34.0	34.0	33.5	33.5	33.5	33.5	33.5	33.5	33.0	33.0	33.0	33.0	33.0	33.0	32.5	32.5	32.5
53																									35.5	35.5	35.5	35.5	35.0	35.0	35.0	35.0	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.0	34.0	34.0	34.0
54																										36.5	36.5	36.5	36.5	36.5	36.5	36.0	36.0	36.0	36.0	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.0	35.0
55																										38.0	38.0	37.5	37.5	37.5	37.5	37.5	37.0	37.0	37.0	37.0	37.0	37.0	37.0	36.5	36.5	36.5	36.5	36.5
56																										39.0	39.0	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.0	38.0	38.0	38.0	38.0	38.0	37.5	37.5
57																												40.0	40.0	40.0	40.0	40.0	40.0	39.5	39.5	39.5	39.5	39.5	39.0	39.0	39.0	39.0	39.0	39.0
58																												41.0	41.0	41.0	41.0	41.0	41.0	41.0	41.0	41.0	40.5	40.5	40.5	40.5	40.5	40.0	40.0	40.0
59																												42.0	42.0	42.0	42.0	42.0	42.0	42.0	42.0	42.0	42.0	42.0	41.5	41.5	41.5	41.5	41.5	41.5
60																												43.5	43.5	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	42.5	42.5	42.5
61																												44.5	44.5	44.5	44.5	44.5	44.5	44.5	44.5	44.0	44.0	44.0	44.0	44.0	44.0	44.0	43.5	43.5
62																												45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time for cooling times less than 10 years.

Note: Table U.2-12 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table U.2-7

PWR Fuel Qualification Table for Zone 2 Fuel with 0.8 kW per Assembly for the NUHOMS®-32PTH1 DSC (Fuel without CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																																											
	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25									4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
28									5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
30									6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
32									6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
34									7.5	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
36									8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
38									9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0
39									10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5
40									10.5	10.5	10.5	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
41									11.5	11.5	11.0	11.0	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5
42																					10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0
43																					11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5
44																					12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5
45																					13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0
46																					14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0
47																					15.0	15.0	15.0	14.5	14.5	14.5	14.5	14.0	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0
48																					16.5	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.0	15.0	15.0	15.0	15.0	14.5	14.5	14.5	14.5	14.5	14.5	14.0	14.0	13.5	13.5	13.5
49																								17.0	17.0	16.5	16.5	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.5	15.0	15.0	15.0	15.0	15.0	14.5	14.5	14.5
50																								18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.0	17.0	17.0	17.0	17.0	16.5	16.5	16.5	16.5	16.5	16.0	15.5	15.5	15.5
51																								19.0	19.0	19.0	19.0	18.5	18.5	18.0	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.0	17.0	17.0	17.0	17.0	17.0
52																									20.0	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.0	19.0	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	17.5
53																									21.5	21.0	21.0	21.0	20.5	20.5	20.5	20.5	20.5	20.0	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.0
54																										22.5	22.5	22.0	22.0	21.5	21.5	21.5	21.5	21.0	21.0	21.0	21.0	21.0	20.5	20.5	20.5	20.5	20.5	20.0
55																										23.5	23.5	23.0	23.0	23.0	23.0	23.0	23.0	22.5	22.5	22.5	22.5	22.0	21.5	21.5	21.5	21.5	21.5	21.0
56																										25.0	25.0	24.5	24.5	24.5	24.0	24.0	24.0	24.0	23.5	23.0	23.0	23.0	23.0	23.0	22.5	22.5	22.5	22.5
57																												25.5	25.5	25.5	25.5	25.0	25.0	25.0	25.0	24.5	24.5	24.5	24.5	24.5	23.5	23.5	23.5	23.5
58																												27.0	27.0	26.0	26.0	26.0	26.0	26.0	25.5	25.5	25.5	25.5	25.5	25.0	25.0	25.0	25.0	25.0
59																												27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.0	27.0	27.0	27.0	26.0	26.0	26.0	26.0	26.0	25.5
60																												29.0	28.5	28.5	28.5	28.5	28.0	28.0	28.0	28.0	28.0	28.0	27.5	27.5	27.5	27.5	27.5	27.0
61																												29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.0	29.0	29.0	29.0	29.0	28.0	28.0	28.0	28.0
62																												31.5	30.5	30.5	30.5	30.5	30.5	30.0	30.0	30.0	30.0	30.0	29.5	29.5	29.5	29.5	29.5	29.5

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time for cooling times less than 10 years.

Note: Table U.2-12 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table U.2-8

PWR Fuel Qualification Table for Zone 3 or Zone 4 Fuel with 1.0 kW per Assembly for the NUHOMS®-32PTH1 DSC (Fuel without CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																																										
	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25									4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28									4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
30									5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
32									5.5	5.5	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0
34									6.0	6.0	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
36									6.5	6.5	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
38									7.5	7.0	7.0	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0
39									7.5	7.5	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
40									8.0	8.0	7.5	7.5	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5
41									8.5	8.5	8.0	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
42																					7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0
43																					7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
44																					8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5
45																					8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0
46																					8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
47																					9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5
48																					10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0
49																								10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5
50																								10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0
51																								11.5	11.5	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5
52																									12.0	12.0	12.0	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0
53																									13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	10.5
54																										13.5	13.5	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5
55																										14.5	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0	12.0
56																										15.5	15.0	15.0	15.0	14.5	14.5	14.5	14.0	14.0	14.0	13.5	14.0	13.5	13.5	13.5	13.5	13.0	13.0	13.0
57																												16.0	15.5	15.5	15.5	15.5	15.0	15.0	15.0	14.5	14.5	14.5	14.5	14.0	14.0	14.0	14.0	13.5
58																												17.0	16.5	16.5	16.5	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.0	15.0	15.0	14.5	14.5	14.5
59																												18.0	17.5	17.5	17.5	17.0	17.0	17.0	16.5	16.5	16.5	16.0	16.0	16.0	16.0	15.5	15.5	15.5
60																												19.0	18.5	18.5	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.0	17.0	17.0	16.5	16.5	16.5	16.5
61																												20.0	19.5	19.5	19.0	19.5	19.0	18.5	18.5	18.5	18.0	18.0	18.5	18.0	17.5	17.5	17.5	17.5
62																												20.5	20.5	20.5	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.0	19.0	19.0	18.5	18.5	18.5	18.5

Not Analyzed

Note: *If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time for cooling times less than 10 years.*

Note: Table U.2-12 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table U.2-9

PWR Fuel Qualification Table for Zone 5 Fuel with 1.3 kW per Assembly for the NUHOMS®-32PTH1 DSC (Fuel without CCs)

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

BU GWd/MTU	Assembly Average Initial Enrichment (wt. % U-235)																																										
	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25									3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28									3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30									4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32									4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
34									4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
36									5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
38									5.5	5.5	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
39									6.0	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
40									6.0	6.0	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
41									6.5	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
42																					5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
43																					5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5
44																					5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
45																					5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
46																					6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0
47																					6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
48																					6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
49																								6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5
50																								7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
51																								7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0
52																									7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
53																									7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5
54																										8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
55																										8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0
56																										8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5
57																												9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
58																												9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0
59																												10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5
60																												10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0
61																												11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5
62																												11.5	11.5	11.5	11.5	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time for cooling times less than 10 years.

Note: Table U.2-12 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table U.2-10

PWR Fuel Qualification Table for Zone 6 Fuel with 1.5 kW per Assembly for the NUHOMS®-32PTH1 DSC (Fuel without CCs)

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

BU	Assembly Average Initial Enrichment (wt. % U-235)																																										
GWd/MTU	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25									3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28									3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30									3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32									4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34									4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36									4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
38									5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
39									5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
40									5.5	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5
41									5.5	5.5	5.5	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
42																					4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
43																					4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
44																					5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0
45																					5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
46																					5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
47																					5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
48																					5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5
49																								5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
50																								5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
51																								6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0
52																									6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
53																									6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
54																										6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5
55																										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
56																										7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0
57																												7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
58																												7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5
59																												7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	7.0
60																												8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0
61																												8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
62																												8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time.

Note: Table U.2-12 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table U.2-11

PWR Fuel Qualification Table for Zone 5 with Damaged Fuel with 1.2 kW per Assembly for the NUHOMS®-32PTH1 DSC (Fuel without CCs)

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

Assembly Average Initial Enrichment (wt. % U-235)

BU GWd/MTU	0.7	0.8	0.9	1	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25									3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
28									4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5
30									4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
32									5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0
34									5.5	5.5	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5
36									6.0	6.0	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0
38									6.5	6.5	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5
39									7.0	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5
40									7.0	7.0	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0
41									7.5	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0
42																				
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BU GWd/MTU	2.7	2.8	2.9	3	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
30	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
32	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
34	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
36	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
38	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5
39	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
40	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0
41	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
42	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
43	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5
44	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
45	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
46	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0
47	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
48	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5
49				8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0
50				8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
51				9.0	9.0	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5
52					9.5	9.5	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
53					9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0
54						10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0
55						11.0	10.5	10.5	10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0
56						11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.0	10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5
57								11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0
58								12.5	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5
59								13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.0	11.0	11.0
60								13.5	13.5	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0
61								14.5	14.0	14.0	13.5	14.0	13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5
62								15.0	15.0	15.0	14.5	14.5	14.5	14.5	14.0	14.0	14.0	13.5	13.5	13.5	13.0	13.0	13.0	13.0

Not Analyzed

Note: If irradiated stainless steel rods are present in the reconstituted fuel assembly, add an additional year of cooling time for cooling times less than 10 years.

Note: Table U.2-12 provides the explanatory notes and limitations regarding the use of this table.

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Table U.2-12
Notes for Tables U.2-6 through U.2-11

- *BU = Assembly Average burnup.*
- *Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *For a fuel assembly with Control Components, for a given enrichment and burnup, increase the cooling time obtained from an FQT by one year.*
- *Fuel with an assembly average initial enrichment less than 0.7 (or less than the minimum provided above for each burnup) and greater than 5.0 wt. % U-235 is unacceptable for storage.*
- *Fuel with a burnup greater than 62 GWd/MTU is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 3-years cooling.*
- *See Figure U.2-1 through Figure U.2-3 for a description of the Heat Load Zones.*
- *For reconstituted fuel assemblies with UO_2 rods and/or Zr rods or Zr pellets and/or stainless steel rods, use the assembly average equivalent enrichment to determine the minimum cooling time.*
- *The cooling times for damaged and intact assemblies are identical.*
- *Example: An INTACT FUEL ASSEMBLY without CCs, with a decay heat load of 1.5 kW or less, an initial enrichment of 3.65 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a 4.0 year cooling time as defined by 3.6 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) in Table U.2-10. If the fuel assembly has CCs, the minimum cooling time is increased by an additional one year, resulting in five year minimum cooling time prior to storage.*

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Table U.2-17
B10 Specification for the NUHOMS®-32PTH1 Poison Plates

NUHOMS®-32PTH1 DSC Basket Type	*Minimum B10 Areal Density, (grams/cm^2)*	
	Borated Aluminum or MMC	*Boral®*
1A or 2A	*0.007*	*0.009*
1B or 2B	*0.015*	*0.019*
1C or 2C	*0.020*	*0.025*
1D or 2D	*0.032*	*N/A*
1E or 2E	*0.050*	*N/A*

Table U.2-18
Maximum Allowable Heat Load for the NUHOMS®-32PTH1 System

System Configuration	32PTH1 DSC Type	32PTH1 Basket Type(1),(2)	HSM Configuration	TC Configuration	Max. Heat Load (kW) per DSC
1	32PTH1-S, 32PTH1-M or 32PTH1-L	1A, 1B, or 1C or 1D or 1E	HSM-H/ HSM-HS	OS200FC	40.8 (HLZC 1, with intact or damaged fuel)
				OS200FC	31.2 (HLZC 2, with damaged fuel)
				OS200	31.2 (HLZC 2, with intact fuel)
				OS200	24 (HLZC 3 with intact or damaged fuel)
2	32PTH1-S, 32PTH1-M or 32PTH1-L	2A, 2B, or 2C or 2D or 2E	HSM-H HSM-HS	OS200FC	31.2 (HLZC 2)
				OS200	24.0 (HLZC 3)

Notes:

(1) Basket Type 1 (1A, 1B, 1C, 1D 1E) has aluminum transition rails in the DSC basket.

(2) Basket Type 2 (2A, 2B, 2C, 2D, 2E) has steel transition rails in the DSC basket.

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	Zone 6	Zone 6	Zone 6	Zone 6	
Zone 6	Zone 5*	Zone 5*	Zone 5*	Zone 5*	Zone 6
Zone 6	Zone 5*	Zone 1*	Zone 1*	Zone 5*	Zone 6
Zone 6	Zone 5*	Zone 1*	Zone 1*	Zone 5*	Zone 6
Zone 6	Zone 5*	Zone 5*	Zone 5*	Zone 5*	Zone 6
	Zone 6	Zone 6	Zone 6	Zone 6	

*denotes location where intact or damaged fuel assembly can be stored.

	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Max. Decay Heat / FA (kW)	0.6	N/A	N/A	N/A	1.3(1)	1.5
Max. Decay Heat / Zone (kW)	2.4	N/A	N/A	N/A	15.6	24.0
Max. Decay Heat / DSC (kW)	40.8(2)					

Notes: (1) 1.2 kW per FA is the maximum decay heat allowed for damaged fuel assemblies.
(2) Adjust payload to maintain 40.8 kW heat load.

Figure U.2-1
Heat Load Zoning Configuration No. 1 for 32PTH1-S, 32PTH1-M and 32PTH1-L DSCs (Type 1 Baskets)

Table U.3.1-1
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Confinement Boundary
(Part 1 of 2)

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Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	All	Not compliant with NCA. *Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section U.2 may be used for construction, but in no case earlier than three years before that specified in Section U.2. Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section U.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	Requirements for Code Stamping of Components, *Code reports and certificates, etc.*	Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-2130	Material must be supplied by ASME approved material suppliers.	Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability & certification are maintained in accordance with TN's NRC approved QA program.
NB-4121	Material Certification by Certificate Holder	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT	The shell to the outer top cover weld, the shell to the inner top cover/shield plug weld (including optional design configurations for the inner top cover as described in the 32PTH1 DSC drawings), the siphon/vent cover welds, and the vent and siphon block welds to the shell are all partial penetration welds. As an alternative to the NDE requirements of NB-5230, for Category C welds, all of these closure welds are multi-layer welds and receive a root and final PT examination, except for the shell to the outer top cover weld. The shell to the outer top cover weld will be a multi-layer weld and receive multi-level PT examination in accordance with the guidance provided in ISG-15 for NDE. The multi-level PT examination provides reasonable assurance that flaws of interest will be identified. The PT examination is done by qualified personnel, in accordance with Section V and the acceptance standards of Section III, Subsection NB-5000. All of these welds are designed to meet the guidance provided in ISG-15 for stress reduction factor.
NB-1132	Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.	*Bottom shield plug and outer bottom cover plate* are outside code jurisdiction; these components together are much larger than required to provide stiffening for the *inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is* subject to root and final PT examinations.

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Table U.3.1-1
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Confinement Boundary
(Part 2 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NB-6100 and 6200	All pressure retaining components and completed systems shall be pressure tested. The preferred method shall be hydrostatic test.	The NUHOMS® 32PTH1 DSC is not a complete vessel until the top closure is welded following placement of fuel assemblies within the DSC. Due to the inaccessibility of the shell and lower end closure welds following fuel loading and top closure welding, as an alternative, the pressure testing of the DSC is performed in two parts. The DSC shell and inner bottom plate/forging (including all longitudinal and circumferential welds), are pressure tested and examined at the fabrication facility. The shell to the inner top cover/shield plug closure weld (including optional design configurations for the inner top cover as described in the 32PTH1 DSC drawings) is pressure tested and examined for leakage in accordance with NB-6300 in the field. The siphon/vent cover welds are not pressure tested; these welds and the shell to the inner top cover/shield plug closure weld (including Optional design configurations for the inner top cover as described in the 32PTH1 DSC drawings) are helium leak tested after the pressure test. Per NB-6324 the examination for leakage shall be done at a pressure equal to the greater of the design pressure or three-fourths of the test pressure. As an alternative, if the examination for leakage of these field welds, following the pressure test, is performed using helium leak detection techniques, the examination pressure may be reduced to ≥1.5 psig. This is acceptable given the significantly greater sensitivity of the helium leak detection method.
NB-7000	Overpressure Protection	No overpressure protection is provided for the NUHOMS® DSC*s*. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.
NB-5520	*NDE Personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of the Recommended Practice SNT-TC-1A to include up to the most recent 2011 edition.*

Table U.3.1-2
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Basket Assembly
(Part 1 of 2)

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Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section U.2 may be used for construction, but in no case earlier than 3 years before that specified in Section U.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section U.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG-1100	Requirements for Code Stamping of Components, *Code reports and certificates, etc.*	Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp or *to* be ASME Certified.
NG-2000	Use of ASME Material	*Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness.* Material properties in the ASME Code for Type 6061 aluminum are limited to 400°F to preclude the potential for annealing out the hardening properties. Annealed properties (as published by the Aluminum Association and the American Society of Metals) are conservatively assumed for the aluminum transition rails for use above the Code temperature limits.
NG/NF-2130	Material must be supplied by ASME approved material suppliers.	Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NG-2130 is not possible. Material traceability & certification are maintained in accordance with TN's NRC approved QA program.
NG/NF-4121	Material Certification by Certificate Holder	
NG-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA Data packages are prepared in accordance with the requirements of TN's approved QA program.
NG-3000/ Section II, Part D, Table 2A	Maximum temperature limit for Type 304 plate material is 800°F.	Not compliant with ASME Section II Part D Table 2A material temperature limit for Type 304 steel for the postulated transfer accident case (117°F, loss of sunshade, loss of neutron shield) and blocked vent accident (117°F, 40 hr). The calculated maximum steady state temperatures for transfer accident case and blocked vent accident case are less than 1000 °F. The only primary stresses in the basket grid are deadweight stresses. The ASME Code allows use of SA240 Type 304 stainless steel to temperatures up to 1000°F, as shown in ASME Code, Section II, Part D, Table 1A. In the temperature range of interest (near 800°F), the S_m values for SA240 Type 304 shown in ASME Code, Section II Part D, Table 2A are identical to the allowable S value for the same material shown in Section II, Part D, Table 1A. The recovery actions following these accident scenarios are as described in the UFSAR.

Table U.3.1-2
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Basket Assembly
(Page 2 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NG-3352	Table NG 3352-1 lists the permissible welded joints.	The fusion welds between the stainless steel insert plates and the stainless fuel compartment tube are not included in Table NG-3352-1. These welds are qualified by testing. The required minimum tested capacity of the welded connection (at each side of the tube) shall be 45 kips (at room temperature). The capacity shall be demonstrated by qualification and production testing. Testing shall be performed using, or corrected to, the lowest tensile strength of material used in the basket assembly or to minimum specified tensile strength. Testing may be performed on individual welds, or on weld patterns representative of one wall of the tube. ASME Code Section IX does not provide tests for qualification of these type of welds. Therefore, these welds are qualified using Section IX to the degree applicable together with the testing described here. The welds will be visually inspected to confirm that they are located over the insert plates, in lieu of the visual acceptance criteria of NG-5260 which are not appropriate for this type of weld.
		A joint efficiency (quality) factor of 1.0 is utilized for the fuel compartment longitudinal seam welds. Table NG-3352-1 permits a joint efficiency (quality) factor of 0.5 to be used for full penetration weld examined by ASME Section V visual examination (VT). For the 32PTH1 DSC, the compartment seam weld is thin and the weld will be made in one pass. Both surfaces of weld (inside and outside) will be fully examined by VT and therefore a factor of 2 x 0.5=1.0, will be used in the analysis. This is justified as both surfaces of the single weld pass/layer will be fully examined, and the stainless steel material that comprises the fuel compartment tubes is very ductile.
NG-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of the Recommended Practice SNT-TC-1A to include up to the most recent 2011 edition.*

Table U.3.1-3

Alternatives to *the* ASME Code to the NUHOMS® OS200 *and OS200FC* Transfer Casks

(Applies to *TC* structural components only; *l*ead shielding, neutron shielding, and neutron shield jacket of the *TC* are not addressed by this table.)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Exception, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of code editions and addenda*	*Code edition and addenda other than those specified in Section U.2 may be used for construction, but in no case earlier than 3 years before that specified in Section U.2. Materials produced and certified in accordance with ASME Section II material specification from code editions and addenda other than those specified in Section U.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable subsection of the Section III edition and addenda used for construction.*
NC-1100	Requirements for code stamping of components	The OS200/*OS200FC* TC is designed and fabricated to the requirements of Subsection NC, to the maximum extent practical. However, the transfer cask does not have a code stamp. Code Stamping is not required by 10 CFR 72 regulation. Therefore, the fabricator is not required to be ASME Certified.
NC-2000	ASME code materials are to be used.	The *TC* bottom ram access cover plate is made of ASTM A240, a non-ASME material. This cover plate is a water tight closure used during fuel loading/unloading operations in the fuel/reactor building only. This is not a pressure boundary component, and its failure does not result in any public safety concerns.
NC-2130	Material must be supplied by ASME approved material suppliers.	Materials designated as ASME on the *UFSAR* Chapter U.1 drawings are obtained by TN approved suppliers with Certified Material Test Reports (CMTRs). Material is certified to meet all ASME code criteria but is not eligible for certification or code stamping, if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NC-2130 is not possible.
NC-4120	Material certification by certificate holder	Material traceability & certification are maintained in accordance with TN's NRC approved QA program.
NC-5254	Category D joints shall be RT or UT examined.	The trunnion-to-shell weld is a Category D joint which does not allow adequate UT or RT examination. This weld is not a pressure boundary but serves as lifting point for the TC. During fabrication, this weld is progressive PT examined and then load- tested to three times the design load. The weld between the ram access penetration forging and bottom end plate is a Category D joint which does not allow meaningful RT or UT examination. This weld is PT examined root and final layers. This is not a pressure boundary component and its failure does not result in any public safety concerns.
NC-6000	All completed pressure retaining systems shall be pressure tested.	With respect to pressure testing requirements, the transfer cask is not a pressure retaining component. Therefore, no pressure testing is required. However, the liquid neutron shield cavity, cask bottom neutron shield cavity, and the bottom cover plate assembly are pressure and leak tested.
NC-7000	Overpressure protection	The *TC* is not a pressure retaining component. Therefore, no overpressure protection is provided for the transfer cask, except that a pressure relief valve is provided for the annular neutron shielding.
NC-8000	Requirements for nameplates, stamping & reports per NCA-8000	The *TC* nameplate provides the information required by 10CFR72. Code stamping is not required for the *TC*. QA data packages are prepared in accordance with the requirements of 10CFR72 and TN's NRC approved QA program.
NC-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of the Recommended Practice SNT-TC-1A to include up to the most recent 2011 edition.*

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- The welding and backfilling process takes 8 hours to complete.

Under these assumptions, the hydrogen concentration in the space between the water and the shield plug is a function of the time water is in the DSC prior to backfilling with helium. The hydrogen concentration is (0.313 ft^3 H_2/hr)*(8 hr) / (105.1 ft^3) = 2.38%. Monitoring of the hydrogen concentration before and during welding operations is performed to ensure that the hydrogen concentration does not exceed 2.4%, which is well below the ignitable limit of 4%. If the hydrogen concentration exceeds 2.4%, welding operations are suspended and the DSC is purged with an inert gas. In an inert atmosphere, hydrogen will not be generated.

Effect of Galvanic Reactions on the Performance of the System

There are no significant reactions that could reduce the overall integrity of the DSC or its contents during storage. The DSC and fuel cladding thermal properties are provided in Chapter U.4. The surface emissivity of the fuel compartment tube is 0.46, which is typical for non-polished stainless steel surfaces. If the stainless steel is oxidized, this value would increase, improving heat transfer. The fuel rod emissivity value used is 0.80, which is a typical value for oxidized Zircaloy. Therefore, the passivation reactions would not reduce the thermal properties of the component cask materials or the fuel cladding.

There are no reactions that would cause binding of the mechanical surfaces or the fuel to basket compartment boxes due to galvanic or chemical reactions.

There is no significant degradation of any safety components caused directly by the effects of the reactions or by the effects of the reactions combined with the effects of long term exposure of the materials to neutron or gamma radiation, high temperatures, or other possible conditions.

If an independent spent fuel storage installation site is located in a costal salt water marine atmosphere, then any load-bearing carbon steel DSC support structure rail components of any associated HSM-H shall be procured with a minimum 0.20 percent copper content for corrosion resistance.

U.3.4.2 Positive Closure

Positive closure is provided by the OS200 TC. No change *to Section 3.3.2.*

U.3.4.3 Lifting Devices

The evaluations for the OS200 TC trunnions presented in Section U.3.6.1.6 are based on a critical lift weight of 250,000 lb. As shown in Table U.3.2-1, the maximum critical lift weight with a NUHOMS® 32PTH1 DSC is approximately 241,300 lbs.

U.3.4.4 Heat and Cold

U.3.4.4.1 Summary of Pressures and Temperatures

Temperatures and pressures for the 32PTH1 DSC and basket are calculated in Chapter U.4. Section U.4.4 provides the thermal evaluation of the HSM-H loaded with a 32PTH1 DSC.

Proprietary Information Withheld Pursuant to 10 CFR 2.390

U.3.7.11.2 DSC Fatigue Evaluation

Although the normal and off-normal internal pressures for the NUHOMS® 32PTH1 DSC are higher relative to the NUHOMS® 24P DSC, the range of pressure fluctuations due to seasonal temperature changes are essentially the same as those evaluated for the NUHOMS® 24P DSC. Similarly, the normal and off-normal temperature fluctuations for the NUHOMS® 32PTH1 DSC due to seasonal fluctuations are essentially the same as those calculated for the NUHOMS® 24P DSC. Therefore, the fatigue evaluation presented in UFSAR Section 8.2.10.2 for the 24P DSC remains applicable to the NUHOMS® 32PTH1 DSC.

U.3.7.11.3 TC Load Combination Evaluation

The load combinations considered for the OS197/OS197H for normal, off-normal, and postulated accident loadings are shown in UFSAR Table 3.2-7. Service Levels A and B allowables are used for all normal operating and off-normal loadings. Service Levels C and D allowables are used for load combinations which include postulated accident loadings. The TC load combinations presented in UFSAR Table 3.2-7 are also applicable to the OS200 TC. For the OS200 TC evaluations, the load combinations A1 through C2 in UFSAR Table 3.2-7 are addressed by the nine load cases for normal and off-normal loads in Section U.3.6.1.5.3. In these evaluations a bounding 2g load applied in the axial, transverse and vertical direction is used to bound the handling/transfer and Level C seismic load combinations in combination with deadweight. The bounding 2g load case is also combined with thermal loads resulting from the 31.2 kW and 40.8 kW thermal distributions from the thermal analysis. Load combinations D1 through D3 in UFSAR Table 3.2-7 address the accident drop load combinations, evaluated in Section U.3.7.4.4. The high seismic criteria of 1.0g maximum horizontal and vertical accelerations is a Level D event and is considered bounded by the 75g accident drop evaluations.

U.3.7.11.4 TC Fatigue Evaluation

No change *to Section 8.2.10.3.*

U.3.7.11.5 HSM–H/HSM-HS Load Combination Evaluations

U.3.7.11.5.1 HSM–H/HSM-HS Concrete Component Evaluation

The required strength, U, for critical sections of concrete is calculated in accordance with the requirements of ANSI 57.9 [3.16] and Chapter 9 of ACI 349 [3.28], including the strength reduction factors defined in ACI 349, Section 9.3.

The concrete design loads are multiplied by load factors and combined to simulate the most adverse load conditions. The load combinations described in Table U.3.7-23 are used to evaluate the concrete components.

U.3.7.11.5.2 HSM-H/HSM-HS Support Structure Evaluation

The required steel strength, S, and required shear strength, S_v for critical sections of steel structure are calculated in accordance with the requirements of AISC Allowable Stress Design (ASD) method [3.21].

U.3.7.11.6.6 Evaluation of the HSM-H/HSM-HS Heat Shields

The top and side heat shields of the HSM-H/HSM-HS loaded with a 32PTH1 DSC consist of flat stainless steel plates (Type 304). These heat shield are evaluated for bounding seismic loads. The maximum stresses in the top and side heat shields are 9.0 ksi and 3.1 ksi, respectively. The allowable stress for Type 304 stainless steel at temperature (400°F) is 15.5 ksi. The studs supporting the side heat shields are also evaluated considering bounding seismic loads. The maximum axial and bending stress in the stud is 1.9 ksi and 106.2 ksi, respectively. For the stud material (ASTM A193, Gr B), the allowable axial and bending stress is 67.9 ksi (corresponding to a KL/r of 71) and 112.3 ksi, respectively. Thus the interaction ratio is 0.97.

U.3.7.11.6.7 Evaluation of the HSM-H/HSM-HS Seismic Retainers

The seismic retainer evaluation for the HM-H is not changed from that described in Appendix P, Section P.3.7.11.6.7.

For the HSM-HS the seismic restraint is similar to that described in [3.2] and consists of two drop-in retainers. Each retainer consists of a capped tube steel embedment located within each rail extension baseplate embedment and a tube retainer assembly that drops into the embedment cavity after the 32PTH1 transfer is complete. The maximum seismically induced load in the retainer is 76 kips. The maximum shear stress in the retainer is 19 ksi and the allowable stress (using ASTM A500, Gr B at 300°F) is 22.7 ksi.

U.3.7.11.6.8 Evaluation of Seismic Ties for HSM-HS

To provide stability for an ISFSI array during a design basis seismic event, the HSM-HS's are tied together in a minimum array size of three HSM-HS's plus shield walls. The HSM-HS's are tied together using two tie beams at the roof and eight 2" diameter tie rods at the base. The top tie beams and tie rods resist module separation due to out of phase tipping and relative sliding between the modules in the transverse and longitudinal directions. The top tie beams are integral with the roof structure. The base tie rods are grouted to resist tipping and relative slipping between the modules. The top tie beams and the bottom tie rodss are also designed to accommodate a 5% accidental torsional load due to seismic excitation.

The roof attachment to the base includes two shear keys located on the underside of the roof. The roof is also attached to the base in the vertical direction by four threaded rods or four stiffened brackets that attach the roof to the front and rear walls of the HSM-HS.

The maximum tensile load per tie rod is 26.3 kips. The tensile load capacity of the tie rod is 188.4 kips. The maximum shear force in the vertical and longitudinal direction is 134.2 kips and 52.5 kips, respectively. The shear capacity per tie is 163.2 kips. The shear-tension interaction ratio is 0.96. Thus, the ties are qualified for the postulated seismic loads.

U.3.7.11.6.9 Thermal Cycling of the HSM-H

No change *to Section 8.2.10.5.*

U.8.1 Procedures for Loading the Cask

U.8.1.1 Preparation of the TC and DSC

All changes on this page are AMD 11

1. Prior to placement in dry storage, the candidate intact and damaged fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification *2.1.*

2. Prior to being placed in service, the TC is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification *5.2.4.d.*

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

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

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

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

7. Record the DSC serial number which is located on the grapple ring. Verify the correct DSC type, basket type and poison material types against the DSC serial number. Verify that the DSC is appropriate for the specific fuel loading campaign per the criteria specified in Technical Specification *2.1.*

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

9. If damaged fuel assemblies are included in a specific loading campaign, place the required number of bottom end caps provided (up to a maximum of 16) into the cell locations per Technical Specification *2.1.* Optionally, this step may be performed at any prior time.

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

11. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification *3.2.1.*

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

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

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

   - A cask/DSC loading plan is developed to verify that the damaged and/or intact fuel assemblies, and CCs, if applicable, meet the burnup, enrichment and cooling time parameters of Technical Specification *2.1*. AMD 11
   - The loading plan is independently verified and approved before the fuel load.
   - A fuel movement schedule is then written, verified and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance of the fuel movement schedule.
   - If loading damaged fuel assemblies, verify that the required number of bottom end caps are installed in appropriate fuel compartment tube locations before fuel load.

6. Prior to loading of a spent fuel assembly (and CCs, if applicable) into the DSC, the identity of the assembly (and CCs, if applicable) is to be verified by two individuals using an underwater video camera or other means. Verification of CC identification is optional if the CC has not been moved from the host fuel assembly since it's last verification. Read and record the identification number from the fuel assembly (and CCs, if applicable) and check this identification number against the DSC loading plan which indicates which fuel assemblies (and CCs, if applicable) are acceptable for dry storage.

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Step *6 and* 7 for each SFA loaded into the DSC. A maximum of 16 damaged fuel assemblies may be loaded into the basket per Technical Specification *2.1*. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly and CCs, if applicable, in the DSC. If loading damaged fuel assemblies, place top end caps over each damaged fuel assembly placed into the basket. AMD 11 & 72.48

8. After all the SFAs and CCs, if applicable, have been placed into the DSC and their identities verified, position the lifting yoke and the top shield plug and lower the shield plug onto the DSC. Note that separate rigging may be used to install the shield plug prior to engaging the trunnions with the lifting yoke.

   CAUTION: Verify that all the lifting height restrictions as a function of temperature specified in Technical Specification *5.3.1.A* can be met in the following steps which involve lifting of the TC. AMD 11

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

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

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

12. Inspect the top shield plug to verify that it is properly seated onto the DSC. If not, lower the cask and reposition the top shield plug. Repeat Steps 8 through 12 as necessary.

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

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

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

16. Drain a minimum of 50 gallons of water from the DSC cavity. Optionally, up to approximately 900 gallons of water (as indicated by the flowmeter) may be drained from the DSC back into the pool or other suitable location to meet the weight limit on the crane. Use 1 to 3 psig of helium to backfill the DSC with helium per ISG-22 [8.2] guidance as water is being removed from the DSC cavity.

17. Lift the TC from the fuel pool. As the cask is raised from the pool, continue to spray the cask with water and decon as directed. Provisions shall be made to assure that air will not enter the DSC cavity. One way to achieve this is by replenishing the helium in the DSC cavity during cask movement from the fuel pool to the decon area in case of malfunction of equipment used for cask movement.

18. Move the TC with loaded DSC to the cask decon area.

18a. Replace the water removed from the DSC cavity in Step 16 with water from the fuel pool or an equivalent source which meets the requirements of Technical Specifications *3.2.1*. AMD 11

19. If applicable to keep the occupational exposure ALARA, temporary shielding may be installed as necessary to minimize personnel exposure. Install cask seismic restraints if required by Technical Specification *4.3.3* (required only on plant specific basis). AMD 11

U.8.1.3 DSC Drying and Backfilling

CAUTION: During performance of steps listed in Section U.8.1.3, monitor the TC/DSC annulus water level and replenish *as* necessary *to maintain cooling*.

1. Check the radiation levels along the perimeter of the cask. The cask exterior surface should be decontaminated as necessary. Temporary shielding may be installed as necessary to minimize personnel exposure.

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

3. Disengage the rigging cables from the top shield plug and remove the eyebolts. Disengage the lifting yoke from the trunnions and position it clear of the cask.

4. Decontaminate the exposed surfaces of the DSC shell perimeter and remove the inflatable TC/DSC annulus seal.

4a. In accordance with Technical Specification 5.2.4.a, verify that the neutron shield (NS) is filled before the draining operation in Step 5 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

5. Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately twelve inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with the Technical Specification *5.2.4.d* limits. AMD 11

   CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

5a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 6 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

6. Drain approximately 900 gallons of water (as indicated on a flowmeter) from the DSC back into the fuel pool or other suitable location, *if* not drained in *Section U.8.1.2, Step 16.* Consistent with ISG-22 [8.5] guidance, helium at 1-3 psig is used to backfill the DSC with an inert gas (helium) as water is being removed from the DSC. Only helium may be used to assist in the removal of water. AMD 11

7. *Not used.*

8. Install the automatic welding machine onto the inner top cover plate and place the inner top cover plate with the automatic welding machine onto the DSC. Optionally, the inner top cover plate and the automatic welding machine can be placed separately. Verify proper fit-up of the inner top cover plate with the DSC shell.

9. Check radiation levels along surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure. AMD 11

10. Insert a 1/4-inch tubing of sufficient length and adequate temperature resistance through the vent port such that it terminates just below the DSC shield plug. Connect the flexible tubing to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner cover plate, in compliance with Technical Specification *5.2.6*. Optionally, other methods may be used for continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner top cover plate, to comply with the Technical Specification. AMD 11

11. Cover the cask/DSC annulus to prevent debris and weld splatter from entering the annulus.

12. Ready the automatic welding machine and tack weld the inner top cover plate to the DSC shell. Install the inner top cover plate weldment and remove the automatic welding machine.

    CAUTION: Continuously monitor the hydrogen concentration in the DSC cavity using the arrangement or other alternate methods described in Step 10 during the inner top cover plate cutting/welding operations. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [8.2 and 8.3]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with approximately 2-3 psig helium via the tubing to reduce the hydrogen concentration safely below the 2.4% limit.

13. Perform dye penetrant weld examination of the inner top cover plate weld in accordance with the Technical Specification *5.2.4.b* requirements. AMD 11

14. Remove purge lines and connect the VDS to the DSC siphon and vent ports.

15. Install temporary shielding to minimize personnel exposure throughout the subsequent welding operations as required.

15a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 16 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

16. a. If using blowdown method to remove water, engage helium supply (up to 15 psig) and open the valve on the vent port and allow helium to force the water from the DSC cavity through the siphon port.

    b. Alternatively a suction pump may be used to remove water from DSC.

17. Once the water stops flowing from the DSC, close the DSC siphon port and disengage the helium source or turn off the section pump, as applicable.

17a. Verify that the TC axial surface dose rates are compliant with limits specified in Technical Specification 5.2.4.e. The configuration for determining the TC axial dose rates shall be in accordance with Technical Specification 5.2.4.e. AMD 11

18. Connect the hose from the vent port and the siphon port to the intake of the vacuum pump. Connect a hose from the discharge side of the VDS to the plant's radioactive waste system or spent fuel pool. Connect the VDS to a helium source.

    Note: Proceed cautiously when evacuating the DSC to avoid freezing consequences.

    CAUTION: During the vacuum drying evolution, personnel should be in the area of loading operations, or in nearby low dose areas in order to take proper action in the event of a malfunction.

19. Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the DSC cavity. The cavity pressure should be reduced in steps of approximately 100 mm Hg, 50 mm Hg, 25 mm Hg, 15 mm Hg, 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level (these levels are optional), the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less as specified in Technical Specification *3.1.1*. AMD 11

    Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC. AMD 11

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

20. Open the valve to the vent port and allow the helium to flow into the DSC cavity.
21. Pressurize the DSC with helium up to 0 to 15 psig.
22. Helium leak test the inner top cover plate weld for a leak rate of 1 x 10^{-4} atm-cm^3 /sec. This test is optional. AMD 11
23. If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.
24. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.
25. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored level (these levels are optional). When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less in accordance with Technical Specification *3.1.1* limits. AMD 11

    Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC. AMD 11

26. Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC between *18.5 and 20.0* psig and hold for 10 min. Depressurize the DSC cavity by releasing the helium through the VDS to the plant spent fuel pool or radioactive waste system to about 2.5 psig in accordance with Technical Specification *3.1.2.b* limits. AMD 11 AMD 11

    CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

27. Close the valves on the helium source.

U.8.1.4 <u>DSC Sealing Operations</u>

CAUTION: During performance of steps listed in Section U.8.1.4, monitor the Cask/DSC annulus water level and replenish as necessary to maintain cooling.

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports. Inject helium into blind space just prior to completing welding, and perform a dye penetrant weld examination in accordance with the Technical Specification *5.2.4.b* requirements. *Use of an optional test head is acceptable to perform the helium leak test of* AMD 11 & 72.48

the inner top cover plate and vent/siphon port welds in accordance with Technical Specification 5.2.4.c. If an optional test head is not used, proceed to Step 2. AMD 11 & 72.48

2. Temporary shielding may be installed as necessary to minimize personnel exposure. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Optionally, outer top cover plate may be installed separately from the welding machine. Verify proper fit up of the outer top cover plate with the DSC shell.

3. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

4. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification *5.2.4.c* limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.4]. Alternatively this can be done with a test head in step 1 of Section U.8.1.4. AMD 11

5. If a leak is found, remove the outer cover plate root pass (if not using test head), the vent and siphon port plugs and repair the inner cover plate welds. Repeat procedure steps from U.8.1.3 Step 19.

6. Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification *5.2.4.b* requirements. AMD 11

7. Install and seal weld the prefabricated plug, if applicable, over the outer cover plate test port and perform dye penetrant weld examinations in accordance with Technical Specification *5.2.4.b* requirement. AMD 11

8. Remove the automatic welding machine from the DSC.

8a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 9 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled. AMD 11

9. Open the cask drain port valve and drain the water from the cask/DSC annulus.

10. Rig the cask top cover plate and lower the cover plate onto the TC.

11. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

    CAUTION: Monitor the applicable time limits of Technical Specification *3.1.3* until the completion of DSC transfer Step 6 of Section U.8.1.6. AMD 11

12. Verify that the *TC radial* dose rates *measured at the surface of the Transfer Cask* are compliant with limits specified in Technical Specification *5.2.4.e. The configuration for determining the TC radial surface dose rates shall be in accordance with Technical Specification 5.2.4.e.* AMD 11

U.8.1.5 TC Downending and Transfer to ISFSI

Note: Alternate Procedure for Downending of Transfer Cask: Some plants have limited floor hatch openings above the cask/trailer/skid, which limit crane travel (within the hatch opening) that would be needed in order to downend the TC with the trailer/skid in a stationary position. For these situations, alternate procedures are to be developed on a plant-specific basis, with detailed steps for downending.

1. Re-attach the TC lifting yoke to the crane hook, as necessary. Ready the transfer trailer and cask support skid for service.

2. Move the scaffolding away from the cask as necessary. Engage the lifting yoke and lift the cask over the cask support skid on the transfer trailer.

3. The transfer trailer should be positioned so that cask support skid is accessible to the crane with the trailer supported on the vertical jacks.

4. Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.

5. Move the crane forward while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.

6. Inspect the positioning of the cask to insure that the cask and trunnion pillow blocks are properly aligned.

7. Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.

8. Inspect the trunnions to ensure that they are properly seated onto the skid and install the trunnion tower closure plates, if required.

9. Remove the bottom ram access cover plate from the cask if integral rem/trailer is not used. Install the two-piece temporary neutron/gamma shield plug to cover the bottom ram access. Install the ram trunnion support frame on the bottom of the TC. (The temporary shield plug and ram trunnion support frame are not required with integral ram/trailer.)

U.8.1.6 DSC Transfer to the HSM

1. Prior to transferring the cask to the ISFSI or prior to positioning the transfer cask at the HSM designated for storage, remove the HSM door using a porta-crane, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.

   CAUTION: The insides of empty modules have the potential for high dose rates due to adjacent loaded modules. Proper ALARA practices should be followed for operations inside these modules and in the areas outside these modules whenever the door from the empty HSM has been removed.

2. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.

   CAUTION: Verify that the requirements of Technical Specification *5.3.1.B* "TC/DSC Transfer Operations at High Ambient Temperatures (32PTH1 DSC only)" are met prior to next step.

3. Using a suitable vehicle, transfer the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

4. Once at the ISFSI, position the transfer trailer to within a few feet of the HSM.

5. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

6. Using crane, unbolt and remove the cask top cover plate.

   CAUTION: Verify that the applicable time limits of Technical Specification *3.1.3* are met.

7. Back the transfer trailer to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer. Extend the transfer trailer vertical jacks.

8. Connect the skid positioning system hydraulic power unit to the positioning system via the hose connector panel on the trailer, and power it up. Remove the skid tie-down bracket fasteners and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.

9. Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.

10. Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.

11. After the cask is docked with the HSM, verify the alignment of the TC using the optical survey equipment.

12. Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug if installed. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

All changes on this page are AMD 11

14. Recheck all alignment marks in accordance with the Technical Specification *5.3.3* limits and ready all systems for DSC transfer.

15. Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening.

19. Install the DSC axial in retainer through the HSM door opening.

20. Install the HSM door using a portable crane and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Technical Specification*s 5.4.1 and 5.4.2.*

21. Replace the TC top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

22. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

23. Close and lock the ISFSI access gate and activate the ISFSI security measures.

24. Ensure the HSM-H maximum air exit temperature requirements of Technical Specification *3.1.4* are met.

All changes on this page are AMD 11

U.8.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform **one** of the two alternate daily surveillance activities listed below:

   a. A daily visual surveillance of the HSM air inlets and outlets to insure that no debris is obstructing the HSM vents in accordance with Technical Specification *5.2.5.a* requirements.

   b. A temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification *5.2.5.b* requirements.

reloaded into a shipping cask using dry transfer techniques, or if the applicant so desires, the initial fuel loading sequence could be reversed and the plant's spent fuel pool utilized. Procedures for unloading the DSC in a fuel pool are presented here. However, wet or dry unloading procedures are essentially identical to those of DSC loading through the DSC weld removal (beginning of preparation to placement of the cask in the fuel pool). Prior to opening the DSC, the following operations are to be performed.

All changes on this page are AMD 11

CAUTION: Verify that the applicable time limits of Technical Specification *3.1.3* are met until the completion of Step U.8.2.2.14.

1. The TC may now be transferred to the cask handling area inside the plant's fuel/reactor building.

2. Position and ready the trailer for access by the crane.

3. Attach the lifting yoke to the crane hook.

4. Engage the lifting yoke with the trunnions of the TC.

5. Visually inspect the yoke lifting hooks to insure that they are properly aligned and engaged onto the TC trunnions.

6. Lift the TC approximately one inch off the trunnion supports.

7. Move the crane backward in a horizontal motion while simultaneously raising the crane hook vertically and lift the TC off the trailer. Move the TC to the cask decon area.

8. Lower the TC into the cask decon area in the vertical position.

9. Wash the TC to remove any dirt which may have accumulated on the TC during the DSC loading and transfer operations.

10. Place scaffolding around the TC so that any point on the surface of the TC is easily accessible to personnel.

11. Unbolt the TC top cover plate.

12. Connect the rigging cables to the TC top cover plate and lift the cover plate from the TC. Set the TC cover plate aside and disconnect the lid lifting cables.

13. Install temporary shielding to reduce personnel exposure as required. Fill the TC/DSC annulus with clean demineralized water and place a protective cover over the annulus.

The process of DSC unloading is similar to that used for DSC loading. DSC opening operations described below are to be carefully controlled in accordance with plant procedures. This operation is to be performed under the site's standard health physics guidelines for welding, grinding, and handling of potentially highly contaminated equipment. These are to include the

U.8.3 Identification of Subjects for Safety Analysis

No Change *to Section 5.1.3.*

U.8.4 Fuel Handling Systems

No Change *to Section 5.2.*

U.8.5 Other Operating Systems

No Change *to Section 5.3.*

U.8.6 Operation Support System

No Change *to Section 5.4.*

U.8.7 Control Room and/or Control Areas

No Change *to Section 5.5.*

U.8.8 Analytical Sampling

No Change *to Section 5.6.*

Personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.2].

The weld between the DSC shell and inner top cover/shield plug and siphon/vent cover welds are also leak tested to an acceptance criteria of 1 x 10^{-7} ref cm^3/s at the field after the fuel assemblies are loaded in the canister.

All changes on this page are AMD 11

U.9.1.4 Components

The NUHOMS® System does not include any components such as valves, rupture discs, pumps, or blowers. The gaskets in the Transfer Cask do not require acceptance testing other than the leak testing cited above. No other components of the NUHOMS® System require testing, except as discussed in this chapter.

U.9.1.5 Shielding Integrity

The Transfer Cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6" × 6" grid, the detector will encompass a 6" × 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

U.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section U.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section U.9.1.7.*6*.

U.9.1.7 Poison Acceptance

> ***CAUTION***
>
> *Sections U.9.1.7.1 through U.9.1.7.4 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.*

All changes on this page are AMD 11

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated *a*luminum

(b) Boron carbide / *a*luminum metal matrix composite (MMC)

(c) *BORAL®*

The 32PTH1 DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials is given in Table U.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to *BORAL®*, which is described later in this section.

U.9.1.7.1 **Borated Aluminum**

See the Caution in Section U.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur)*. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section U.9.1.7.7. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

All changes on this page are AMD 11

U.9.1.7.2 **Boron Carbide / Aluminum Metal Matrix Composites (MMC)**

See the Caution in Section U.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

At least 50% by weight of the B_4C particles in MMCs shall be smaller than 40 microns. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 32PTH1 DSC, MMCs shall pass the qualification testing specified in Section U.9.1.7.*8*, and shall subsequently be subject to the process controls specified in Section U.9.1.7.*9*.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section U.9.1.7.*7*. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

U.9.1.7.3 ***BORAL®***

See the Caution in Section U.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. *Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns.* The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of *BORAL®*. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on a coupon taken *from* the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

All changes on this page are AMD 11

U.9.1.7.4 Visual Inspections of Neutron Absorbers

See the Caution in Section U.9.1.7 before deletion or modification to this section.

Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder's QA procedures. Material that does not meet the following acceptance criteria shall be reworked, repaired, or scrapped. Blisters shall be treated as non-conforming. Inspection of MMCs with an integral aluminum cladding shall also include verification that the matrix is not exposed through the faces of the aluminum cladding and that solid aluminum is not present at the edges. For BORAL®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet.

U.9.1.7.5 *Other Visual Inspections Criteria (non-Technical Specifications)*

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4, "Quality Control, Visual Inspection of Aluminum Mill Products" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable.

U.9.1.7.6 Thermal Conductivity Testing

Testing shall conform to ASTM E1225[1], ASTM E1461[2], or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Previous testing of borated aluminum and metal matrix composite shows that thermal conductivity increases slightly with temperature. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.

If a thermal conductivity test result is below the specified minimum, *at least four* additional tests *shall* be performed on the material from that lot. If the mean value of those tests, *including the original test,* falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the *same primary* boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The measured thermal conductivity values shall satisfy the minimum required conductivities as

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1 *ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique."*

2 *ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method."*

specified in Section U.4.3.

In cases where the specified thickness of the neutron absorber may vary, the equations introduced in Section U.4.3 shall be used to determine the minimum required effective thermal conductivity.

The thermal conductivity test requirement does not apply to aluminum that is paired with the neutron absorber.

All changes on this page are AMD 11

U.9.1.7.7 **Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission**

CAUTION

Portions of Section U.9.1.7.7 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

a) **Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.**

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to ***1.1 inch*** **diameter.**

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of B10 at that energy.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be *no more than 0.75 sq. inch.*

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. *If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.*

b) The following illustrates one acceptable method and is intended to be utilized as an example. Therefore, the following text is not part of the Technical Specifications. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [9.6].

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than *the statistically derived minimum thickness from U.9.1.7.7 a)* or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness. *Edge effects due to manufacturing operations such as shearing, deburring, and chamfering need not be included in this determination.*

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

U.9.1.7.8 Specification for Qualification Testing of Metal Matrix Composites

CAUTION

Section U.9.1.7.8.3.1, Section U.9.1.7.8.4, and Section U.9.1.7.8.5, are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

U.9.1.7.8.1 Applicability and Scope

Metal matrix composites (MMCs) *acceptable for use in the 32PTH1 DSC are described in*

All changes on this page are AMD 11

Section U.9.1.7.2.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section U.9.1.7.*9* so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the *C*ertificate *H*older.

All changes on this page are AMD 11

U.9.1.7.8.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section U.9.1.7.*8*.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section U.9.1.7.*8*.5.

U.9.1.7.8.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

*T*hermal damage and corrosion (hydrogen generation) testing shall be *performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.*

Thermal damage testing is not required for *unclad* MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for MMCs *(clad or unclad)* consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

U.9.1.7.8.3.1 ***Delamination Testing of Clad MMC***

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage.

[3] Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

[4] Boralyn testing submitted to the NRC under docket 71-1027, 1998.

U.9.1.7.8.4 **Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity**

At least three samples, one each from *approximately* the two ends and middle of the *qualification* material run shall be subject to:

a) **room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:**

- **Minimum yield strength, 0.2% offset: 1.5 ksi**
- **Minimum ultimate strength: 5 ksi**
- **Minimum elongation in 2 inches: 0.5%**

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[6]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture,

b) **Testing to verify more than 98% *of theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs.* Testing or examination for interconnected porosity *on the faces and edges of unclad MMC, and on the edges of clad MMC* shall be performed by a means to be approved by the Certificate Holder. *The maximum interconnected porosity is 0.5 volume %,***

and for at least one sample,

c) ***For MMCs with an integral aluminum cladding, thermal durability testing demonstrating that after a minimum 24 hour soak in either pure or borated water, then insertion into a preheated oven at approximately 825°F for a minimum of 24 hours, the specimens are free of blisters and delamination and pass the mechanical testing requirements described in test 'a' of this section.***

U.9.1.7.8.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

Uniformity of the boron distribution shall be verified either by:

a) **Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or**

b) **Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run,**

All changes on this page are AMD 11

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5 ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

6 *ASTM E290, Standard Methods for Bend Testing of Materials for Ductility.*

7 ASTM E94, Recommended Practice for Radiographic Testing.

8 ASTM E142, Controlling Quality of Radiographic Testing.

9 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing.

verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section U.9.1.7.7, or by chemical analysis for boron carbide content in the composite.

All changes on this page are AMD 11

U.9.1.7.8.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

U.9.1.7.9 Specification for Process Controls for Metal Matrix Composites

CAUTION
Sections U.9.1.7.9.1 and U.9.1.7.9.2 are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

U.9.1.7.9.1 Applicability and Scope

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section U.9.1.7.*8* is required, depending on the characteristics of the material that could be affected by the process change.

U.9.1.7.9.2 **Definition of Key Process Changes**

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, *reduce corrosion resistance,* reduce the mechanical strength or ductility of the MMC.

U.9.1.7.9.3 Identification and Control of Key Process Changes

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section U.9.1.7.*9*.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that *are* established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a *magnesium-alloyed* aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature,

f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending, *and*

g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

In no case shall process changes be accepted if they result in a product outside the limits in Sections *9.5.3.1* and *9.5.3.4*.

All changes on this page are AMD 11

U.9.3 References

9.1 ASME Boiler and Pressure Vessel Code, Section III, 1998 Edition with 2000 Addenda.

9.2 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

9.3 *Deleted.*

9.4 ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," February 1998.

9.5 "Aluminum Standards and Data, 2003" The Aluminum Association.

9.6 Natrella, "Experimental Statistics," Dover, 2005.

9.7 *Deleted.*

9.8 *Deleted.*

9.9 *Deleted.*

9.10 *Deleted.*

All changes on this page are AMD 11

Table U.9-1
B10 Specification for the NUHOMS® 32PTH1 Poison Plates

Poison Type	32PTH1 Basket Type	Minimum Poison Loading (B10 mg/cm^2)	% Credit Used in Criticality Analysis
Borated Aluminum /MMC	1A or 2A	7	90
	1B or 2B	15	
	1C or 2C	20	
	1D or 2D	32	
	1E or 2E	50	
BORAL®	1A or 2A	9	75
	1B or 2B	19	
	1C or 2C	25	
	1D or 2D	N/A	
	1E or 2E	N/A	

All changes on this page are AMD 11

All changes on this page are AMD 11

U.11.1.2 Extreme Temperatures

No change. The off-normal maximum ambient temperature of 125°F is used in UFSAR Section 8.1.2.2. For the NUHOMS® 32PTH1 system, a maximum ambient temperature of 117°F is used. Chapter U.3, Section U.3.4.4.3 summarizes the thermal analysis for the 32PTH1 DSC, HSM-H (HSM-HS) and OS200 TC.

U.11.1.2.1 Postulated Cause of Event

No change. See UFSAR Section 8.1.2.2.

U.11.1.2.2 Detection of Event

No change. See UFSAR Section 8.1.2.2.

U.11.1.2.3 Analysis of Effects and Consequences

The thermal evaluation of the NUHOMS® 32PTH1 system for off-normal conditions is presented in Chapter U.4. The 106°F normal condition with insolation bounds the 117°F case without insolation for the DSC in the TC. Therefore the normal condition maximum temperatures are bounding. The 117°F case with the DSC in the HSM-H is not bounded by the normal conditions and therefore evaluated in Chapter U.4.

The structural evaluation of the 32PTH1 DSC for off-normal temperature conditions is presented in Chapter U.3, Section U.3.6.2.2. The structural evaluation of the basket due to off-normal thermal conditions is presented in Chapter U.3, Section U.3.6.1.3. The structural evaluation of HSM-H and OS200 Transfer Cask for off-normal conditions with 32PTH1 DSC are presented in Chapter U.3, Section U.3.6.2.3 and Chapter U.3, Section U.3.6.2.4, respectively.

U.11.1.2.4 Corrective Actions

Restrictions for onsite handling of the TC with a loaded DSC under extreme temperature conditions are presented in Technical Specifications *5.3.1.A* and *5.3.1.B*.

U.11.1.3 Off-Normal Releases of Radionuclides

The NUHOMS® 32PTH1 DSC is designed and tested to the leak tight criteria of ANSI N14.5 [11.2]. Therefore the estimated quantity of radionuclides expected to be released annually to the environment due to normal or off-normal events is zero.

U.11.1.3.1 Postulated Cause of Event

In accordance with the Standard Review Plan, NUREG-1536 [11.3] and ISG-5 Rev. 1 [11.4] for off-normal conditions, it is conservatively assumed that 10% of the fuel rods fail.

U.11.1.3.2 Detection of Event

Failed fuel rods would go undetected, but are not a safety concern since the canister is designed and tested to the leak tight criteria of ANSI N14.5 [11.2].

V.1.2.2.2 Sequence of Operations

The sequence of operations to be performed in loading a DSC containing spent nuclear fuel into the NUHOMS® HSM Model 202 is presented in Chapter V.8.

V.1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

V.1.2.2.3.1 Criticality Prevention

No change *to Section 5.1.3.1.*

V.1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the NUHOMS® HSM Model 202 system.

V.1.2.2.3.3 Operation Shutdown Modes

The NUHOMS® HSM Model 202 is a totally passive system so that consideration of operation shutdown modes is unnecessary.

V.1.2.2.3.4 Instrumentation

No change *to Section 5.1.3.4.*

V.1.2.2.3.5 Maintenance Techniques

No change *to Section 5.1.3.5.*

V.1.2.3 Cask Contents

No change *to Section 1.2.3.*

V.1.3 Identification of Agents and Contractors

Transnuclear, Inc. (TN) provides the design, analysis, licensing support and quality assurance for the NUHOMS® HSM Model 202 system. Fabrication of the NUHOMS® HSM Model 202 is done by one or more qualified fabricators under TN's quality assurance program described in Chapter V.13. This program is written to satisfy the requirements of Subpart G of 10CFR72, [1.2] and covers control of design, procurement, fabrication, inspection, testing, operations and corrective action. Experienced TN operations personnel will assist in the preparation of generic operating procedures and provide training to utility personnel prior to their first use of the NUHOMS® HSM Model 202 system.

Managerial and administrative controls, which are used to ensure safe operation of the casks, will be provided by the host utility. NUHOMS® HSM Model 202 system operations and maintenance will be performed by utility personnel. Decommissioning activities will be performed by utility personnel in accordance with site procedures.

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

V.1.4 Generic Cask Arrays

No change *to Section 1.2.1.*

V.1.5 Supplemental Data

The following Transnuclear drawing is enclosed:

1. NUH-03-7002-SAR Standardized NUHOMS® ISFSI HSM Model 202 Main Assembly

V.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."

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
2	REVISED PER FCN 721004-1063	01/08/14
1	REVISED PER FCNS 721004-614 AND 721004-687	01/28/10
0	INITIAL ISSUE PER FCNS 721004-338, 721004-352 AND 721004-445	01/29/08

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
STANDARDIZED NUHOMS® ISFSI
HSM MODEL 202
MAIN ASSEMBLY

DRAWING NO. NUH-03-7002-SAR | SCALE NONE | SHEET 1 OF 1

Table V.1-1
Comparison of Key Parameters of NUHOMS® HSM Model 202 versus HSM Model 80 and Model 102

Characteristic	HSM Model 80	HSM Model 102	HSM Model 202
Overall Length (without Shield Walls)	19'-10" (BWR)	19'-10" (BWR)	20'-8"
Overall Width (without Shield Walls)	9'-8"	9'-8"	9'-8"
Overall Height	15'-0"	15'-0"	18'-6" (without vent cover)
Roof Thickness	3'-0"	3'-0"	3'-8"
End Shield Wall Thickness	2'-0"	2'-0"	3'-0"
Rear Shield Wall Thickness	2'-0"	2'-0"	3'-0"
Side Wall Thickness	1'-6"	1'-6"	1'-0
Back Wall Thickness	1'-0"	1'-0"	1'-0"
Front Wall Thickness	2'-6"	2'-6"	3'-6"
Floor Thickness	1'-0"	1'-0"	N/A
Door Construction	~ 8" thick consisting of concrete core (~ 6") encased by stainless steel (2")	24" thick consisting of reinforced concrete	Min. of 18-1/2"-thick reinforced concrete attached to a 7-7/8" thick steel plate **Optional Door:** Min. of 25 3/8" thick reinforced concrete attached to a 3" thick steel plate
Inlet Vent Configuration	4 along lower side walls	4 along lower side walls	2 along bottom of side walls
Inlet Vent Area	1200 in^2	1200 in^2	2368 in^2
Outlet Vent Configuration	4 along upper side walls	4 along upper side walls	2 along upper side walls
Outlet Vent Area	1680 in^2	1680 in^2	2368 in^2
Gap Between Adjacent Modules Placed Side-By-Side	6"	6"	0"
Bird Screen Type	Wire Cloth 3/4" mesh x *0.120"* wire	Wire Cloth 3/4" mesh x *0.120"* wire	Wire Cloth 3/4" mesh x *0.120"* wire
Weight – Base Unit (including HSM support steel)	164,403 lbs	167,267 lbs	178,424 lbs
Weight – Roof	80,970 lbs	82,486 lbs	107,261 lbs
Weight Door	6,556 lbs	11,200 lbs	21,510 lbs
DSC Support Steel Configuration	Structural steel frame with rails installed to permit sliding of DSC	Structural steel frame with rails installed to permit sliding of DSC	Guide rails bolted to concrete to permit sliding of DSC
Heat Shield Thickness	12 Gauge (0.1054") Galvanized Steel	12 Gauge (0.1054") Galvanized Steel	2" x 1/8" thick Aluminum Plates to form a Louvered Roof Heat Shield and 1/4" thick Anodized Aluminum Side Heat Shields Alternate Heat Shield Configuration: 12 Gauge (0.1054") flat stainless steel top and side Heat Shields

V.2 Principal Design Criteria

This section provides the principal design criteria for the NUHOMS® HSM Model 202 system. With the exception of the seismic design criteria and the tornado wind pressure loads, the principal design criteria for the NUHOMS® HSM Model 202 are the same as the NUHOMS® HSM Model 80 and Model 102 as described in Chapter 3. Section V.2.1 presents a general description of the spent fuel to be stored. Section V.2.2 provides the design criteria for environmental conditions and natural phenomena. Section V.2.3 provides a description of the systems which have been designated as important to safety. Section V.2.4 discusses decommissioning considerations. Section V.2.5 summarizes the NUHOMS® HSM Model 202 design criteria.

V.2.1 Spent Fuel To Be Stored

The NUHOMS® DSCs are designed to store a total of 24 or 32 PWR fuel assemblies and 52 or 61 BWR fuel assemblies with the same characteristics as those described, respectively, in Chapter 3 and Appendices Chapters J.1.1, K.2, L.2, M.2, N.2, and P.2.

V.2.1.1 General Operating Functions

No change *to Section 3.1.2.*

V.2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The NUHOMS® HSM Model 202 is handled and utilized in the same manner as the existing NUHOMS® HSM Model 80 and Model 102 systems. The environmental conditions, natural phenomena and design criteria are the same as described for the NUHOMS® HSM Model 80 and Model 102 in Chapter 3. Design criteria for the NUHOMS® DSC and TC remain unchanged.

V.2.2.1 Tornado Wind and Tornado Missiles

No change. The applicable maximum design pressures for the design basis tornado evaluations are presented in Sections 3.2.1.1 and 3.2.1.2. The missile criteria used for the HSM Model 202 bounds that provided in Section 3.2.1.2.

V.2.2.2 Water Level (Flood) Design

No change *to Section 3.2.2.*

V.2.2.3 Seismic Design

No change to the seismic criteria described in Section 3.2.3. However, the HSM Model 202 is designed to withstand a horizontal ground acceleration of 0.30g and a vertical ground acceleration of 0.20g, which bounds the seismic criteria used in designing the NUHOMS® HSM Model 80 and Model 102 systems. The seismic design criteria for the HSM Model 80 and HSM Model 102 are 0.25g horizontal and 0.17g vertical acceleration.

The results of the frequency analysis of the HSM Model 202 structure (which includes a simplified model of the DSC having a bounding weight of 110,000 lbs which results in a lower bound frequency) yields a lowest frequency of 23.2 Hz in the transverse direction and 28.4 Hz in the longitudinal direction. The lowest vertical frequency exceeds 33 Hz. Thus, based on the R.G. 1.60 response spectra amplifications, the corresponding seismic accelerations used for the design of the HSM Model 202 are 0.37g and 0.33g in the transverse and longitudinal directions, respectively, and 0.20g in the vertical direction. These are conservative acceleration values since they are based on amplifications from R.G. 1.60 spectra, anchored at 0.3g horizontal and 0.2g vertical. The corresponding accelerations applicable to the DSC are 0.41g and 0.36g in the transverse and longitudinal directions, respectively, and 0.20g in the vertical direction. The seismic analysis of the HSM-H and 24PTH DSC are further discussed in Section P.3.7.

From Section 3.2.3, the lowest frequency of the HSM Model 80 and Model 102 is 19.1 Hz. Because the lowest frequency of the loaded HSM Model 80 and Model 102 is lower than the lowest frequency of HSM Model 202 (loaded with a bounding DSC weight of 110,000 lbs) of 23.2 Hz, the seismic acceleration used for evaluation of the DSCs remain bounding relative to HSM Model 202 seismic accelerations.

V.2.2.4 Snow and Ice Loading

No change *to Section 3.2.4.*

V.2.2.5 Combined Load Criteria

No change. The load combination criteria used for the HSM Model 202 has been reconciled and bound that given in Section 3.2.5.1.

V.2.3 Safety Protection Systems

V.2.3.1 General

No change *to Section 3.3.1.*

V.2.3.2 Protection By Multiple Confinement Barriers and Systems

No change *to Section 3.3.2.*

V.2.3.3 Protection By Equipment and Instrumentation Selection

No change *to Section 3.3.3.*

V.2.3.4 Nuclear Criticality Safety

V.2.3.4.1 Control Methods for Prevention of Criticality

No change *to Section 3.3.4.*

V.2.3.4.2 Error Contingency Criteria

No change *to Section 3.3.4.*

V.2.3.4.3 Verification Analysis-Benchmarking

No change *to Section 3.3.4.*

V.2.3.5 Radiological Protection

No change *to Section 3.3.5.*

V.2.3.6 Fire and Explosion Protection

No change *to Section 3.3.6.*

V.2.4 Decommissioning Considerations

No change *to Section 3.5.*

V.2.5 Summary of NUHOMS® HSM Model 202 Design Criteria

The principal design criteria for the NUHOMS® HSM Model 202 are based on those presented for the NUHOMS® HSM-H shown in Table P.2-18, and reconciled against the HSM Model 80 and Model 102 criteria in Table 3.2-1. The NUHOMS® HSM Model 202 is designed to store a DSC loaded with PWR or BWR fuel assemblies identical to those stored in a NUHOMS® HSM Model 80 or Model 102 as described in Chapter 3 and Appendices Chapters J.2, K.2, L.2, M.2, N.2, and P.2.

V.3 Structural Evaluation

V.3.1 Structural Design

V.3.1.1 Discussion

This section describes the structural evaluation of the NUHOMS® HSM Model 202 system. The HSM Model 202 is based on to the NUHOMS® HSM-H described in Appendix P.3, and is capable of storing the 24P, 52B, 61BT, 24PT2, 32PT, 24PHB, and 24PTH-S-LC DSCs. Sections that do not have an effect on the evaluations presented in the UFSAR include a statement that there is no change to the UFSAR.

V.3.1.1.1 General Description of the HSM Model 202

The HSM Model 202 is a freestanding reinforced concrete structure designed to provide environmental protection and radiological shielding for the DSC. The HSM Model 202 is designed to accommodate all DSC configurations with a maximum heat-load of 24 kW (24P, 52B, 61BT, 24PT2, 32PT, 24PHB, and 24PTH-S-LC DSC). Each HSM Model 202 provides a self-contained modular structure for the storage of a DSC containing up to 32 PWR spent fuel assemblies (SFAs) or 61 BWR SFAs. The HSM Model 202 is based on the HSM-H described in Section P.3.1.1.2.

V.3.1.2 Design Criteria

The design criteria for the HSM Model 202 are provided in Section V.2.2. The design criteria for the DSCs, as presented in Chapter 3 and Appendices K, L, M, N and P (24PTH-S-LC only); and TC, as presented in Chapter 3, are not changed.

V.3.2 HSM Weights

Table V.3-1 shows the weights of the various components of the NUHOMS® HSM Model 202 system. The dead weights of the components are determined based on the nominal dimensions.

V.3.3 Mechanical Properties of Materials

The material and section properties used for different components of the HSM Model 202 and the internal DSC support structure are identical to those used for the HSM-H as described in Section P.3.3.2.

V.3.4 General Standards for Casks

No change to *Sections 3.4.1, K.3.4, M.3.4, P.3.4, T.3.4, and U.3.4.*

V.3.5 Fuel Rods

No change to *Section 3.1.1.*

V.3.6 Structural Analysis (Normal and Off-Normal Operations)

In accordance with NRC Regulatory Guide 3.48 [3.1] the design events identified by ANSI/ANS 57.9-1984, [3.2] form the basis for the accident analyses performed for the Standardized NUHOMS® system. Four categories of design events are defined. Design event Types I and II cover normal and off-normal events and are addressed in Section 8.1. Design event Types III and IV cover a range of postulated accident events and are addressed in Section 8.2. The purpose of this section of the appendix is to present the structural analyses for normal and off-normal operating conditions for the NUHOMS® HSM Model 202 system using a format similar to the one used in Section 8.1 for analyzing the NUHOMS® HSM Model 80 and Model 102 systems.

V.3.6.1 Normal Operation Structural Analysis

Table 8.1-1 shows the normal operating loads for which the NUHOMS® safety-related components are designed. The table also lists the individual NUHOMS® components which are affected by each loading. The magnitude and characteristics of each load are described in Section V.3.6.1.1.

The method of analysis and the analytical results for each load are described in Section V.3.6.1.4.

V.3.6.1.1 Normal Operating Loads

The normal operating loads for the NUHOMS® system components are:

1. Dead Weight Loads
2. Design Basis Internal and External Pressure Loads
3. Design Basis Thermal Loads
4. Operational Handling Loads
5. Design Basis Live Loads

These loads are described in detail in the following paragraphs.

A. Dead Weight Loads

Table V.3-1 shows the weights of various components of the NUHOMS® HSM Model 202 system. The deadweight of the component materials is determined based on nominal component dimensions.

B. Design Basis Internal and External Pressure

No change *to Section 8.1.1.1.B.*

V.3.7.5 Lightning

No change *to Section 8.2.6.*

V.3.7.6 Blockage of HSM Model 202 Air Inlet and Outlet Openings

This accident conservatively postulates the complete blockage of the HSM Model 202 ventilation air inlet and outlet openings on the HSM Model 202. Since the NUHOMS® HSM Model 202s are located outdoors, there is a remote probability that the ventilation air inlet and outlet openings could become blocked by debris. The NUHOMS® design features such as the perimeter security fence, the above ground location of the air inlet opening and protected location of the outlet vent opening and the vent screens reduces the probability of occurrence of such an accident. Nevertheless, for this conservative generic analysis, such an accident is postulated to occur and is analyzed.

The structural consequences due to the weight of the debris blocking the air inlet and outlet vent openings are negligible and are bounded by the HSM Model 202 loads induced for a postulated tornado (Section V.3.7.2) or earthquake (Section V.3.7.3).

The thermal effects of this accident for various NUHOMS® DSCs with a 24 kW heat load are described in Sections V.4 and V.11.

V.3.7.7 HSM Model 202 Load Combination Evaluations

The load categories associated with normal operating conditions, off-normal conditions and postulated accident conditions are described and analyzed in previous sections. The load combination results for the NUHOMS® HSM Model 202 are presented in this section. The load combinations used for the evaluation of the HSM Model 202 are as described in Table P.3.7-16 and P.3.7-17, and are equivalent to those shown in Table 3.2-5 and 3.2-8 for the concrete and steel components, respectively.

V.3.7.8.4 Evaluation of HSM Model 202 Support Steel

The evaluation of the HSM Model 202 support steel is described in Section P.3.7.11.6.4 since the HSM Model 202 is based on the HSM-H.

V.3.7.8.5 Evaluation of HSM Model 202 Shield Door

The evaluation of the HSM Model 202 shield door is described in Section P.3.7.11.6.5 since the HSM Model 202 is based on the HSM-H.

V.3.7.8.6 Evaluation of HSM Model 202 Heat Shields

The evaluation of HSM Model 202 heat shields is described in Section P.3.7.11.6.6 since the HSM Model 202 is based on the HSM-H.

V.3.7.8.7 Evaluation of HSM Model 202 Seismic Retainers

The evaluation of HSM Model 202 seismic retainers is described in Section P.3.7.11.6.7 since the HSM Model 202 is based on the HSM-H.

V.3.7.8.8 Thermal Cycling of the HSM Model 202

No change *to Section 8.2.10.5.*

V.3.7.8.9 Evaluation of HSM Model 202 Concrete Components with Temperature Exceeding Code Limits

The maximum concrete temperature under off-normal condition for the HSM Model 202 are 238/243°F (for 117°F and 125°F ambient conditions). The normal condition is bounded by the off-normal condition. Although the maximum concrete temperatures exceed 225°F in the off-normal condition, they do not exceed 300°F. Therefore, as specified in [3.4], no tests or reduction in concrete strength are required to demonstrate the capability of the concrete to adequately handle the elevated temperatures provided Type II cement is used and special aggregates are selected which are acceptable for concrete in this temperature range. This approach is consistent with standardized HSM design, for which special aggregates for the roof concrete mix are provided.

The maximum concrete temperature for a 40-hour blocked vent condition is 376/381°F (for 117°F and 125°F ambient conditions), which exceeds the 350°F limit specified in [3.4]. As noted in [3.4], use of any Portland cement concrete where accident temperature exceeds 350°F will require testing be performed on the exact concrete mix. Elevated temperature testing of the exact concrete mix (cement type, additives, water-cement ratio, aggregates, proportions) is to be performed for the HSM Model 202. The use of high temperature concrete testing is explicitly accepted by the NRC, as documented in the NRC's SER [3.4], Section 3.0, Page 3-5. The testing shall demonstrate the level of strength reduction is less than that which was applied, and show that the increased temperatures do not cause deterioration of the concrete.

V.9 Acceptance Tests and Maintenance Program

V.9.1 Acceptance Tests

The addition of the HSM Model 202 to the Standardized NUHOMS® system does not result in any change to the Pre-Operational Tests described in Section 9.2 since the transfer cask involved is not changed and the HSM Model 202 is very similar to the HSM Model 102 from an operations perspective.

Prior to operation of the ISFSI for a particular plant, the licensee should perform functional tests of the in-plant operations, the on-site transfer operations, and DSC insertion and retrieval (operations at the ISFSI). These tests are intended to verify that the storage system components (e.g., DSC, HSM, transfer cask, transfer equipment, etc.) operate safely and effectively. Such a program has been successfully completed for the NUHOMS® ISFSIs at Duke Power Company's Oconee Nuclear Station, Baltimore Gas and Electric Company's Calvert Cliffs Nuclear Power Plant, Toledo Edison's Davis Besse Nuclear Station and Pennsylvania Power and Light's Susquehanna Nuclear Station.

V.9.1.1 Visual Inspection

Visual inspections are performed at the fabricator's facility to ensure that the DSC and the HSM conform to the drawings and specifications. The visual inspections include verifying dimensions and the application of specified coatings and that the DSC is clean and free of defects. Visual inspections are performed in accordance with the requirements and acceptance criteria specified by the codes applicable to the associated components.

Upon arrival at the site, the DSCs and HSMs are again inspected to ensure that they have not been damaged during shipment. Conditions which are not in conformance with the drawings and specifications will be repaired or evaluated, in accordance with 10CFR 72.48, for the effect of the condition on the safety function of the components.

V.9.1.2 Structural Tests

No change *to Section 9.2* associated with the addition of the HSM Model 202.

V.9.1.3 Leak Tests and Pressure Tests

No change *to Section 9.2* associated with the addition of the HSM Model 202.

V.9.1.4 Component Tests

No change *to Section 9.2* associated with the addition of HSM Model 202.

V.9.1.5 Shielding Integrity Tests

No change *to Section 9.2* associated with the addition of the HSM Model 202.

V.9.1.6 Thermal Acceptance Tests

AMD 11 & 72.48

No change *to Section 9.2* associated with the addition of the HSM Model 202.

V.9.1.7 Neutron Absorber Tests

No change *to Section 9.2* associated with the addition of the HSM Model 202.

V.9.2 Maintenance Program

The NUHOMS® HSM Model 202 system is designed to be totally passive and require minimal maintenance. The DSC does not require any maintenance once it is loaded into the HSM Model 202.

V.9.3 Training Program

No change *to Section 9.3*.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

All changes on this page are AMD 11

W.1 General Description

Appendix W to the NUHOMS® *u*pdated *f*inal *s*afety *a*nalysis *r*eport (UFSAR) addresses the *i*mportant to *s*afety aspects of adding the OS197L TC to the *s*tandardized NUHOMS® system described in the UFSAR. The OS197L TC is added to the UFSAR as an alternative to the OS197 and OS197H TCs. The primary reason for adding the OS197L TC design is to include a *reduced weight* transfer cask that can be used *to transfer specific payloads at* facilities.

The format of this *a*ppendix follows the guidance provided in NRC Regulatory Guide 3.61 [1.1]. The analyses presented in this Appendix demonstrate that the OS197L TC system meets all the requirements of 10 CFR 72 [1.2].

Several sections of this *a*ppendix have been identified as "No Change." For these sections, the description or analysis presented in the *identified* sections of the UFSAR for the *s*tandardized NUHOMS® system is also applicable to the OS197L TC. In addition, tables and figures presented in the UFSAR which remain unchanged due to the addition of the OS197L TC to the *s*tandardized NUHOMS® system are not repeated in this *a*ppendix. Table W.1-2 provides a summary of the sections of the main body of the UFSAR applicable to the OS197 TC and addresses the impact of the OS197L TC on these sections.

Note: References to sections or chapters within this *a*ppendix are identified with a prefix W (e.g., Section W.2.3 or Chapter W.2). References to sections or chapters of the UFSAR outside of this *a*ppendix (i.e., main body of the UFSAR) are identified with the applicable UFSAR section, chapter number or *a*ppendix number (e.g., Section 2.3, Chapter 2 or Appendix K). The references used in this *a*ppendix are identified as [X.X] (e.g., [1.1] is Reference 1.1 at the end of Chapter W.1).

OS197 and OS197H TCs in the remainder of this *a*ppendix will be referred to as OS197 TC.

W.1.1 Introduction

As stated in Section *1.3.2.1,* the body of this UFSAR is dedicated to *four* on-site transfer cask types: the *standardized* cask, NUHOMS®-OS197, NUHOMS®-OS197H, *and OS200* TCs. The purpose of this *a*ppendix is to provide the safety analysis of the design of a *fifth* type of on-site transfer cask, designated as the NUHOMS® OS197L TC, for use with the standardized NUHOMS® system.

W.1.2 General Description of the NUHOMS® OS197L TC

The OS197L TC on-site transfer cask is designed to accommodate *fuel transfer needs of plants where the payload is limited to a maximum of 13.0 kW.* The major differences between the OS197L TC and the OS197 casks are:

- *r*educed cask weight;

- *no integral* lead shielding (one 2.68" nominal thickness steel shell instead of a combination of a 0.5" nominal thickness steel inner liner, 3.5" nominal thickness lead shield and 1.5" nominal thickness steel structural shell);

- *only authorized for transfer of the NUHOMS®-61BT and 32PT DSCs with a maximum heat load of 13.0 kW;*
- *o*ne-piece solid trunnion configuration for the upper and lower cask trunnions;
- *t*wo-piece neutron shield (inner and outer shell of 1/4" nominal thickness versus an outer shell of 3/16" nominal thickness);
- *a 6" nominal thickness steel decontamination area supplemental shield (see Figure W.1-2) within which the cask is placed for personnel shielding during fuel loading operations;*
- *a cask support skid supplemental shielding (see Figure W.1-3), described in Section W.1.2.1.1, to be used for personnel shielding during transfer operations;*
- *remote crane operations in conjunction with laser/optical targeting and cameras are to be used for handling the OS197L TC when it is not within the decontamination area shielding.*

The OS197L TC key design parameters are compared to the OS197 TC in Table W.1-1.

The OS197L TC *when used in conjunction with the supplemental shielding provided (see Figures W.1-2 and W.1-3), including the remote cask handling procedures described in Chapter W.8,* provides shielding and protection from potential hazards during the DSC fuel loading/unloading operations and transfer to the *h*orizontal *s*torage *m*odule (HSM). The design and configuration of the OS197L TC is a modified version of the NRC approved OS197 and OS197H TCs described in Section 1.3.2.1 of the UFSAR and is limited to on-site use under 10CFR72. The OS197L TC can be configured to meet a gross weight limit of *77* Te (*85* tons).

The empty weight of the OS197L, with the neutron shield full of water and the stainless steel top cask lid installed is approximately 62,000 lb (31 tons). The nominal loaded weight in the "wet" configuration (water in the DSC, water in the DSC/TC annulus, top cask lid not installed) is approximately 85 tons. The nominal loaded weight in the "dry" configuration (after water in the DSC and DSC/TC annulus has been drained and the top cask lid is installed) is approximately 82 tons. SAR Table W.3-1 has been revised to provide this additional weight information.

Figure W.1-1 provides an overview of the OS197L TC *without the supplemental shielding.* The OS197L TC configuration also requires the use of additional shielding in the decontamination area (see Figure W.1-2) and on the skid/trailer (see Figure W.1-3).

W.1.2.1.1 Transfer Equipment

Transfer Trailer: The NUHOMS® OS197L TC transfer trailer consists of a heavy industrial trailer with a payload capacity of 136 Te (150 tons), including the skid and loaded cask. The OS197L TC transfer trailer is the same as the one shown in Figure 1.3-7 of the UFSAR.

All changes on this page are AMD 11

Cask Support Skid: The OS197L TC support skid differs from the OS197 TC support skid shown in UFSAR Figure 1.3-8 as described below:

All changes on this page are AMD 11

1. The OS197L TC support skid has permanently mounted 2.5" thick side shielding and accommodates an additional 3" thick side shielding bolted to the permanent shielding when transferring the OS197L TC.

2. The OS197L TC *support skid* also has a 2.5" shielding inner top cover and an additional 3" shielding outer top cover to shield the upper sections of the cask.

The OS197L TC support skid utilized for the standardized NUHOMS® system is illustrated in Figure W.1-3.

Hydraulic Ram: The high capacity hydraulic ram system is similar to the hydraulic ram system described in the UFSAR. The capacity of this ram is increased in order to increase the ram capacity margin (and to accommodate other future DSC designs). There is no change to the maximum ram forces allowed (80 kips) during system operation.

A picture of the OS197L TC system is provided in Figure W.1-4.

W.1.2.2 Operational Features

The primary operations with the OS197L TC (in sequence of occurrence) for the NUHOMS® system are the same as the systems operation described in Section 1.3.3 of the UFSAR except as noted below for operations 8 and 13 (of Section 1.3.3):

Lifting Cask from Pool: The loaded OS197L TC is lifted out of the pool *for placement* (in the vertical position) in a decontamination area shield on the drying pad in the decon pit. During *bare* cask movement from the fuel pool to the decontamination area, remote crane operation*s in conjunction with laser/optical targeting and cameras or other similar equipment for confirmation of the cask locations are to* be used to minimize personnel exposure due to the reduced shielding configuration of the OS197L TC during this transit movement. The licensee shall meet the specific radiation protection program requirements associated with the use of OS197L TC as specified in applicable Technical Specification *5.2.4.a.*

The cask is then placed inside the decontamination area lower shield and the upper shield or bell is then placed on top (see Figure W.1-2).

Placement of Cask on Transfer Trailer Skid: The OS197L TC is then lifted onto the cask support skid. The plant's crane is used to downend the cask from a vertical to a horizontal position. *The inner top skid* shielding is added to the skid and the cask is also covered with an additional outer top shielding. The outer top additional *skid* shielding is to be installed inside the fuel handling building if the floor loads can accommodate it (if floor loading is a concern, the *outer top trailer* shielding may be placed on the skid outside the fuel handling building). The cask is then secured to the skid and readied for the subsequent transfer operations.

The "cask support skid supplemental shielding" described in the SAR drawing (NUH-03-8011-SAR) refers to the temporary shielding required for the OS197L TC during transfer operations. This shielding is also referred to as "shielding, supplemental shielding, trailer area shielding, cask skid supplemental shielding, supplemental OS197L TC shielding, supplemental skid shielding, shield transfer trailer, auxiliary shielding, shielded transfer skid, transfer skid shields, supplementary shielding, supplemental trailer shields, skid shielding, and trailer shielding" in the safety analysis descriptions interchangeably through out this appendix.

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W.1.3 Identification of Agents and Contractors

Transnuclear, Inc. (TN) provides the design, analysis, licensing support and quality assurance for the NUHOMS® OS197L TC. Fabrication of the NUHOMS® OS197L TC is done by one or more qualified fabricators under TN's quality assurance program described in Chapter W.13. This program is written to satisfy the requirements of Subpart G of 10CFR72, [1.2] and covers control of design, procurement, fabrication, inspection, testing, operations and corrective action.

TN provides specialized services for the nuclear fuel cycle that support transportation, storage and handling of spent nuclear fuel, radioactive waste and other radioactive materials. TN is the holder of NUHOMS® CoC 1004 [1.3].

W.1.4 Generic Cask Arrays

No change. *The content presented in Sections 1.2.1 and 1.3.4 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.1.5 Supplemental Data

The following TN drawings are enclosed:

1. NUHOMS® OS197L Onsite Transfer Cask, Cask Body Assembly, Drawing NUH-03-8008-SAR.

2. NUHOMS® OS197L Onsite Transfer Cask, Light Neutron Shield Assembly, Drawing NUH-03-8009-SAR.

3. NUHOMS® OS197L Onsite Transfer Cask, OS197L Main Assembly, Drawing NUH-03-8010-SAR.

4. *NUHOMS® OS197L Onsite Transfer Cask Support Skid Supplemental Shielding, Drawing NUH-03-8011-SAR.*

5. *NUHOMS® OS197L Onsite Transfer Cask Decon Area Cask Shielding Assemblies, Drawing NUH-03-8012-SAR.*

All changes on this page are AMD 11

W.1.6 References

1.1 U.S. 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 NUHOMS® Certificate of Compliance for Dry Spent Fuel Storage Casks, Certificate Number 1004, Amendment No. *9, April 2007* (Docket 72-1004).

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER FCN 721004-1063	01/08/14
0	INITIAL ISSUE PER FCN 721004-321	01/29/08

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® OS197L
ONSITE TRANSFER CASK
CASK BODY ASSEMBLY

DRAWING NO. NUH-03-8008-SAR | SCALE NONE | SHEET 1 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-8008-SAR	3 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-8008-SAR	4 OF 8	1

NUH-03-8008-SAR 5 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH-03-8008-SAR 5 OF 8 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-8008-SAR	6 OF 8	1

PROPRIETARY AND
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DRAWING NO. NUH-03-8008-SAR | SHEET 7 OF 8 | REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® OS197L
ONSITE TRANSFER CASK
LIGHT NEUTRON SHIELD ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH-03-8009-SAR	NONE	1 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-8009-SAR	2 OF 6	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
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DRAWING NO. NUH-03-8009-SAR | SHEET 3 OF 6 | REVISION 1

NUH-03-8009-SAR 4 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH-03-8009-SAR SHEET 4 OF 6 REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH-03-8009-SAR	5 OF 6	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
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DRAWING NO.	SHEET	REVISION
NUH-03-8009-SAR	6 OF 6	1

NUH-03-8009-SAR 6 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® OS197L
ONSITE TRANSFER CASK
OS197L MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH-03-8010-SAR	NONE	1 OF 2

PROPRIETARY AND
SECURITY RELATED INFORMATION
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DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® OS197L
ONSITE TRANSFER CASK SUPPORT SKID SUPPLEMENTAL SHIELDING

DRAWING NO.	SCALE	SHEET
NUH-03-8011-SAR	NONE	1 OF 4

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® OS197L
ONSITE TRANSFER CASK
DECON AREA CASK SHIELDINGASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH-03-8012-SAR	NONE	1 OF 1

All changes on this page are AMD 11

Table W.1-1
Comparison of Key Parameters of NUHOMS® OS197 Versus OS197L TCs

Characteristic	OS197 TC	OS197L TC	Same? (Yes/No) Note No.
Physical Data			
Outside Diameter	85.50"	80.36"	No (1)
Outside Length	207.22"	207.22"	Yes
Cavity Diameter	68"	68"	Yes
Cavity Length	197.75"	197.75"	Yes
Ram Access Penetration Diameter	22"	22"	Yes
Weight, Empty	106,670 lbs (includes cask top cover plate assembly and neutron shield without water)	57,400 lbs (includes cask top cover plate assembly and neutron shield without water)	No (2)
Cask Materials			
Outer Jacket	3/16" thick plate, ASTM A240, Type 304	1/4" thick plate, ASTM A240, Type 304	No (3)
Neutron Shielding	3" of Water in annulus	3" of Water in annulus	Yes
Structural Shell	1-1/2" thick plate, ASME SA-240 Type 304	2.68" thick plate, ASME SA-240 Type 304	No (4)
Gamma Shielding	3.56" thick, ASTM B29 Chemical Copper Lead	No lead shielding	No (4)
Inner Liner	1/2" thick plate, ASME SA-240 Type 304	No separate inner liner (consists of structural shell)	No (4)
Top Cover Assembly	Consists of 3" thick ASME SA-240, Type 304 structural plate with a thin 1/4" thick shell encapsulating a solid Neutron Absorbing Material (NS-3)	Consists of 3" thick ASME SA-240, Type 304 structural plate with a thin 1/4" thick shell encapsulating a solid Neutron Absorbing Material (NS-3)	Yes
Top Flange	ASME SA-182, Type F304N	ASME SA-182, Type F304N	Yes
Upper Lifting Trunnion	ASME SA-564, Grade 630 steel trunnion with sleeve encapsulating a solid Neutron Absorbing Material (NS-3)	Solid monolithic Trunnion made of ASME SA-182, Type FXM-19	No
Lower Support Trunnion	ASME SA-240, Type F304 steel trunnion with sleeve encapsulating a solid Neutron Absorbing Material (NS-3)	Solid monolithic Trunnion made of ASME SA-182, Type F304	No
Canister Rails	ASTM A240 Nitronic 60	ASTM A240 Nitronic 60	Yes
Bottom End Plate	2" thick, ASME SA-240, Type 304	2" thick, ASME SA-240, Type 304	Yes
Bottom Support Ring	ASME SA-182, Type F304N	ASME SA-182, Type F304N	Yes
Ram Access Penetration Ring	ASME SA-182, Type F304N	ASME SA-182, Type F304N	Yes
Cask Payload			
DSC Type	24P, 52B, 61BT, *61BTH Type 1,* 24PHB, 24PT2, *24PTH,* 32PT	61BT *and* 32PT *Only with modified payloads*	*No*
Heat Load	24 kW	*13* kW	*No*

Notes:

1. The diameter of the OS197L TC is smaller, reflecting the reduced radial shielding. The 2.68" thick SS structural shell replaces the combined thickness of ½" of inner liner, 3.50" of lead, and 1.50" of structural shell, a reduction of approximately 5.5" diametrical.
2. The reduced weight of the OS197L TC reflects the reduced radial shielding.
3. The outer panel of the neutron shield is increased in thickness to stiffen the assembly.
4. The reduced shielding is a result of the lead shielding that is eliminated and the combined inner liner and structural shell.

All changes on this page are AMD 11

Table W.1-2
OS197L TC UFSAR Sections Affected
(Part 1 of 3)

Seq	Section/Page	Description	OS197L
1	1.1(3)/1.1-3	Description of TC for transfer of DSC	No Change
2	Figures 1.1-2/1.1-6	NUHOMS® System Components including TC	See Section W.1
3	Figures 1.1-3/1.1-7	NUHOMS® System Components including TC	See Section W.1
4	1.2.3/1.2-3	Description of Operating and Handling Systems including TC	Changes addressed in Section W.1.
5	Table 1.2-2/1.2-8	Key Design Parameters for NUHOMS® System	See Section W.1
6	Table 1.2-3/1.2-9	NUHOMS® System Operations Overview	See Section W.8
7	Section 1.3.2.1/1.3-3	Description of On-Site TC	See Section W.1
8	Section 1.3.2.2/1.3-4	Description of Transfer Equipment (Trailer and Skid)	See Section W.1
9	Table 1.3-1/1.3-10	Components, Structures and Equipment for the Standardized NUHOMS® System	See Section W.1
10	Figure 1.3-6/1.3-18	NUHOMS® On-Site TC	See Section W.1
11	Figure 1.3-8/1.3-20	Cask Support Skid for NUHOMS® System	See Section W.1
12	Figure 1.3-10/1.3-22	NUHOMS® System Operational Overview	See Section W.1
13	2.0	Site Characteristics	No Change
14	3.1.2.1/3.1-4	On-Site Transfer Cask	No change to loading conditions See Section W.1 for OS197L description.
15	Table 3.1-7/3.1-13	NUHOMS® Transfer Equipment Criteria	No Change
16	3.2.5.3/3.2-7	On-Site Transfer Cask Load Combinations and Structural Design Criteria	No Change to load combinations or criteria. See Section W.3 for OS197L structural results
17	Table 3.2-1/3.2-11	Summary of NUHOMS® Component Design Loadings	No Change
18	Table 3.2-7/3.2-20	On-Site Transfer Cask Load Combinations and Service Levels	No Change
19	Table 3.2-11/3.2-25	Structural Design Criteria for On-Site Transfer Cask	No Change
20	Table 3.2-12/3.2-26	Structural Design Criteria for Bolts	No Change
21	3.3.5.2/3.3-31	Radiological Protection-Shielding	See Section W.5
22	Table 3.3-1/3.3-36	NUHOMS® System Components Important To Safety	See Table W.2-1
23	3.4.4.1/3.4-2	Classification of Structures, Components, and Systems- Transfer Cask and Yoke	No Change
24	3.4.4.2/3.4-2	Classification of Structures, Components, and Systems- Other Transfer Equipment	No Change
25	Table 3.4-1/3.4-4	NUHOMS® Major Components and Safety Classification	See Table W.2-1
26	4.2.1/4.2-1	Storage Structures – Structural Specifications	No Change
27	4.2.3.3/4.2-9 *and 4.2-10*	Individual Unit Description - On-Site Transfer Cask	See Sections W.1 and W.3 for trunnion load test.
28	Figure 4.2-10/4.2-21	Composite View of NUHOMS® Transfer Cask-24P	*Not Applicable*
29	Figure 4.2-11/4.2-22	Composite View of NUHOMS® Transfer Cask-52B	*Not Applicable*
30	Figure 4.2-12/4.2-23	NUHOMS® On-Site Transfer Cask with BWR Collar	*Not Applicable*
31	Figure 4.2-15a/4.2-26a	NUHOMS® *OS197L* Transfer Cask Lifting Yoke	*Figure title revised to indicate that this is an alternate lifting yoke*

All changes on this page are AMD 11

Table W.1-2
OS197L TC UFSAR Sections Affected
(Part 2 of 3)

Seq	Section/Page	Description	OS197L
32	4.5/4.5-1	Transfer Cask and Lifting Hardware Repair and Maintenance	No Change
33	4.7.3.2/4.7-5	Individual Unit Descriptions - Transfer Cask	See Section W.1
34	4.7.3.8/4.7-10	Individual Unit Descriptions – Cask Support Skid	See Section W.1
35	4.9/4.9-1	ASME Code Exceptions List for the Transfer Cask	See Section W.3
36	Table 4.9-1/4.9-3	ASME Code Exceptions List for the Transfer Cask	See Section W.3
37	5.0/5.1-1	Operation Systems	See Section W.8
38	6.0	Waste Confinement and Management	No Change
39	7.1/7.1-1	Radiation Protection-design Considerations	See Section W.5
40	7.3.2.2.F/7.3-6	Transfer Cask Surface Dose Rates	See Section W.5
41	Tables 7.3-2 through 7.3-5/7.3-9 through 7.3-14	Shielding Analysis Results	See Section W.5
42	7.4.1/7.4-1	Operational Dose Assessment	See Section W.5
43	Table 7.4-1/7.4-3	NUHOMS® System Operations – Occupational Dose Calculations	See Section W.5
44	8.0	Analysis of Design Events	See: Section W.3 – Structural Section W.4 – Thermal Section W.11 – Accident
45	9.0	Conduct of Operations	No Change
46	*Table 10-2* *Section B 10.5.3.4*	Operating Controls and Limits	*See Section W.12*
47	11.0	Quality Assurance	No Change
48	Appendix A	Details of Shielding Models of the NUHOMS® System	See Section W.5
49	Appendix B	Details of Heat Transfer Analysis of the NUHOMS® System	No Change
50	Appendix C.1	Deleted	No Change
51	Appendix C.2	Transfer Cask Drop Analysis	See Section W.3
52	Appendix C.3	Transfer Cask Side Drop Analysis	See Section W.3
53	Appendix C.4.1	DSC Fatigue Evaluation	No Change
54	Appendix C.4.2	Transfer Cask Fatigue Evaluation	No Change
55	Appendix C.5	Transfer Cask Structural Analysis NRC Question Resolutions	See Section W.3 for DBT events
56	Appendix C.6	References	No Change
57	Appendix D	Review of Concrete Behavior under Sustained elevated Temperature	No Change
58	Appendix E	Drawings	See Section W.1.5
59	Appendix F	NUHOMS® 24P Topical Report – NRC Questions	No Change
60	Appendix G	Deleted	No Change
61	Appendix H	NUHOMS® 24P – Long Cavity DSC Evaluation for Storing PWR fuel without BPRAs	No Change
62	Appendix I	Deleted	No Change
63	Appendix J	NUHOMS® 24P – Long Cavity DSC Evaluation for Storing PWR fuel with BPRAs	No Change

Table W.1-2
OS197L TC UFSAR Sections Affected
(Part 3 of 3)

Seq	Section/Page	Description	OS197L
64	Appendix K	NUHOMS® -61BT evaluation for storage in HSM and transfer in OS197 TC	No Change
65	Appendix L	NUHOMS®-24PT2 evaluation for storage in HSM and transfer in standardized TC and OS197 TC	No Change
66	Appendix M	NUHOMS®-32PT evaluation for storage in HSM and transfer in OS197/OS197H TC	No Change
67	Appendix N	NUHOMS®-24PHB evaluation for storage in HSM and transfer in OS197/OS197H TC	No Change
68	Appendix P	NUHOMS®-24PTH evaluation for storage in HSM and transfer in OS197/OS197H TC/OS197FC	No Change
69	Appendix R	Evaluation of NUHOMS® HSM Model 152	No Change
70	Appendix T	NUHOMS®-61BTH evaluation for storage in HSM and transfer in OS197/OS197H TC/OS197FC	No Change
71	Appendix U	NUHOMS®-32PTH1 evaluation for storage in HSM and transfer in OS200 or OS200FC	No Change
72	Appendix V	Evaluation of NUHOMS® HSM Model 202	No Change

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All changes on this page are AMD 11

Security-Related Information Figure Withheld Under 10 CFR 2.390.

Figure W.1-1
OS197L TC Configuration

Security-Related Information
Figure Withheld Under 10 CFR 2.390.

Figure W.1-2
NUHOMS® OS197L TC System Decontamination Area Shielding

Security-Related Information
Figure Withheld Under 10 CFR 2.390.

Figure W.1-3
OS197L Transfer Equipment Schematic

Figure W.1-4
OS197L TC System on Transfer Trailer with Shielding

W.2 Principal Design Criteria

This section provides the principal design criteria for the NUHOMS® OS197L TC System. The principal design criteria for the NUHOMS® OS197L TC are the same as the NUHOMS® OS197 TC as described in Chapter 3. *However, OS197L TC shall only be used for the handling and transfer of NUHOMS®-61BT or 32PT DSC with a heat load of 13 kW or less.* Section W.2.1 presents a general description of the spent fuel to be stored *in 61BT and 32PT DSC when using OS197L TC.* Section W.2.2 provides the design criteria for environmental conditions and natural phenomena. Section W.2.3 provides a description of the systems which have been designated as important to safety. Section W.2.4 discusses decommissioning considerations. Section W.2.5 summarizes the NUHOMS® OS197L TC design criteria.

All changes on this page are AMD 11

W.2.1 Spent Fuel To Be Stored

When using the OS197L TC for loading and transfer activities, the number of PWR and BWR fuel assemblies authorized for storage in the NUHOMS® 32PT and 61BT DSCs remain unchanged at up to a total of 32 and 61, respectively with the same physical characteristics as those described in Appendix M.2 and K.2 respectively. However, the thermal and radiological characteristics of the spent fuel authorized for storage have been modified as described in the following sections.

W.2.1.1 NUHOMS®-61BT DSC Contents

The physical characteristics of the intact and damaged spent fuel assemblies authorized for storage in the NUHOMS®-61BT DSC are as described in Appendix K, Tables K.2-1 and K.2-2, respectively. However, to minimize the dose consequences when using the OS197L TC for the loading and transfer of the 61BT DSC, the contents allowed shall meet the Heat Load Zoning Configuration requirements of Figure W.2-1 and the DSC heat load shall be limited to 12.0 kW or less. The intact and damaged BWR fuel assembly characteristics allowed for storage in the NUHOMS®-61BT when using the OS197L TC are summarized in Tables W.2-2 and Table W.2-3, respectively. Tables W.2-4 and W.2-5 provide the Fuel Qualification Tables for the Zone 1 and Zone 2 fuel assemblies, respectively.

There is no change in the design configuration of the 61BT DSC or the three 61BT DSC basket types (Type A, B or C) relative to that described in Section K.2.1. Finally, the following criteria described in Section K.2.1 also remain unchanged:

- *The maximum fuel cladding temperatures allowed for normal, off-normal, and accident conditions, and*
- *The maximum DSC internal pressures for normal, off-normal, and accident conditions.*

W.2.1.2 NUHOMS®-32PT DSC Contents

The physical characteristics of the intact spent fuel assemblies, with or without control components (CCs), authorized for storage in the NUHOMS®-32PT DSC are as described in Appendix M, Tables M.2-1. However, to minimize the dose consequences when using the

OS197L TC for the loading and transfer of the 32PT DSC, the contents allowed shall meet the Heat Load Zoning Configuration requirements of Figure W.2-2 and the DSC heat load shall be limited to 13.0 kW or less. The intact PWR fuel assembly characteristics allowed for storage in the NUHOMS®-32PT when using the OS197L TC are summarized in Tables W.2-6. Tables W.2-7 and W.2-8 provide the Fuel Qualification Tables for the Zone 1 and Zone 2 fuel assemblies, respectively.

When using the OS197L TC, all four 32PT DSC design configurations (32PT-S100, 32PT-L100, 32PT-S-125 and 32PT-L-125) are allowed. There is no change in these 4 DSC design configurations or in the four DSC basket configurations (Type A, B, C or D) relative to those described in Section M.2.1. The minimum boron-10 content for the poison plates is unchanged at 0.0070 g/cm^2. The number of reconstituted fuel assemblies allowed also remains unchanged.

Finally, the following criteria described in Section M.2.1 also remain unchanged:

- *The maximum fuel cladding temperatures allowed for normal, off-normal and accident conditions, and*
- *The maximum DSC internal pressures for normal, off-normal and accident conditions.*

All changes on this page are AMD 11

W.2.1.3 General Operating Functions

No change. *The content presented in Section 3.1.2 remains applicable and is not affected by the use of the OS197L transfer cask. Additional operational features applicable to the OS197L are presented in Section W.1.2.2.*

W.2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The NUHOMS® OS197L TC is *in general* handled and utilized in the same manner as the existing NUHOMS® OS197 TC System. *Differences in the operation/handling of the OS197L TC include:*

- *Increased use of plant ALARA measures such as remote monitoring devices to keep exposures ALARA due to the high dose rates on the bare OS197L TC during lifts from the fuel pool to the decontamination area and from the decontamination area to the transfer trailer,*
- *Placement of the bare OS197L TC into the decontamination area shield and placement of "shield bell," and*
- *Placement of the bare OS197L TC on the transfer trailer with supplemental shielding.*

The above differences are described in detail in Chapter W.8.

The environmental conditions, natural phenomena and design criteria are the same as described for the NUHOMS® OS197 TC in Chapter 3.

All changes on this page are AMD 11

W.2.2.1 Tornado Wind and Tornado Missiles

No change. *The tornado wind and tornado missiles criteria presented in Section 3.2.1 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.2 Water Level (Flood) Design

No change. *The flood design criteria presented in Section 3.2.2 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.3 Seismic Design

No change. *The seismic design criteria presented in Section 3.2.3 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.4 Snow and Ice Loading

No change. *The snow and ice loading criteria presented in Section 3.2.4 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.5 Combined Load Criteria

No change. *The loads, load combinations, and design criteria for the OS197 summarized in Tables 3.2-1, 3.2-7, 3.2-11, and 3.2-12 remain applicable and are not affected by the use of the OS197L transfer cask. Additional design criteria applicable to the OS197L are presented in Section W.3.2.*

W.2.3 Safety Protection Systems

W.2.3.1 General

The discussion presented in Section 3.3.1 is applicable, except for Table 3.3-1 which is replaced with Table W.2-1.

Table W.2-1 provides the safety classification of the OS197L TC system components.

W.2.3.2 Protection By Multiple Confinement Barriers and Systems

No change. *The content presented in Section 3.3.2 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.2.3.3 Protection By Equipment and Instrumentation Selection

No change. *The content presented in Section 3.3.3 remains applicable and is not affected by the use of the OS197L transfer cask.*

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W.2.3.4 Nuclear Criticality Safety

The content presented in Section 3.3.4 remains applicable and is not affected by the use of the OS197L transfer cask. As referenced in Sections 3.3.4.4 and 3.3.4.6, the criticality evaluations for the 61BT, and 32PT DSCs are contained in Appendices K and M, respectively.

W.2.3.5 Radiological Protection

The bare OS197L TC provides less shielding than the OS197 TC system. The reduced shielding of the bare TC results in higher dose rates on and around the TC when being lifted from the fuel pool to the decontamination area and from the decontamination area to the transfer trailer. To mitigate the effect of these high dose rates on occupational workers, these operations are done remotely as described in Chapter W.8. In addition, when the TC is in the decontamination area and on the transfer trailer, supplemental shielding is used to reduce the dose rates down to those commensurate with the OS197 TC System. Therefore, with the use of remote crane handling operation in combination with the decontamination area and skid shielding features of the OS197L TC, the occupational workers and members of the public are protected against direct radiation and releases of radioactive material.

W.2.3.6 Fire and Explosion Protection

No change. *The fire and explosion protection discussion presented in Section 3.3.6 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.2.4 Decommissioning Considerations

No change. *The content presented in Section 3.5 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.2.5 Summary of NUHOMS® OS197L TC Design Criteria

The principal design criteria for the NUHOMS® OS197L TC are the same as those presented for the NUHOMS® OS197 TC in Chapter 3. The NUHOMS® OS197L TC is designed to handle a *61BT or 32PT* DSC loaded with *BWR* or *PWR* fuel assemblies, *respectively, as described in this appendix.*

All changes on this page are AMD 11

Table W.2-1
OS197L TC System Components and Safety Classification

OS197L TC System Components	Safety Classification
Onsite Transfer Cask	
– Structural Shell and Cover Plates	Important to Safety(1)
– Upper and Lower Trunnions	Important to Safety(1)
– Decontamination Area Shield	Important to Safety(1)
– *Cask support skid supplemental shielding*	Important to Safety(1)
Transfer Equipment	
– Cask Lifting Yoke	Safety Related(2)
– Transfer Trailer/Skid	Not Important to Safety(1)
– Ram Assembly	Not Important to Safety(1)
– Dry Film Lubricant	Not Important to Safety(1)

Notes:

(1) Structures, systems and components "important to safety" are defined in 10CFR 72.3 as those features of the ISFSI whose function is (1) to maintain the conditions required to store spent fuel safety, (2) to prevent damage to the spent fuel container during handling and storage, or (3) to provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

(2) Yoke and rigid or sling lifting members are classified as "Safety Related" in accordance with 10 CFR 50.

Table W.2-2
Intact BWR Fuel Assembly Characteristics (NUHOMS®-61BT DSC)

PHYSICAL PARAMETERS:	
Fuel Design:	*7x7, 8x8, 9x9, or 10x10 intact BWR fuel assemblies manufactured by General Electric or Exxon/ANF or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Appendix K, 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*
Maximum Assembly length (Unirradiated)	*176.2 in*
Nominal Assembly Width (excluding channels)	*5.44 in*
Maximum Assembly Weight	*705 lbs*
Thermal/Radiological Parameters:	
Maximum Initial Enrichment:	*See Appendix K, Table K.2-1*
Fuel Burnup, Initial Assembly Average Enrichment and Cooling Time:	*Per Table W.2-4, Table W.2-5 and Figure W.2-1.*
Maximum Initial Uranium Content:	*198 kg/assembly*
Maximum Decay Heat:	*300 W/assembly(1)*

Note:

(1) For FANP9 9x9-2 fuel assemblies, the maximum decay heat is limited to 0.21 kW/assembly.

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Table W.2-3

Damaged BWR Fuel Assemblies Characteristics (NUHOMS®-61BT DSC)

PHYSICAL PARAMETERS:	
Fuel Design:	*7x7, 8x8 BWR damaged fuel assemblies manufactured by General Electric or Exxon/ANF or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Appendix K, 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®-61BT Canister.*
Channels:	*Fuel may be stored with or without fuel channels.*
Maximum Assembly Length (unirradiated)	*176.2 in*
Nominal Assembly Width (excluding channels)	*5.44 in*
Maximum Assembly Weight	*705 lbs*
THERMAL/RADIOLOGICAL PARAMETERS:	
Maximum Initial Lattice Average Enrichment:	*4.0 wt. % U-235*
Fuel Burnup, Initial Bundle Average Enrichment, and Cooling Time:	*Per Table W.2-4, Table W.2-5 and Figure W.2-1.*
Maximum Pellet Enrichment:	*4.4 wt. % U-235*
Maximum Initial Uranium Content:	*198 kg/assembly*
Maximum Decay Heat:	*300 W/assembly*

Table W.2-4
Fuel Qualification Table for 0.3 kW BWR FAs in Zone 1 of a NUHOMS®-61BT DSC Contained in an OS197L TC

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

BU GWd/MTU	Assembly Average Initial Enrichment (wt. % U-235)																													
	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4
10	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4
15	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4
20	5	5	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	4	4	4
28					6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30					7	7	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
32					8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	6	6	6	6	6	6	6	6	6
34					9	9	9	9	9	9	8	8	8	8	8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7	7
36		Not Analyzed Domain			11	11	11	10	10	10	10	10	9	9	9	9	9	9	9	9	8	8	8	8	8	8	8	8	8	8	8
38					14	13	13	12	12	12	12	11	11	11	11	11	10	10	10	10	10	10	9	9	9	9	9	9	9	9	9
39					15	14	14	14	13	13	13	12	12	12	12	11	11	11	11	11	10	10	10	10	10	10	10	9	9	9	9
40					16	16	15	15	15	14	14	14	13	13	13	12	12	12	12	12	11	11	11	11	11	10	10	10	10	10	10

Notes for Tables W.2-4 and W.2-5:

- *BU = Assembly average burnup.*
- *Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with an initial enrichment less than 1.4 and greater than 4.4 wt. % U-235 is unacceptable for storage.*
- *Fuel with a burnup greater than 40 GWd/MTU is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 4 years cooling.*
- *Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 39.5 GWd/MTU is acceptable for storage after a eleven-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 40 GWd/MTU (rounding up) on the qualification table.*

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Table W.2-5
Fuel Qualification Table for 0.17 kW BWR FAs in Zone 2 of a NUHOMS®-61BT DSC Contained in an OS197L TC

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

BU GWd/MTU	Assembly Average Initial Enrichment (wt. % U-235)																													
	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4
10	21.5	20.5	20.5	20.5	20.5	19.5	19.5	19.5	19.5	19.5	19.5	18.5	18.5	18.5	18.5	18.5	18.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
11	22.0	22.0	22.0	22.0	21.0	21.0	21.0	21.0	21.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0
12	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0	22.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0
13	24.0	24.0	24.0	23.0	23.0	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0	22.0	22.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0
14	25.0	25.0	25.0	24.0	24.0	24.0	24.0	24.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0
15	26.0	26.0	25.0	25.0	25.0	25.0	25.0	25.0	24.0	24.0	24.0	24.0	24.0	24.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0
16	27.0	26.0	26.0	26.0	26.0	26.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0
17	27.0	27.0	27.0	27.0	27.0	26.0	26.0	26.0	26.0	26.0	26.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	23.0
18	28.0	28.0	27.0	27.0	27.0	27.0	27.0	27.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0
19	28.0	28.0	28.0	28.0	28.0	28.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0
20	29.0	29.0	29.0	29.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0
21	30.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	26.0	26.0	26.0	26.0
22	30.0	30.0	30.0	30.0	30.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	28.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0
23	30.5	30.5	30.5	30.5	30.5	30.5	30.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	27.5	27.5	27.5	27.5
24	31.5	31.5	31.5	31.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5
25	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5
26					31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	29.5	29.5	29.5	29.5	29.5
27					32.5	32.5	32.5	32.5	32.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5
28					32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	30.5
29					33.5	33.5	33.5	33.5	33.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5
30					33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5
31					34.5	34.5	34.5	34.5	34.5	34.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	32.5	32.5	32.5	32.5	32.5
32	Not Analyzed Domain				34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5
33					35.5	35.5	35.5	35.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	33.5	33.5	33.5	33.5	33.5
34					35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5
35					35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	34.5	34.5	34.5	34.5	34.5	34.5
36					36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5
37					36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5
38					37.5	37.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5
39					37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5
40					37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5

Note: The explanatory notes and limitations provided for Table W.2-4 are also applicable to this table.

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Table W.2-6
Intact PWR Fuel Assembly Characteristics (NUHOMS®-32PT DSC)

PHYSICAL PARAMETERS: *Fuel Class*	*Only intact (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies or equivalent reload fuel manufactured by other vendors that are enveloped by the fuel assembly design characteristics listed in Appendix M, Table M.2-2.*
Reconstituted Fuel Assemblies	*≤32 assemblies per DSC with up to 56 stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods per assembly.*
Fuel Cladding Material	*Zircaloy*
Fuel Damage	*Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact PWR Fuel."*
Control Components (CCs)	• *Up to 32 CCs are authorized for storage in 32PT DSC.* • *Authorized CCs include Burnable poison Rod Assemblies (BPRAs), Thimble Plug Assemblies, (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources.* • *Design basis thermal and radiological characteristics for the CCs are listed in Appendix M, Table M.2-2a.*
Maximum Assembly plus CC Weight	*-1365 lbs for 32PT-S100 & 32PT-L100 DSC System* *-1682 lbs for 32PT-S125 & 32PT-L125 DSC System.*
CC Damage	*CCs with cladding failures are acceptable for loading.*
THERMAL/RADIOLOGICAL PARAMETERS: *Fuel Burnup and Cooling Time with or without CCs*	*Per Table W.2-7, Table W.2-8 and Figure W.2-2.*
Maximum Planar Average Initial Fuel Enrichment	*Appendix M, Table M.2-3 and Figure M.2-4 or Figure M.2-5 or Figure M.2-6.*

Table W.2-7
Fuel Qualification Table for 0.6 kW PWR FAs in Zone 1 of a NUHOMS®-32PT DSC Contained in an OS197L TC (Fuel with or without CCs)

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

BU GWd/ MTU	Assembly Average Initial Enrichment (wt. % U-235)																																			
	1.1	1.2	1.4	1.6	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
6	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
8	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
10	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
15	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
20	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
25	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.0	5.0	5.0	5.0
28	8.0	8.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
30	9.0	9.0	9.0	9.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
32	10.5	10.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
34	12.0	12.0	12.0	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.0	9.0	9.0	9.0	9.0	9.0
36	14.5	14.5	14.0	14.0	13.5	13.5	13.0	13.0	13.0	13.0	13.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0
38	17.5	17.5	16.5	16.5	16.5	16.0	16.0	15.5	15.5	15.0	15.0	15.0	15.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0
39	19.5	19.0	18.5	18.0	17.0	16.5	16.5	16.5	16.5	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0
40	20.5	20.0	20.0	19.0	19.0	18.5	18.5	18.5	18.0	18.0	18.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	15.0	15.0	15.0	15.0
41	22.5	21.5	21.0	21.0	20.0	20.0	19.5	19.5	19.5	19.0	19.0	19.0	19.0	19.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	16.0
42	24.0	22.5	22.5	21.5	21.5	21.5	21.0	21.0	21.0	21.0	21.0	20.0	20.0	20.0	20.0	20.0	20.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0
43	25.0	24.5	24.5	23.5	23.5	23.0	22.0	22.0	22.0	21.5	21.5	21.5	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	19.0	19.0	19.0	19.0	19.0	19.0
44	26.5	26.5	25.0	25.0	24.0	24.0	24.0	24.0	23.5	23.5	23.5	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.0	20.0
45	27.5	27.5	27.0	26.0	26.0	25.0	25.0	25.0	25.0	24.5	24.5	24.5	24.0	24.0	24.0	24.0	24.0	24.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0

Note: The page that follows Table W.2-8 provides the explanatory notes and limitations regarding the use of this table.

All changes on this page are AMD 11

Table W.2-8

Fuel Qualification Table for 0.4 kW PWR FAs in Zone 2 of a NUHOMS®-32PT DSC Contained in an OS197L TC (Fuel with or without CCs)

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

BU GWd/MTU	Assembly Average Initial Enrichment (wt. % U-235)																																							
	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
11	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
12	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
13	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
14	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
15	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
16	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
17	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
18	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
19	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
20	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
21	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
22	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
23	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
24	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
25	18.5	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
26	19.0	18.5	18.5	18.0	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
27	19.5	19.0	19.0	19.0	18.5	18.5	18.5	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
28	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
29	19.5	19.5	19.5	19.5	19.5	19.5	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
30	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5
31	23.5	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	21.5	21.5	21.5	21.5	21.5	21.5	21.5	21.5	21.5
32	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.0
33	28.0	28.0	27.5	27.5	27.5	27.5	27.5	27.0	27.0	27.0	27.0	27.0	27.0	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5	26.5	25.5	25.5	25.5	25.5	25.5	25.5	25.5	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0
34	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0
35	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	30.0	30.0	30.0	30.0	30.0	30.0	30.0	30.0	30.0	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0	29.0
36	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	30.5	30.5	30.5	30.5	30.5	30.5	30.5	30.0
37	34.5	34.5	34.5	34.5	34.5	34.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.0	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	31.5	31.5	31.5	31.5	31.5	31.5
38	36.0	36.0	36.0	35.5	35.5	35.5	35.0	35.0	35.0	35.0	35.0	35.0	35.0	35.0	35.0	35.0	35.0	35.0	35.0	35.0	34.5	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	33.0	33.0	33.0
39	37.5	37.5	37.5	37.5	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	35.0
40	39.5	39.0	39.0	39.0	39.0	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5
41	41.0	41.0	40.5	40.5	40.5	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0
42	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5
43	44.0	44.0	43.5	43.5	43.5	43.5	43.5	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	43.0	42.0	42.0	42.0	42.0	42.0	42.0	42.0	42.0
44	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0
45	47.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	46.5	45.5	45.5	45.5	45.5	45.5	45.5	45.5

Explanatory notes and limitations regarding the use of this table follow.

All changes on this page are AMD 11

Notes for Tables W.2-7 and W.2-8:

- *BU = Assembly Average burnup.*
- *Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *For fuel assemblies with CCs, increase the indicated cooling time by 1.5 years. This applies to 0.6 kW FAs only.*
- *For fuel assemblies reconstituted with up to 10 stainless steel rods, increase the indicated cooling time by 1.5 years. If more than 10 stainless steel rods are present, increase the indicated cooling time by 6 years.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with an initial enrichment less than 1.1 and greater than 5.0 wt. % U-235 is unacceptable for storage.*
- *Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.*

Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a nineteen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

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			2	2	2			
	2	2	2	2	2	2	2	
	2	2	2	1	2	2	2	
2	2	2	1	1	1	2	2	2
2	2	1	1	1	1	1	2	2
2	2	2	1	1	1	2	2	2
	2	2	2	1	2	2	2	
	2	2	2	2	2	2	2	
			2	2	2			

Heat Zone Level	**Zone 1**	**Zone 2**
Max. Decay Heat/FA (kW)	*0.3*	*0.17*
Number of FAs/Zone	*13*	*48*
Max. Decay Heat/Zone (kW)	*3.9*	*8.2*
Max. Decay Heat/DSC (kW)	***12.0***	

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Figure W.2-1
Heat Load Zone Configuration for the 61BT DSC

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	2	2	2	2	
2	2	1	1	2	2
2	1	1	1	1	2
2	1	1	1	1	2
2	2	1	1	2	2
	2	2	2	2	

Heat Zone Level	***Zone 1***	***Zone 2***
Max. Decay Heat/FA (kW)	*0.6*	*0.4*
Number of FAs/Zone	*12*	*20*
Max. Decay Heat/Zone (kW)	*7.2*	*8.0*
Max. Decay Heat/DSC (kW)	***13.0*** (1)	

(1) *Maximum decay heat load allowed in the OS197L TC.*

Figure W.2-2
Heat Load Zone Configuration for the 32PT DSC

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W.3 Structural Evaluation

This section describes the structural evaluation of the NUHOMS® OS197L Transfer Cask (TC). The OS197L TC is a modified version of the OS197/OS197H TCs (henceforth referred *to* as the OS197 TC) designed to *handle a loaded weight of up to 125* tons. The OS197L TC may *only be used for the loading and transfer of 61BT or 32PT DSCs with a heat load of 13kW or less as described in Chapter W.2, Figures W.2-1 and W.2-2.* The structural evaluation for the OS197L TC is based on the OS197 TC evaluations documented in Chapter 8 and in Appendices K *and* M for payloads associated with the 61BT *and* 32PT, respectively. The additional evaluations provided in this section address specific design differences between the OS197L TC and the OS197 TC.

The OS197L TC requires use of supplemental shielding when the transfer cask is in the decontamination area during handling operations and when the transfer cask is placed on the transfer trailer skid. The structural evaluation of the supplemental shielding is summarized in Section W.3.9.

W.3.1 OS197L TC Description

The specific design differences in the OS197L TC relative to OS197 TC are summarized below:

- The 1.5" thick structural shell and the 0.5" thick inner liner (both SA-240 stainless steel) are replaced with a single thicker 2.68" thick shell of the same material. This represents an increase in the TC shell structural capacity relative to the OS197 TC.
- The encapsulated 3.56" thick lead thickness in the OS197 TC is eliminated to achieve the desired weight reduction.
- A neutron shield assembly is provided with the inner and outer shells made from ¼" thick plate material instead of a neutron shield assembly that is integral to the structural shell on the inside and a 3/16" thick outer shell. The neutron shield materials (type 304), total annulus water thickness of 3" and the configuration of the internal stiffening elements remain essentially unchanged.
- The two-piece upper trunnions assemblies made from SA-564 Type 630 steel trunnion and welded into a forged Type 304 steel trunnion sleeve with encapsulated NS-3 for the OS197 TC are replaced with one solid trunnion design made from SA-182 Type FXM-19 stainless steel. This modified trunnion design results in a stronger trunnion as it eliminates the SA564, Type 630 to SA 240, Type 304 weld.
- The two-piece lower trunnions made from Type 304 stainless with encapsulated NS-3 are replaced with solid Type 304 forgings.

Specific evaluations are performed to address the modified OS197L TC trunnion configuration. The evaluations also address the effect on local shell stresses. Thermal stresses of the cask are also evaluated. All other structural analyses for the OS197 TC bound the OS197L TC because the cask structural shell capacity of the OS197L TC is higher than that provided by the OS197 and the top and bottom forging assemblies are unchanged.

All changes on this page are AMD 11

W.3.2 Design Criteria

The structural design criteria for the OS197L TC are the same as that applicable to the OS197 TC as summarized in Chapter 3. Similar to the OS197 TC, the OS197L TC is designed to meet the stress allowables of the ASME Code [3.2] Subsection NC for Class 2 components. The OS197 TC criteria summarized in Table 3.2-1 (component design loadings, as applicable), Table 3.2-7 (load combinations), Table 3.2-11 (stress criteria) and Table 3.2-12 (bolts design criteria) are applicable to the OS197L TC. The OS197 TC ASME Code exceptions described in Table 4.9-1 are also applicable to the OS197L TC.

The test load criteria for the upper trunnions of the OS197L TC are the same as described in Section 4.2.3.3, except that the test load is conservatively equal to 300% of the design load (instead of 150% for the OS197 TC).

The supplemental shielding is designed in accordance with AISC Code, Manual of Steel Construction, Ninth Edition [3.4].

W.3.3 OS197L TC Weight

The *empty and loaded* weights of the OS197L TC *are* presented in Table W.3-1. The total *empty* weight of the cask *(without any payload but including neutron shield full of water and the cask lid)* is approximately 62,000 lb. This compares with the corresponding weight of 111,250 lb for the OS197 TC.

The OS197L TC weights as described in Table W.3-1 are to be used in conjunction with the payload weights for the various DSCs as described in the applicable sections in Appendices K.3 *and* M.3. Each specific user is to evaluate the total under-the-hook lift weights against plant specific crane capacity limits in accordance with the requirements of 10CFR72.212.

W.3.4 Mechanical Properties of Materials

The materials properties for the OS197L TC are specified in Section 8.1, Table 8.1-3.

W.3.5 General Standards for Casks

The OS197L is fabricated using the same materials as the OS197 TC. Thus, there are no changes to the documentation in Appendices K.3 *and* M.3 relative to chemical and galvanic reactions.

The evaluation of the OS197L TC is based on critical lift weights of 250,000 lb *(125 tons)*.

The thermal analysis of the OS197L along with a summary of the effect on pressures and temperatures is described in Chapter W.4.

As reported in Section 8.2.5.1C, the g loads for the OS197 TC were determined to be 59 g for the end drop, 49 g for the side drop and 25 g for a corner drop. Based on these accelerations, bounding accelerations of 75g for the horizontal (side) and vertical drops and 25g for the corner drop were used for the OS197 TC drop evaluations. The OS197 TC evaluations are documented in Chapter 8. Using the same methodology as that described in Section 8.2.5.1C for the OS197 TC, the equivalent loads for the OS197L TC are 75 g for an end drop, 61 g for a side drop and 25 g for a corner drop. Therefore, the 75g accident drop evaluation results for the side and end drops and the 25g evaluations for the corner drop performed for the OS197 TC and reported in Section 8.2 remain bounding and are applicable to the OS197L TC. These g-loads are conservative with respect to shell stresses since the thicker OS197L TC shell has a higher load capacity than the OS197 TC shell configuration. Hence, all the cask accident drop results reported in Section 8.2 *(8.2.5.2)*, and Appendices K.3 *(K.3.7.5) and* M.3 *(M.3.7.5)* remain bounding and, thus, are not affected.

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As with the OS197 TC and as documented in UFSAR Section 8.2.5.3, complete loss of neutron shield is postulated for the OS197L TC as a consequence of the drop accident event. This event is evaluated in Sections W.4 and W.5 for thermal and shielding effects, respectively. The post accident recovery actions as discussed in Section 8.2.5.4 are applicable to the OS197L TC.

W.3.8 Effect of OS197L Temperatures on DSC Shell and Basket Components

Based on the thermal analysis documented in Chapter W.4, the maximum temperature*s* applicable to the *61BT and 32PT* DSC shell and basket components *for transfer in the OS197L TC (normal, off-normal, and accident conditions) are much lower than the corresponding temperatures for transfer in the OS197 TC presented in Appendices K.4 and M.4. Thus, there is no adverse affect on* the material properties or the allowables used for the evaluations of *61BT and 32PT* DSCs as documented in Appendix K, Chapter K.3 *and* Appendix M, Chapter M.3, *respectively*. Furthermore, the accident pressures used in the structural evaluations bound those calculated in Chapter W.4.

W.3.9 *Structural Evaluation of OS197L TC Supplemental Shielding Components*

The OS197L TC supplemental shielding when the transfer cask is placed in the decontamination area consists of an upper cask shield (shielding bell) and a lower cask shield (shielding sleeve) made from carbon steel plate 6" thick. The supplemental shielding when the transfer cask is mounted on the trailer's skid consists of massive (2.5" thick combined with additional 3" thick) carbon steel plates which are integral (i.e., welded) to the skid or bolted to each other. The supplemental shielding components are evaluated using the stress allowable criteria of AISC Code, Manual of Steel Construction, Ninth Edition, summarized in Table W.3-5. The decontamination area shielding is evaluated for deadweight, lifting loads, and seismic loads. The skid-mounted supplemental shielding is evaluated for deadweight and conservatively defined (2g) handling loads. Conservatively evaluated bounding stresses are summarized in Table W.3-6 and Table W.3-7.

The above evaluations assume that the supplemental shielding components are handled using a single failure proof crane when these components are handled inside the fuel/reactor building. If a single failure proof crane is not used, the licensee is to evaluate the accidental drop of these *shielding components under the provisions of 10 CFR 50.59 and 10 CFR 72.212, and evaluate consequences of the accident drops.*

The only component that may be handled outside the fuel/reactor building is the top (outer) skid shielding. The evaluation of accidental drop of this component is provided in Section W.11.1.5.

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W.3.10 References

3.1 *Not used.*

3.2 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1983 Edition with Winter 1985 Addenda.

3.3 American National Standard, "For Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More," ANSI N14.6-1986, American National Standards Institute, Inc. New York, New York (1993)

3.4 *AISC Manual of Steel Construction, Ninth Edition, 1989.*

Table W.3-1
Summary of OS197L TC Weights

Item	Weight Configuration	
	Wet (lbs)	Dry (lbs)
OS197L Cask Body with Neutron Shield Assembly	52,236	52,236
Neutron Shield Water	4,606	4,606
Top Cask Lid	-	5,147
Water in DSC and DSC/TC Annulus	12,708	-
Bounding Nominal Payload (1)	99,133	102,222
Loaded OS197L TC Total Nominal Weight (2)	168,683	164,211
	(84.3 tons)	(82.1 tons)

Notes:

(1) Bounding of 32PT and 61BT DSCs

(2) The weight does not include the decontamination area shielding or the support skid supplemental shielding

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Table W.3-5
OS197L Supplemental Shielding Allowable Stress Criteria

Item/Component	***Stress Category***	***Allowable Stress***
Steel plates	*Tension, compression or bending*	*0.60 S_y*
Steel plates	*Shear stress*	*0.40 S_y*
Steel plates	*Bearing*	*0.90 S_y*
Welds (groove or fillet)	*Tension, compression*	*Lesser of 0.6 S_y (base metal), or 0.30 S_u(weld metal)*
Welds (groove or fillet)	*Shear*	*0.30 S_u*
Bolts	*Tension*	*0.33F_u*
Bolts	*Shear*	*0.17F_u*

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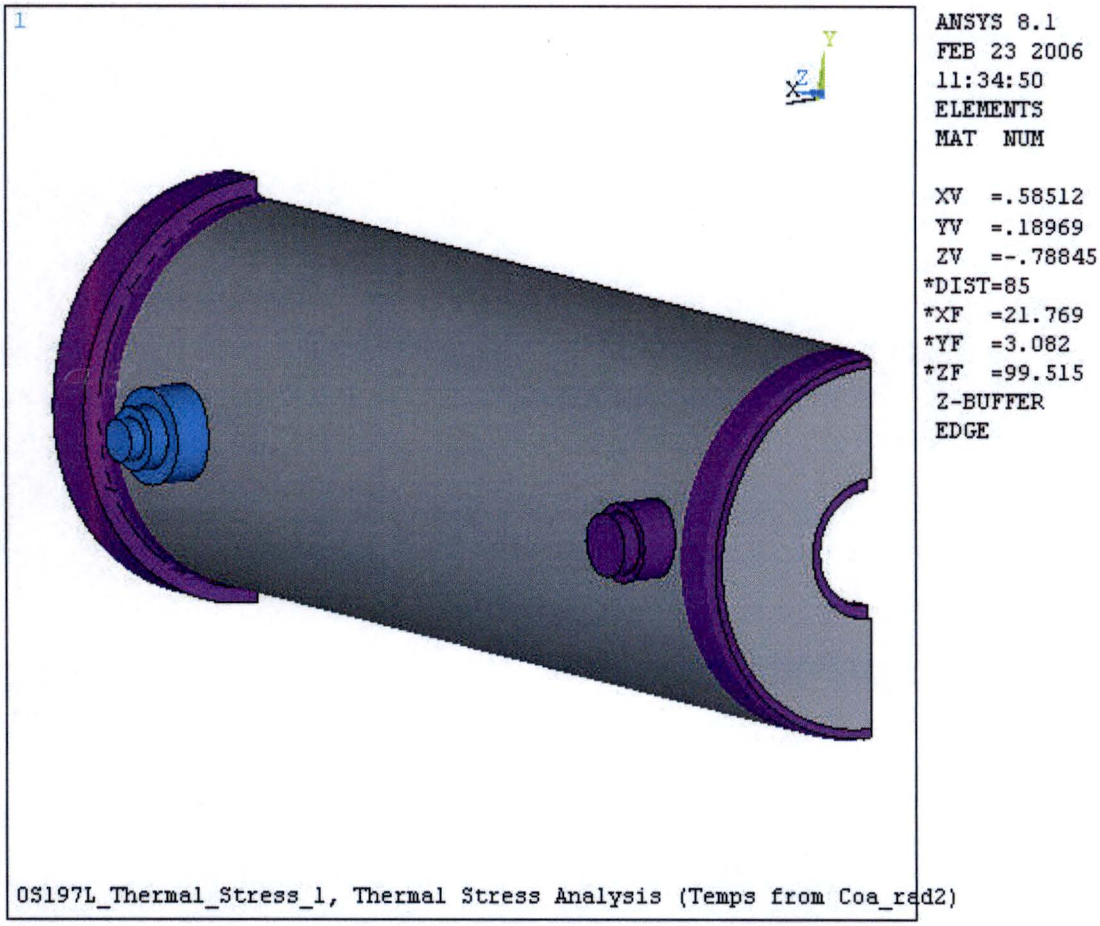
Table W.3-6
Summary of Results for Decon Area Cask Supplemental Shielding

Component / Evaluation Description	Maximum Stress (ksi)	Allowable Stress (ksi)	Stress Ratio
Decon Area Cask Shielding Deadweight Axial Stress	*0.29*	*21.18*	*0.014*
Decon Area Cask Shielding Seismic Axial Stress	*8.02*	*28.23*	*0.284*

Table W.3-7
Summary of Results for Cask Skid Supplemental Shielding for 2g Handling Loads

Component / Evaluation Description	Maximum Stress (ksi)	Allowable Stress (ksi)	Stress Ratio
Skid Shield Plate Bending Stress	*6.85*	*21.18*	*0.323*
Skid Shield Weld Shear Stress	*8.37*	*21.00*	*0.399*
Skid Shielding Bolt Shear Stress	*10.01*	*21.25*	*0.471*
Skid Tie-Down Plate Bending Stress	*8.93*	*21.18*	*0.422*
Skid Tie-Down Plate Weld Shear Stress	*11.17*	*21.00*	*0.532*
Skid Tie-Down Connection Bolt Shear Stress	*8.37*	*21.25*	*0.394*

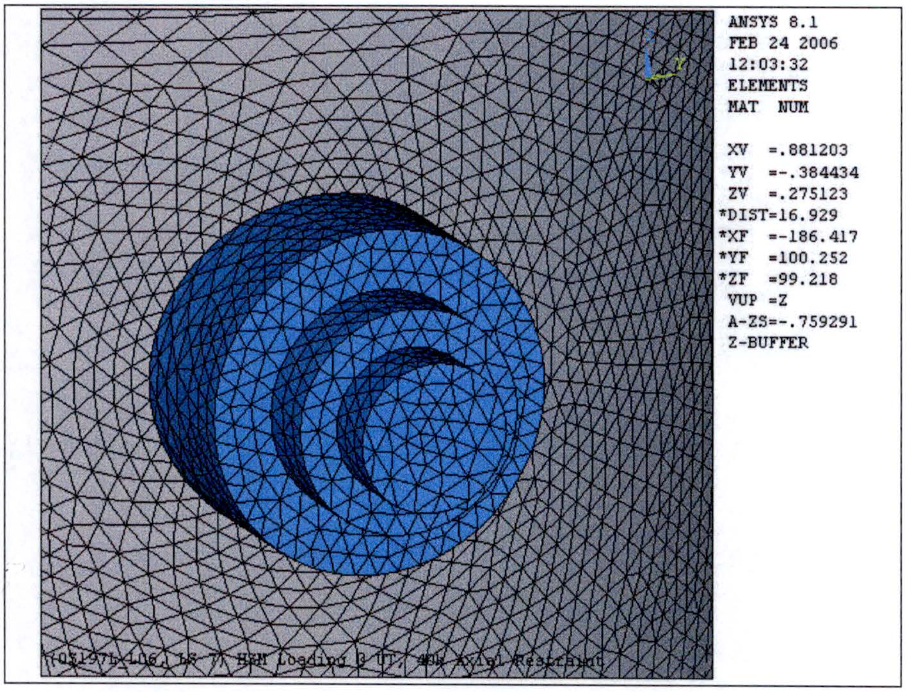


Note:

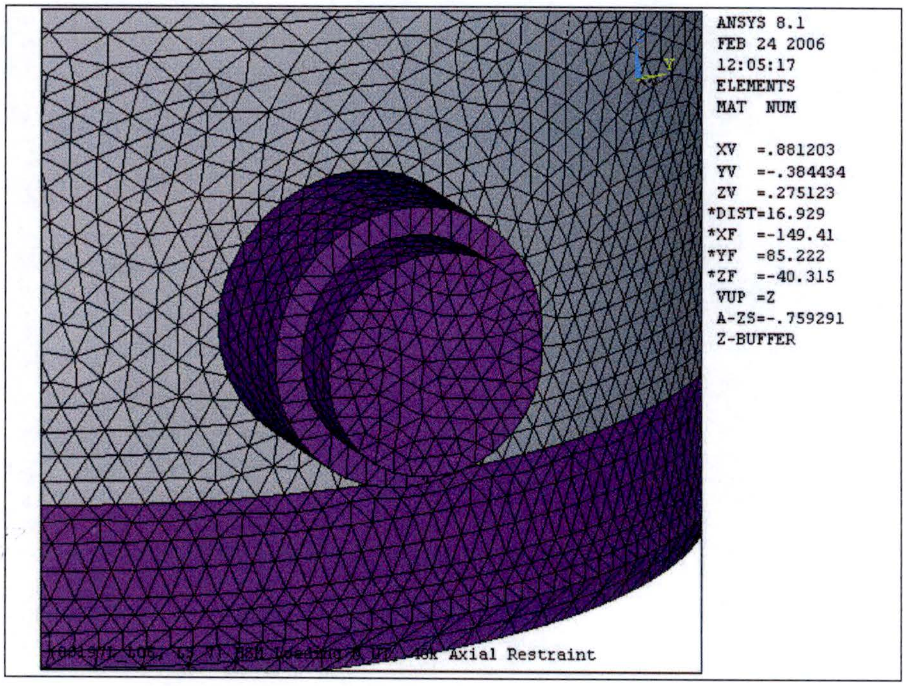
Material properties were assigned as follows:

Purple = SA-182 Type F304N (forgings)
Gray = SA-240 Type 304
Blue= SA-182 Type FXM-19 (Type XM-19 Forging)

Figure W.3-1
OS197L TC ANSYS Stress Analysis Model



Upper Trunnion



Lower Trunnion

Figure W.3-2
OS197L TC ANSYS Analysis Model – Upper and Lower Trunnions Detail

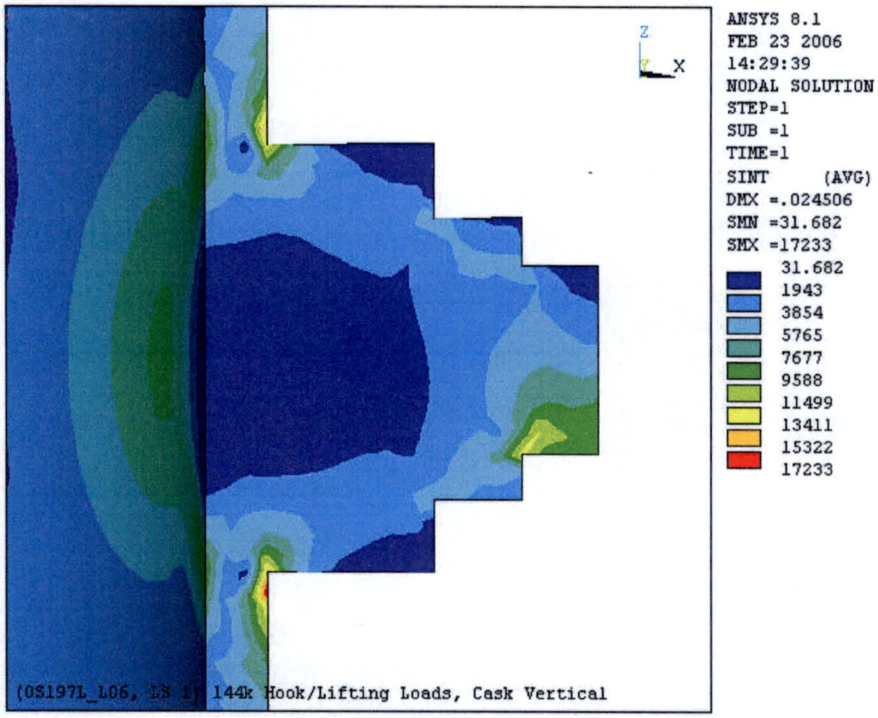


Figure W.3-3
ANSYS Model Stress Analysis Results – Upper Trunnion Region

W.4 Thermal Evaluation

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W.4.1 Discussion

This chapter documents the thermal evaluation of the OS197L TC for the loading and transfer of the *NUHOMS® 61BT and NUHOMS® 32PT DSCs with a heat load of 13 kW or less. The authorized contents of the 61BT and 32PT DSC are as described in Appendix W.2.*

The OS197L TC is a modified version of the OS197/OS197H TCs (henceforth referred to as OS197 TC). From a thermal analyses perspective, the following relevant modifications are implemented in the OS197L TC relative to the OS197 TCs:

- The 1.5" thick structural shell, the encapsulated 3.56" thick lead and the 0.5" thick inner liner (both SA-240, Type 304 stainless steel) in the OS197 are replaced with a single 2.68" thick shell made of SA-240, Type 304 stainless steel material.

- The neutron shield assembly that is integral to the structural shell on the inside and includes a 3/16" thick outer shell in the OS197 TC is replaced with a neutron shield assembly consisting of inner and outer shells made from ¼" thick plate material in the OS197L TC. The neutron shield materials, total water annulus thickness of 3" and the configuration of the internal stiffening elements remain essentially unchanged.

- Supplemental shielding is used around the OS197L TC as part of the OS197L TC system, when the TC is in the vertical orientation in the decontamination area and when the TC is in the horizontal orientation on the transfer trailer/skid.

- The supplemental OS197L TC shielding system in the decontamination area consists of a two-part assembly, with 6" thick cylindrical shaped upper and lower shields made from A-36 steel with rectangular openings at the top and at the bottom that allow free convection boundary layer development along the DSC shell. The decontamination area supplemental shielding is shown in Figure W.1-2.

- The supplemental OS197L TC shielding system used on the transfer skid consists of a series of plates that are attached to the sides and ends of the transfer skid. Two upper sections fit like a clamshell over the cask and skid after the cask is placed on the transfer skid. Clearances provided at the support legs of the skid and other openings at the ends of the skid permit cooling airflow to enter the enclosure and pass around the enclosed cask and exit via a long slot opening at the top of the upper sections of the shielding. The transfer skid supplemental shielding is shown in Figure W.1-3.

- The loading procedures when using the OS197L TC are modified *as described in Appendix W.8*. See Section W.4.8 for the effects of these modifications *on the thermal analysis.*

The applicable normal, off normal and accident conditions for the transfer of 61BT *and* 32PT DSC*s* when using OS197L TC remain unchanged from those defined in the UFSAR for the transfer of these payloads when using OS197 TC.

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The OS197L TC without the supplemental shielding *has* less radial thermal resistance than the OS197 TC since the thermal resistance associated with the lead to shell gaps and the combination of lead and steel in the OS197 TC is larger than the thermal resistance due to the gap between the neutron shield assembly and the structural shell in the OS197L TC. Therefore, OS197L TC will result in lower DSC shell temperatures compared to the OS197 TC for the same heat load and ambient conditions.

W.4.2 Summary of Thermal Properties of Materials

The thermal properties of the materials (including the effective thermal conductivity of fuel with helium backfill) used in the thermal evaluation in this section are the same as those specified in Appendix M, Section M.4.2 for 32PT DSC *and* Appendix K, Section K.4.2 for 61BT DSC.

The effective thermal conductivity of water and air-filled neutron shield of OS197L TC, are calculated based on the same methodology as described in Appendix M, Section M.4-9. *The effective thermal conductivity of water and air-filled neutron shield* are listed in Table W.4-1.

W.4.3 Specifications for Components

The mechanical properties of the materials applicable to *the* OS197L TC are the same as those described in Section 8.1, Table 8.1-3 for *the* OS197 TC.

W.4.4 Effect of the Decontamination Shield on OS197L TC Thermal Performance

An evaluation is performed to confirm that the radial gap between the OS197L TC and the inner diameter of the decontamination shield is sufficiently large, and that the size of the top and bottom cut out openings are of sufficient size *and does* not adversely affect the thermal performance of the OS197L TC. The evaluation is based on analysis of the free convection turbulent boundary layer development along the outer OS197L TC surface during vacuum drying and helium backfilling operations. The results of the evaluation confirm that the DSC shell-decontamination shield gap and the area of the inlet and outlet openings are adequate and, thus, the decontamination area shield does not adversely impact the cask boundary conditions assumed in the thermal analysis.

W.4.5 Effect of the Supplemental Skid Shielding on OS197L TC Thermal Performance

As discussed above, supplemental shielding is installed on the OS197L skid to compensate for the reduced shielding capability of the OS197L TC. Three-dimensional views of the supplemental skid shielding are shown in Figure *W.4-3*. The shielding enclosure is provided with openings between the skid beams and the trailer deck to allow air to enter the enclosure, flow around the OS197L TC, and exit the enclosure through an opening at the top.

To address the potentially offsetting effect of the supplemental skid shielding on the thermal performance of the OS197L TC with a *13* kW heat load DSC, a computational fluid dynamics (CFD) analysis is performed for the OS197L TC with the supplemental shielding.

All changes on this page are AMD 11

The FLUENT™ and GAMBIT™ codes [4.1] are used for this analysis. The FLUENT™ code is a general-purpose computational fluid dynamics (CFD) code that is recognized internationally as one of the premier codes in its class. The general modeling capabilities of the code as they relate to this application include:

- Meshing flexibility using structured and unstructured mesh generation with hexahedra, non-hexahedra, and tetrahedral mesh types
- Capability to model low speed, buoyancy driven flow regimes
- Steady-state and transient flows
- Inviscid, laminar, and turbulent flows
- Heat transfer including forced, natural, and mixed convection, conjugate heat transfer, and radiation
- Custom materials property database
- Integrated problem set-up and post-processing

GAMBIT™ is an interactive, object-based software code that allows complex geometries to be modeled and meshed using a combination of shapes. Quadrilateral and triangular elements are used for 2D simulations, while hexahedra, tetrahedra, prisms, and pyramid shaped elements are available for 3D simulations. The GAMBIT™ module does not perform any CFD related numerical calculations itself, but serves as a preprocessor to the FLUENT™ code to generate a computational mesh. GAMBIT™ has many automated features for building or joining hybrid meshes with attention to boundary layers, non-uniform sizing, and core regions of hexahedral cells.

The verification and validation of the FLUENT™ and GAMBIT™ codes for the computation of generic buoyancy driven convection heat transfer within an enclosure is documented in [4.2].

The FLUENT CFD methodology is used to compute the neutron shield shell temperatures for the OS197L TC while on the shielded transfer trailer. The computed neutron shield shell temperatures are then used as boundary temperatures for the detailed ANSYS model of the OS197L TC described in Section W.4.6.

<u>CFD Model of OS197L *TC* and Transfer Skid</u>

A three-dimensional model of the OS197L *TC* and transfer skid was created using GAMBIT™ using the drawings provided in Section W.1.5. For the purposes of this analysis, the cask is represented as a surface with a uniform heat flux simulating the outer shell of the liquid neutron shield.

The methodology of using a uniform heat flux for the evaluation of the heat transfer mechanisms outside of the DSC shell and the determination of the canister shell temperature distribution has *been used for numerous safety analyses previously approved by the NRC. These safety analyses include the following approved applications:*

1) *Standardized Advanced NUHOMS® Horizontal Storage System (Docket No. 72-1029) for models NUHOMS®-24PT1 and -24PT4,*

2) *NUHOMS® HD System (Docket No. 72-1030); models NUHOMS®-32PTH,*

3) *Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel (Docket No. 72-1004); models NUHOMS®-24P, -24PHB, -32PT, and -24PTH.*

In each of these safety analyses, the canister shell temperature distribution was determined using a uniform heat flux boundary condition on the canister shell. Further, for the NUHOMS® HD System (Docket No. 72-1030) thermal evaluation, the effective thermal conductivity within the neutron shield of the OS187H transfer cask (i.e., a water filled annular region that surrounds the cask's structural shell) is based on the use of a uniform heat flux boundary condition applied to the cask's structural shell. In each case, the resulting peak fuel cladding and system temperatures predicted using this methodology were validated by NRC models that included the effects of non-uniform heat fluxes.

All changes on this page are AMD 11

As described in Section W.4.6, a separate model of the *OS197L TC* is used to evaluate the heat transfer within the *TC* using the computed temperatures on the neutron shield as a boundary condition. Based on a 183.85-inch length for the water cavity in the neutron shield, an outside radius of 40.18-inches for the neutron shield shell, and a design decay heat loading of *13* kW for the 32PT DSC, the uniform heat flux applied over the surface area of the shell is computed as:

$$\ddot{q} = \frac{13\,\text{kW} \cdot 3412.1415 \frac{\text{Btu}/\text{hr}}{\text{kW}}}{\left(2 \cdot \pi \cdot 40.18\,\text{in} \cdot 183.85\,\text{in}/144 \frac{\text{in}^2}{\text{ft}^2}\right)} = 137.62 \frac{\text{Btu}}{\text{hr} \cdot \text{ft}^2}$$

A 12-inch long segment at the center of the OS197L TC on the shielded transfer trailer (see Figure *W.4-4*) is used to compute the flow field around the TC. Radiation exchange is modeled using the discrete ordinate methodology.

The computational mesh extends *660*-inches in the x-direction and *1,000*-inches in the y-direction. Figure *W.4-5* illustrates perspective and plane views of the computational mesh at the centerline. Figure *W.4-6* presents enlarged views of the computation mesh illustrating the boundary layer mesh on the cask shell and the inner surface of the shields. A grid sensitivity study, conducted as part of the analysis for a variant of the OS197L TC design, demonstrated that the meshing used for this evaluation was appropriate.

While the computational model is 3-D in its construction, the resulting analysis is effectively 2-D since symmetry conditions are assumed at the axial ends of the model and a uniform heat flux is assumed at the inner surface.

Assumptions Used in CFD Modeling

The general assumptions used in the CFD modeling are:

1. Heat removed through the cask end plugs and by conduction via trunnion contact with the transfer skid is conservatively neglected.

2. The total decay heat is considered to be evenly distributed over the outer surface of the cask's liquid neutron shield shell. The assumption of uniform heat flux is consistent with previous OS197 analysis methodology (See Chapter M.4) and reflects the axial spreading of the decay heat load due to the high axial conductivity of the DSC basket and rails, and the water filled neutron shield.

3. The CFD modeling need only address the geometry of the OS197L TC and its shielded transfer skid as it exists between the front and rear trunnion towers. See the justification of this assumption provided below.

4. The outer surfaces of the auxiliary shielding on the transfer skid are assumed to be finished with a 'dark blue' color coating that yields a solar absorptivity of 0.90 or less and an emissivity of 0.85 or greater. Similarly, the inner surface of the auxiliary shielding is assumed to have a similar finish that yields an emissivity of 0.85 or greater.

5. The regulatory insolation [10CFR Part 71] averaged over 24 hours is applied to the outer surfaces of the auxiliary shielding. The thickness of the auxiliary shielding, combined with the thermal mass of the OS197 cask and payload, justifies the use of 24-hour averaged values. The 24-hour average insolation on the roof of the transfer skid is assumed to be 122.9 Btu/hr-ft^2, 30.75 Btu/hr-ft^2 on the vertical surfaces, and 61.45 Btu/hr-ft^2 on the angled portion of the auxiliary shielding. These incident heating values are reduced by 10% to account for the assumed solar absorptivity of 0.90 for the coating used on the shields (see Assumption 4).

6. The ground is conservatively treated as an adiabatic surface.

The analysis for the off-normal ambient condition of 117°F is conducted assuming a 24 hour average steady-state ambient temperature of 107°F. A steady-state analysis at this temperature level has been shown by previous analyses to bound the transient thermal performance achieved using a diurnal cycle for ambient air with a peak temperature of 117°F.

Justification of CFD Model Segmentation

The use of a quasi 2-D thermal model located at the center of the OS197L TC to evaluate the bounding thermal performance of the OS197L TC on its shielded transfer skid is justified for the following reasons:

1. Heat transfer near the center of the OS197L TC is via radiation and convection, while heat transfer near the ends of the cask also involves direct heat transfer through the trunnions to the transfer skid. In addition, the cask's lid and end closures add additional heat transfer area. As such the peak temperatures on the OS197L TC will occur near its center.

2. A flow area clearance of approximately 3.8 inches will exist in the vicinity of the cask trunnions vs. approximately 3.3 inches at the center of the cask.

3. The 113.5 inches between the front and rear trunnion towers spans 62% of the liquid neutron shield length, over which the majority of the heat rejection occurs.

4. The assumption of a uniform heat flux means there will be no significant variation in thermal conditions between the front and rear trunnion towers. Thus, modeling a short segment will yield similar results to modeling the entire length.

5. The airflow through the shielded transfer skid is driven by buoyancy forces which, in turn, are driven by the temperature on the shell of the neutron shield. As such, the convective airflow over the surface of the TC is self-correcting, with the level achieved representing a balance between the local flow resistance and surface heat flux. For this reason the use of a relatively short axial segment to represent the flow regime within the transfer skid enclosure does not entail a risk of over-estimating the prototypic airflow since the computed airflow results from the computed local temperatures and not as the result of airflow occurring at another axial location.

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CFD Analysis Results

The FLUENT™ model of the OS197L cask and transfer skid described above was used to compute the flow and temperature distribution for the bounding normal and off-normal hot *transfer* conditions with *a* decay heat load of *13* kW. A second order discretization scheme for energy, momentum, turbulence, and the discrete ordinate calculation, and the PRESTO solution scheme for pressure are used for the solution. Since the boundary layer mesh yields y+ values of approximately 1.0, the *standard* turbulence model with enhanced wall functions is used to compute the turbulent heat transfer at the surface of the cask.

Figure *W.4-7* and Figure *W.4-8* present the temperature distribution for the surface of the cask exterior shell and the transfer skid shields, respectively, for normal ambient condition of 100°F with insolation and with a *13* kW decay heat loading. The peak temperature on the cask shell is predicted to occur back from the centerline of the cask, at the point where the flow separates from the cask and heads towards the exit. The fact that the cask shell temperature reaches a peak and then decreases slightly at the very top of the cask is attributed to the presence of flow recirculation in this region. Because of this recirculation, the surface flow does not stagnate at the top, center of the cask as it would for an isolated cask and a lower surface temperature is achieved. Instead, the peak temperature point occurs away from the centerline of the cask, where the flow separation point is predicted to occur.

Figure *W.4-9* illustrates the velocity profiles at the centerlines of the model. The minimum cask-shield gap for the modeled section of the OS197L cask and transfer skid combination is approximately 3.3-inches. Figure *W.4-10* illustrates an enlarged view of the velocity profile at the exit from the auxiliary shielding enclosure. The predicted region of flow stagnation and flow recirculation under the hat section of the enclosure can be seen in *lower left region of* the figure.

The peak neutron shield shell temperature with a *13* kW decay heat loading and a normal 100°F ambient condition is predicted to be *248*°F, while the area-weighted average temperature is predicted to be *217*°F.

The evaluation was repeated for the bounding off-normal hot condition of 117°F ambient with insolation. Figure *W.4-11* presents the temperature distribution for the surface of the cask exterior shell. The *peak neutron shield* cask temperature predicted for this condition is *253*°F, while the area-weighted average temperature over the cask's exterior shell is predicted to be *222*°F. The velocity profiles developed are similar to those presented in Figure *W.4-9* and Figure *W.4-10*.

The computed temperatures on the neutron shield from the CFD analysis are used as boundary conditions for the thermal analysis of the OS197L TC using the ANSYS model described in Section W.4.6.

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W.4.6 Thermal Analysis of OS197L TC with *13* kW Heat Loads

A two-dimensional model of the OS197L TC and DSC shell is developed using ANSYS Computer Code [4.3]. The 2D model considers the hottest cross-section of the fuel and conservatively neglects heat transfer in the axial direction. The model represents the neutron shield, the cask structural shell, cask rails, and the DSC shell. The OS197L TC model is shown in Figure *W.4-12*. The TC thermal model and analysis methodology are consistent with the methodology described in Appendix M, Section M.4.4.1.6 for the OS197 TC with a 32PT DSC payload with changes implemented to account for the configuration changes in the OS197L TC relative to the OS197 TC.

The justification for the use of a uniform heat flux on the inner shell of the DSC for predicting the temperature distribution on the outside of the DSC shell is based on the discussion provided for Section W.4.5. The effects of the non-uniform gap between the DSC and the TC inner shell is addressed by 2D ANSYS OS197L TC model and is reflected in the predicted DSC surface temperature distribution. The effects of the support rails (and other fuel basket details) plus the positioning of the active fuel region and the fuel peaking factor are addressed by the 3D ANSYS model of the DSC described in Section W.4.7.

The heat flux applied to the DSC outer shell in OS197 TC model for the 13 kW heat load is calculated using method described in Appendix M, Section M.4.4.1.6 using corresponding DSC shell inner diameter and DSC cavity length as follows.

1) For 32PT DSC in OS197L TC

$$\ddot{q} = \frac{13\,kW \cdot 3412.3\,\frac{\mathrm{Btu}/\mathrm{hr}}{\mathrm{kW}}}{60\,\frac{\mathrm{min}}{\mathrm{hr}} \cdot (\pi \cdot 66\,\mathrm{in} \cdot 167.1\,\mathrm{in})} = 0.0213\,\frac{\mathrm{Btu}}{\mathrm{min} \cdot \mathrm{in}^2},$$

Note: Conservative inner diameter of 66" of 32PT DSC shell is used similar to Appendix M, Section M.4.4.1.6,

2) *For 61BT DSC in OS197L TC*

$$\ddot{q} = \frac{13\,kW \cdot 3412.3\,\frac{\text{Btu}/\text{hr}}{\text{kW}}}{60\,\frac{\text{min}}{\text{hr}} \cdot (\pi \cdot 66.25\,\text{in} \cdot 179.31\,\text{in})} = 0.0198\,\frac{\text{Btu}}{\text{min} \cdot \text{in}^2},$$

Note: Conservative inner cavity length of 179.31" for 61BT DSC is used similar to Appendix K, Section K.4.4.1.

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The following table summarizes the OS197L TC dimensions (See OS197L TC Drawings included in Section W.1.5) used in the thermal analyses.

Dimensions used in OS197L TC ANSYS Thermal Analysis Model

DSC/Cask Component	OS197L
DSC Shell Outside Diameter, in.	*67.19*/67.25$^{(1)}$
DSC Shell Thickness, in.	0.5$^{(2)}$
Cask Inner Radius, in.	34.0
Structural Shell Thickness, in.	2.68
Structural Shell Outside Radius, in.	36.68
Structural Shell-Neutron Shield Inner Panel Gap, in.	0.01
Neutron Shield Inside Radius, in.	36.39
Neutron Shield Inner Panel Thickness, in.	0.25
Neutron Shield Inner Radius, in.	36.94
Neutron Shield Thickness, in.	3.0
Neutron Shield Outside Radius/Neutron Shield Outer Panel Inner Radius, in.	39.94
Neutron Shield Outer Panel Thickness, in.	0.25
Neutron Shield Outer Panel Outside Radius, in.	40.19

Note:

1) 67.19 in. is used for the 32PT DSC and 67.25 in. is used for the 61BT DSC.
2) The DSC shell temperature obtained from the OS197L model for *13* kW heat load are based on a DSC shell thickness of 0.5".

As discussed in Section W.4.5, the CFD analyses consider the effect of the supplementary shielding and insolance on the OS197L TC thermal performance. A separate analysis of the OS197L TC loaded with a *13* kW heat load is performed using the *neutron shield outer skin* temperature distribution calculated from the CFD analysis discussed in Section *W.4.5*.

The temperatures of the neutron shield outer skin obtained from the CFD analysis are applied as boundary conditions over the outermost nodes of the 2D model of the OS197L TC and analyses are performed for normal, off-normal and accident conditions. Conservatively, the CFD analysis for off-normal 117°F ambient temperature *condition* includes the effect of insolation on the supplemental shielding.

Two hypothetical accident scenarios for loss of neutron shield and sun shade are considered. In the first scenario, the supplementary shielding remains intact while the liquid neutron shield and sunshade is lost. The supplementary shielding is exposed to direct solar impact in this scenario. In the second scenario, the supplementary shielding and the liquid neutron shielding are both lost and no sun shade is available to protect the neutron shield shell from the insolation. While the supplementary shielding prevents direct insolation heating of the cask surface, it also affects the convective and radiative heat transfer from the cask. Based on thermal analysis results in Section W.4.5, the presence of the supplementary shielding results in higher cask surface temperatures compared with the situation without the supplementary shielding. Therefore, the first hypothetical accident scenario (i.e., loss of neutron shield and sun shade when the supplementary shielding remains in place) represents the bounding accident condition and is chosen in this calculation to provide the bounding cask and DSC component temperatures.

The results of the analysis in terms of maximum TC component temperatures are summarized in Table W.4-2 for the *13* kW case. The maximum temperatures at the top, side, and bottom of the DSC shell retrieved from the 2D model of the TC are also summarized in Table W.4-2. These temperatures are used to define the boundary conditions for the analysis of the 3D models of the DSC shell/basket assemblies as discussed in Section W.4.7.

The 2D model analysis results of the 117°F ambient with insolence and accident case for the OS197L TC are also shown in Figure *W.4-13* for the *32PT DSC with 13 kW head load case.*

W.4.7 Thermal Analysis of NUHOMS® DSCs

W.4.7.1 Thermal Analysis of 32PT DSC inside the OS197L TC

The maximum temperatures at the top, side, and bottom of the DSC shell calculated in Section W.4.6 above define the boundary conditions for the analysis of the 3D ANSYS model of the DSC shell and basket assembly described in Appendix M, Section M.4.4.1.1. The methodology used for the analysis, including the application of the temperature boundary conditions on the DSC shell, is identical to that used for the 32PT DSC and documented in Appendix M.

The 32PT DSC heat load zoning configuration allowed for transfer in OS197L is shown in Figure W.4-1. DSC shell temperature distribution is calculated in Section W.4.6 for the maximum heat load of 13 kW allowed for OS197L TC. Additional conservatism is included in 32PT DSC analyses using the maximum heat load of 16 kW per 32PT DSC based on maximum heat load per fuel assembly shown in Figure W.4-1. The calculation of heat load per fuel assembly is similar to one used in Appendix M, Section M.4.4.1.4. It is to be noted that the heat load zone configuration used for 32PT DSC thermal analysis shown in Figure W.4-1 is conservative relative to the heat load zone configuration shown in Chapter W.2, Figure W.2-2.

The peaking factor profile used in Appendix M is used in this analysis to determine the bounding basket component and fuel cladding temperatures.

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The fuel cladding and *32PT* DSC maximum *component* temperatures for *13 kW heat load* are summarized in Table *W.4-3*. Typical component temperatures are shown in Figure *W.4-14* for the off-normal 117°F ambient condition.

The average cavity gas temperatures are determined from the analysis model results using the same methodology as that used in Appendix M. Based on the ideal gas law, the DSC internal pressure is proportional to the absolute temperature. Thus the 32PT DSC pressures evaluated in Appendix M are multiplied by the ratio of the absolute temperatures obtained from this analysis (OS197L) to those in Appendix M (OS197) to determine the pressures when the *32PT* DSC is transferred in the OS197L TC. The resulting maximum pressures are tabulated in Table *W.4-4*.

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W.4.7.2 Thermal Analysis of 61BT DSC Inside the OS197L TC

The maximum temperatures at the top, side, and bottom of the DSC shell calculated in Section W.4.6 above define the boundary conditions for the analysis of the 3-dimensional, 180° symmetric model of the *61BT* DSC shell and basket assembly documented in Appendix K, Section K.4.4.1.

The 61BT DSC heat load zoning configuration allowed for transfer in OS197L is shown in Figure W.4-2. DSC shell temperature distribution is calculated in Section W.4.6 for the maximum heat load of 13 kW allowed for OS197L TC. Additional conservatism is included in 61BT DSC analyses using the maximum heat load of 14.7 kW per 61BT DSC based on maximum heat load per fuel assembly shown in Figure W.4-2. The calculation of heat load per fuel assembly is similar to one used in Appendix K, Section K.4.4.1. It is to be noted that the heat load zone configuration used for 61BT DSC thermal analysis shown in Figure W.4-2 is conservative relative to the heat load zone configuration shown in Chapter W.2, Figure W.2-1.

The peaking factor profile used in Appendix K is used in *the* analysis to provide basket and fuel cladding temperatures.

The thermal analysis methodology used, including the application of the temperature boundary conditions on the DSC shell in the 3D model, is identical to that used for the 32PT DSC and documented in Appendix M.

The fuel cladding and DSC components maximum temperatures for the controlling (bounding) conditions are summarized in Table *W.4-5*. Typical component temperatures are shown in Figure *W.4-15* for the off-normal 117°F ambient condition.

As discussed in Section K.4.8.2, up to sixteen damaged fuel assemblies in specified positions are allowed for the 61BT DSC basket. As shown in Section K.4.8.2.4, including damaged fuel assemblies has a negligible impact on the fuel cladding and basket material temperatures when compared with the all-intact fuel assembly case and none of the material temperature limits are exceeded for 18.3 kW heat load per DSC. This conclusion remains valid for a lower heat load of 13 kW per DSC with heat load zone configuration shown in Figure W.4-2.

The average cavity gas temperatures are determined from the analysis model results using the same methodology as that used in Appendix K. Based on the ideal gas law, the DSC internal pressure is proportional to the absolute temperature. Thus the 61BT DSC pressures evaluated in

Appendix K are multiplied by the ratio of the absolute temperatures obtained from this analysis (OS197L) to those in Appendix K (OS197) to determine the pressures when the *61BT* DSC is transferred in the OS197L TC. The resulting maximum pressures are tabulated in Table W.4-7.

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W.4.7.3 *Thermal Gradients for 32PT and 61BT DSCs*

To provide input for structural evaluation of 32PT DSC and 61BT DSC (Chapter W.3), Table W.4-7 summarizes the thermal gradients in the 32PT and 61BT DSCs for 13 kW heat load in comparison to the thermal gradients calculated for 24 kW for 32PT DSC and 18.3 kW for 61BT DSC for maximum off-normal transfer conditions.

As seen from Table W.4-6 the maximum fuel cladding and DSC shell temperatures and thermal gradients are lower for a 13 kW heat load for 32PT and 61 BT DSCs, compared to values calculated for 24 kW for the 32PT DSC and 18.3 kW for the 61BT DSC. Therefore, the existing structural analyses using 24 kW for the 32PT DSC and 18.3 kW for the 61BT DSC presented in Appendix M, Chapter M.3 and Appendix K, Chapter K.3, respectively, are bounded (Section W.3).

W.4.8 Effect of Modification of Loading Procedures on OS197L TC Thermal Performance

Following the completion of loading fuel inside the DSC, the OS197L TC is lifted out of the fuel pool with the TC neutron shield and the DSC cavity full of water. This configuration bounds the configuration analyzed for 32PT DSC and 61BT DSC in Appendix M.4 and K.4 where a small quantity of DSC cavity water is drained.

For completion of subsequent operations such as DSC drying and sealing, the OS197L TC is placed inside the decontamination area supplemental shield as described in Appendix W.8. As discussed in Section W.4.4, its thermal performance with the decontamination area shields during such operations, is bounded by the thermal analysis presented in Sections K.4.7 and M.4.7.

W.4.9 Thermal Performance of Various DSCs during *32PT and 61BT* Vacuum Drying Operation

As described in Chapter W.8, helium will be used for the blowdown/draindown of water in the DSC cavity. Therefore, subsequent vacuum drying operations occur with a helium environment in the DSC cavity. Water will be maintained in the DSC/TC annulus during vacuum drying operations. Therefore, the fuel cladding temperatures calculated when the OS197L TC is loaded with a DSC during transfer conditions and with the supplemental trailer shields in place will bound the vacuum drying condition.

W.4.10 Thermal Performance of OS197L TC during Fire Accident Conditions

Based on the operating procedures described in Chapter W.8, the OS197L TC will be transferred back to the handling building after any accident case and will not remain within the supplementary shielding for an extended period of time.

The analysis performed for OS197 TC in Appendix K, Chapter K.4 *and* Appendix M, Chapter M.4 shows that the bounding temperatures for accident conditions occur after an extended period of time during post accident steady state conditions.

As discussed in Appendix W, Section W.4.1, the thermal resistance of the OS197 TC is larger than the thermal resistance of the OS197L TC. Therefore, the maximum temperatures for OS197L TC after an extended period of time are bounded by the steady state temperatures obtained for accident conditions of the OS197 TC discussed in Appendix K, Chapter K.4 *and* Appendix M, Chapter M.4.

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W.4.11 References

4.1 FLUENT™ CFD Code Version 6.2, and Gambit Version 2.2, Fluent Inc., 10 Cavendish Court, Lebanon, NH 03766.

4.2 V&V Test Report, FLUENT™ Version 6.2 / GAMBIT™ Version 2.2, Transnuclear, Inc., File Number QA040.231.0001, Revision 0.

4.3 ANSYS Computer Code and On-line User's Manuals, Version 8.1 and Version 10.0A1.

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Table W.4-1
Effective Thermal Conductivity for Water and Air-Filled Neutron Shield of the OS197L TC

For *13* kW		
Angle from the Top of Neutron Shield	**$K_{eff\ water\ filled\ NS}$, Btu/(min-in-°F)**	**$K_{eff\ air\ filled\ NS}$, Btu/(min-in-°F)**
0	*1.241E-02*	*3.379E-04*
30	*1.240E-02*	*3.569E-04*
60	*1.309E-02*	*3.774E-04*
90	*1.361E-02*	*3.848E-04*
120	*1.187E-02*	*3.617E-04*
150	*7.123E-03*	*2.986E-04*
180	*6.378E-04*	*2.125E-04*

Table W.4-2
Maximum Component Temperatures for OS197L TC with *13* kW Heat Load (Supplemental Skid Shielding Effect Included)

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Operating Conditions	Cask Components			DSC Shell Temperatures		
	$T_{str\ sh}$, (°F)	$T_{NS,\ max}$ (°F)	$T_{NS\ avg}$, (°F)	T_{top}, (°F)	T_{side}, (°F)	T_{bot}, (°F)
32PT DSC						
Normal, transfer OS197L	*251*	*246*	*221*	*359*	*332*	*291*
Off-normal, transfer OS197L	*256*	*251*	*226*	*362*	*336*	*297*
Accident, transfer OS197L (Loss of sun shade and liquid neutron shield – with supplementary shielding intact)	*430*	*418*	*303*	*469*	*455*	*460*
61BT DSC						
Normal, transfer OS197L	*250*	*246*	*220*	*349*	*323*	*286*
Off-normal, transfer OS197L	*255*	*251*	*226*	*353*	*327*	*291*
Accident, transfer OS197L (Loss of sun shade and liquid neutron shield – with supplementary shielding intact)	*415*	*404*	*297*	*454*	*439*	*444*

where

T_{top}, T_{side}, T_{bot} – DSC shell top, side, and bottom maximum temperatures, respectively,

$T_{str\ sh}$ – cask structural shell maximum temperature,

T_{NS} – neutron shield maximum temperature (water or air),

$T_{NS\ avg}$ – neutron shield average temperature (water or air).

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Table W.4-3
Maximum Temperatures for *32PT DSC with 13 kW inside* OS197L TC (Supplemental Skid Shielding Effect Included)

Operating Conditions	*Maximum Fuel Cladding Temperatures*	
	$T_{max,fuel}$ (°F)	*Limit* (°F)
Normal, transfer, 0°F without insolation	<606	752
Normal, transfer, 100°F with Insolation	606	752
Off-normal, transfer, 117°F without insolation[(1)]	609	752
Accident, transfer (Loss of sun shade and liquid neutron shield – with supplementary shielding intact)	710	1058

Operating Conditions	*Maximum Component Temperatures*			
	$T_{max,bskt}$ (°F)	$T_{max,Al}$ (°F)	$T_{max,rail}$ (°F)	$T_{max,DSC}$ (°F)
Normal, transfer, 0°F without insolation	<588	<588	<372	<359
Normal, transfer, 100°F with Insolation	588	588	372	359
Off-normal, transfer, 117°F without insolation[(1)]	591	591	376	362
Accident, transfer (Loss of sun shade and liquid neutron shield – with supplementary shielding intact)	697	696	487	469

Note:
(1) Conservatively the CFD analysis included insolation on the supplemental shielding.

Table W.4-4
Maximum Pressures of 32PT DSC with 13 kW in OS197L TC

Operating Conditions	***DSC Pressure in OS197L TC (psig)***	***Design Pressure*** *[Appendix M, Section M.4.1]* ***(psig)***
Normal	*4.0*	*15.0*
Off-normal	*10.8*	*20.0*
Accident	*86.4*	*105.0*

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Table W.4-5
Maximum Temperatures for 61BT DSC with 13 kW inside OS107L TC (Supplemental Skid Shielding Effect Included)

Operating Conditions	**Maximum Fuel Cladding Temperatures**	
	$T_{max,fuel}$ (°F)	**Limit (°F)**
Normal, transfer, 0°F without insolation	*<561*	*752*
Normal, transfer, 100°F with Insolation	*561*	*752*
Off-normal, transfer, 117°F without insolation(1)	*565*	*752*
Accident, transfer (Loss of sun shade and liquid neutron shield – with supplementary shielding intact)	*666*	*1058*

Operating Conditions	**Maximum Component Temperatures**		
	$T_{max,bskt/Al}$ (°F)	$T_{max,rail}$ (°F)	$T_{max,DSC}$ (°F)
Normal, transfer, 0°F without insolation	*<538*	*<430*	*<349*
Normal, transfer, 100°F with Insolation	*538*	*430*	*349*
Off-normal, transfer, 117°F without insolation(1)	*542*	*434*	*353*
Accident, transfer (Loss of sun shade and liquid neutron shield – with supplementary shielding intact)	*644*	*533*	*454*

Note:
(1) Conservatively the CFD analysis included insolation on the supplemental shielding.

Table W.4-6
Thermal Gradients for 32PT and 61BT DSCs for Different Heat Load

	32PT DSC		**61BT DSC**	
Heat load per DSC	*13 kW*	*24 kW [Table M.4-8, Table M.4-9 of Appendix M, Chapter M.4]*	*13 kW*	*18.3 kW [Table K.4-2 of Appendix K, Chapter K.4]*
Maximum fuel cladding temperature, T_{fuel}, °F	*609*	*715*	*565*	*638*
Maximum DSC shell temperature, T_{DSC}, °F	*362*	*433*	*353*	*378*
T_{fuel}-T_{DSC}, °F	*247*	*282*	*212*	*260*

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Table W.4-7
Maximum Pressures *of 61BT DSC with 13 kW* in OS197L TC

Operating Conditions	DSC Pressure in OS197L TC (psig)	Design Pressure *[Appendix K, Section K.4.1]* (psig)
Normal	*6.2*	10.0
Off-normal	*8.5*	20.0
Accident	*36.9*	65.0

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Table W.4-8
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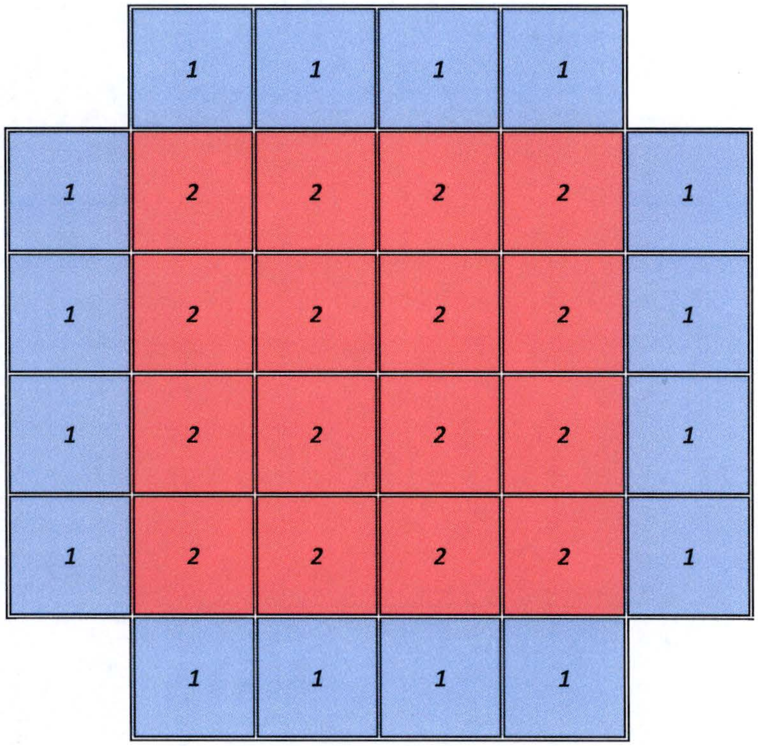
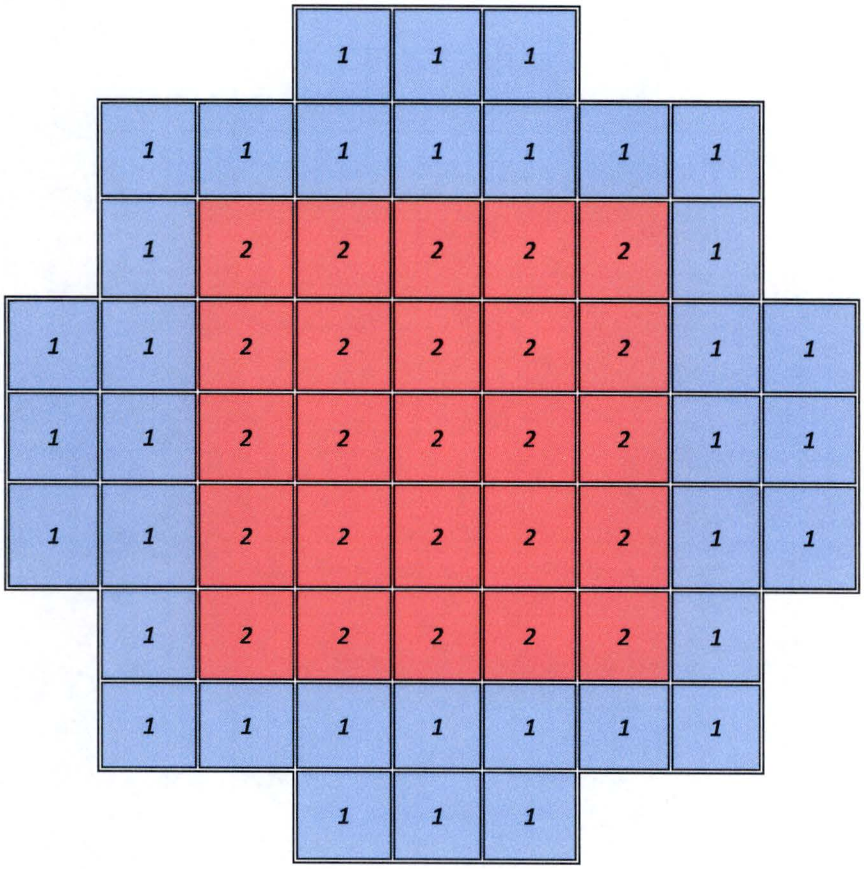


Heat Zone Level	*Zone 1*	*Zone 2*
Max. Decay Heat / FA (kW)	*0.4*	*0.6*
Number of FAs / Zone	*16*	*16*
Max. Decay Heat /Zone (kW)	*6.4*	*9.6*
Max. Decay Heat / DSC (kW)	***13.0*** [(1)]	

Note (1): Maximum decay heat load allowed in the OS197L TC.

***Figure* W.4-1**
Heat Load Zone Configuration for the 32PT DSC



Heat Zone Level	*Zone 1*	*Zone 2*
Max. Decay Heat / FA (kW)	*0.2*	*0.3*
Number of FAs / Zone	*36*	*25*
Max. Decay Heat /Zone (kW)	*7.2*	*7.5*
Max. Decay Heat / DSC (kW)	***13.0*** *(1)*	

Note (1): Maximum decay heat load allowed in the OS197L TC.

***Figure* W.4-2**
Heat Load Zone Configuration for the 61BT DSC

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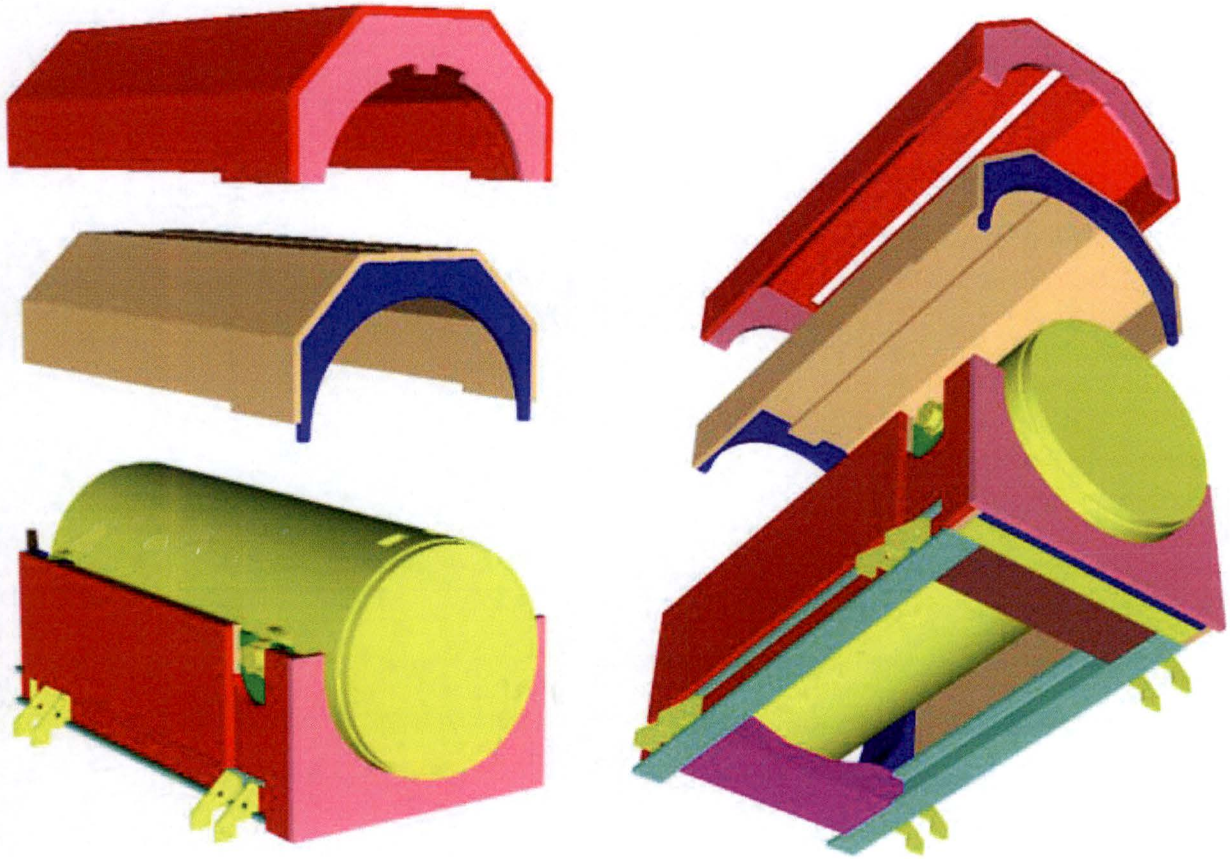

Exploded Side View **Exploded Bottom View**

Figure W.4-3
OS197L TC *Supplemental Shielding*

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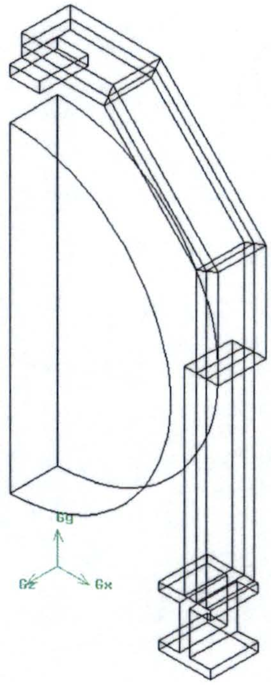

Figure W.4-4
Wire Frame Representation of OS197L Cask and Transfer Skid CFD Model

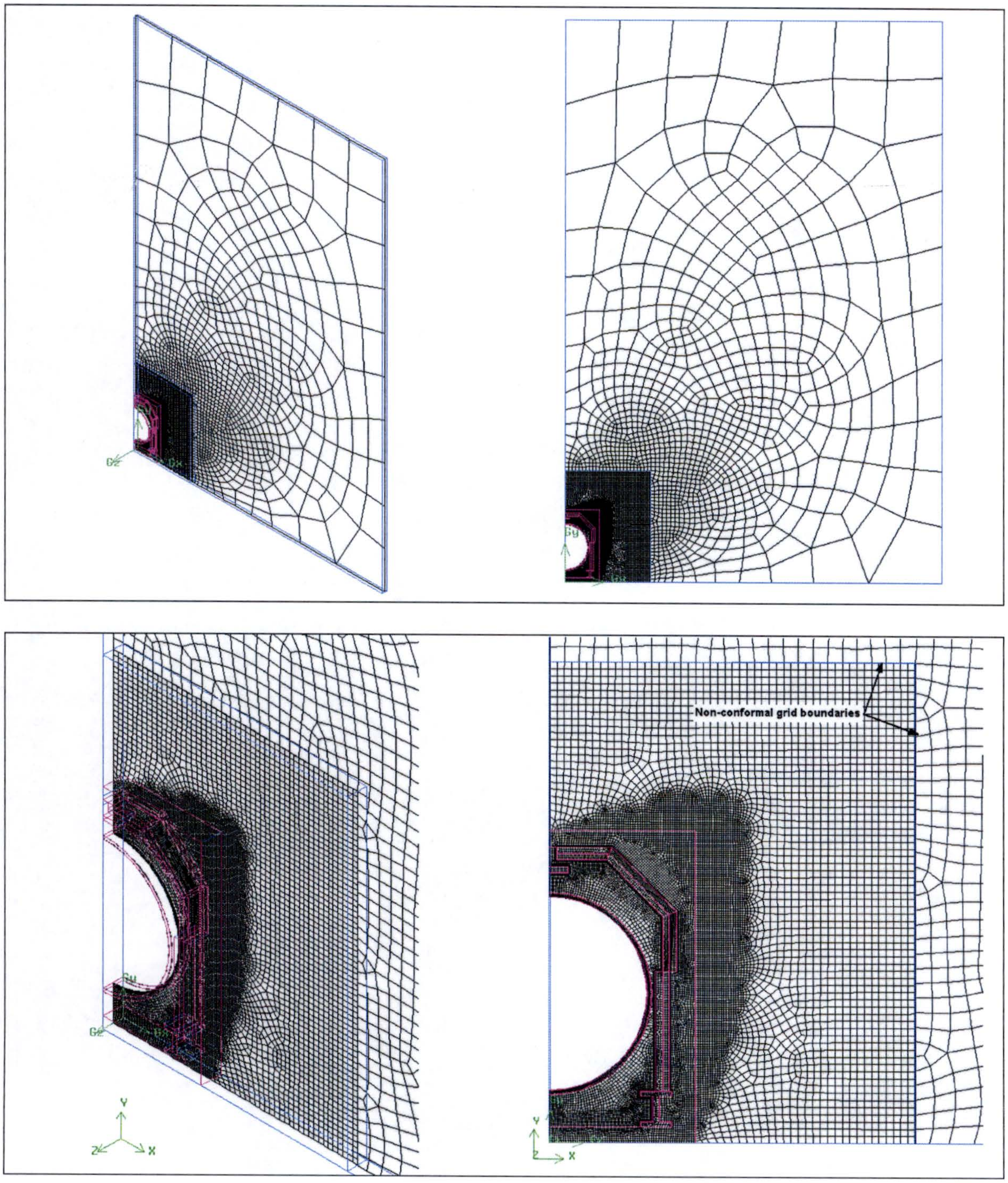

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Figure W.4-5
Perspective and *Elevation* Views of OS197L Cask/Transfer Skid Mesh

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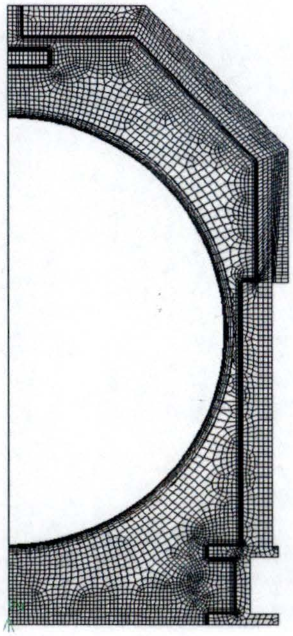
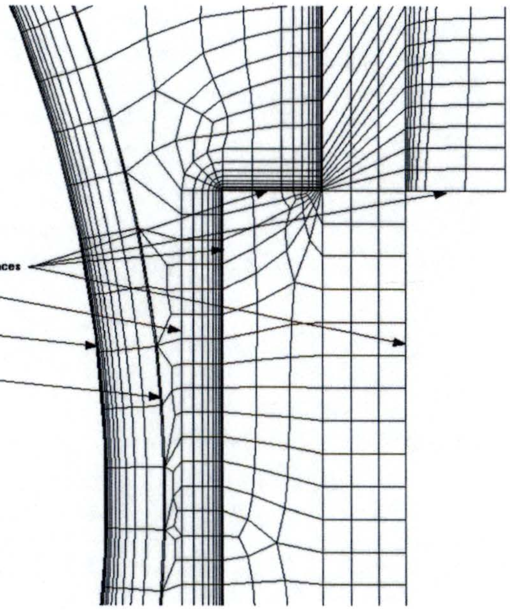
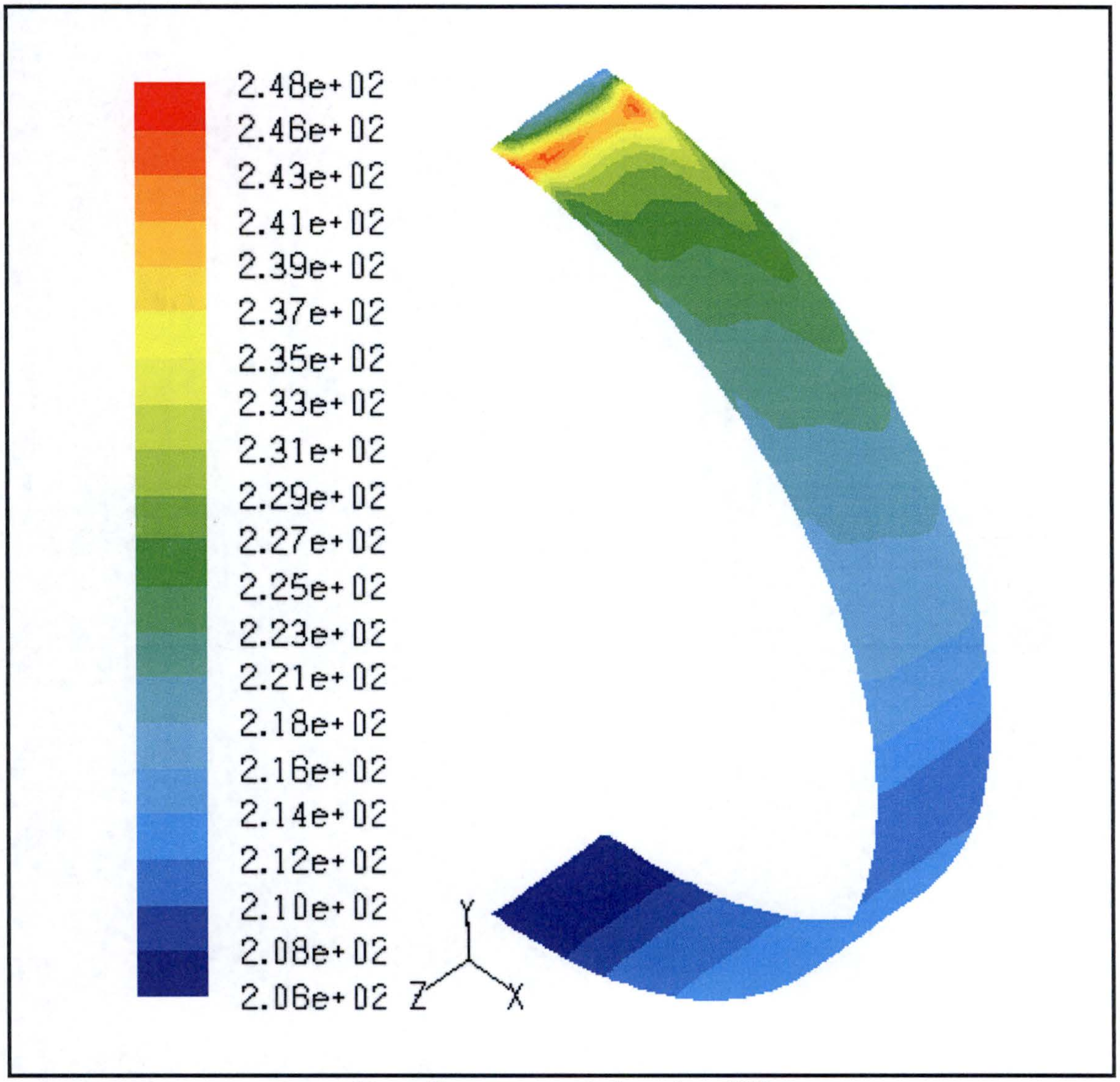


Interior Mesh



Enlarged View at Side of Cask

Figure W.4-6
Views of OS197L Cask/Transfer Skid Mesh



Note: Temperature is in units of °F

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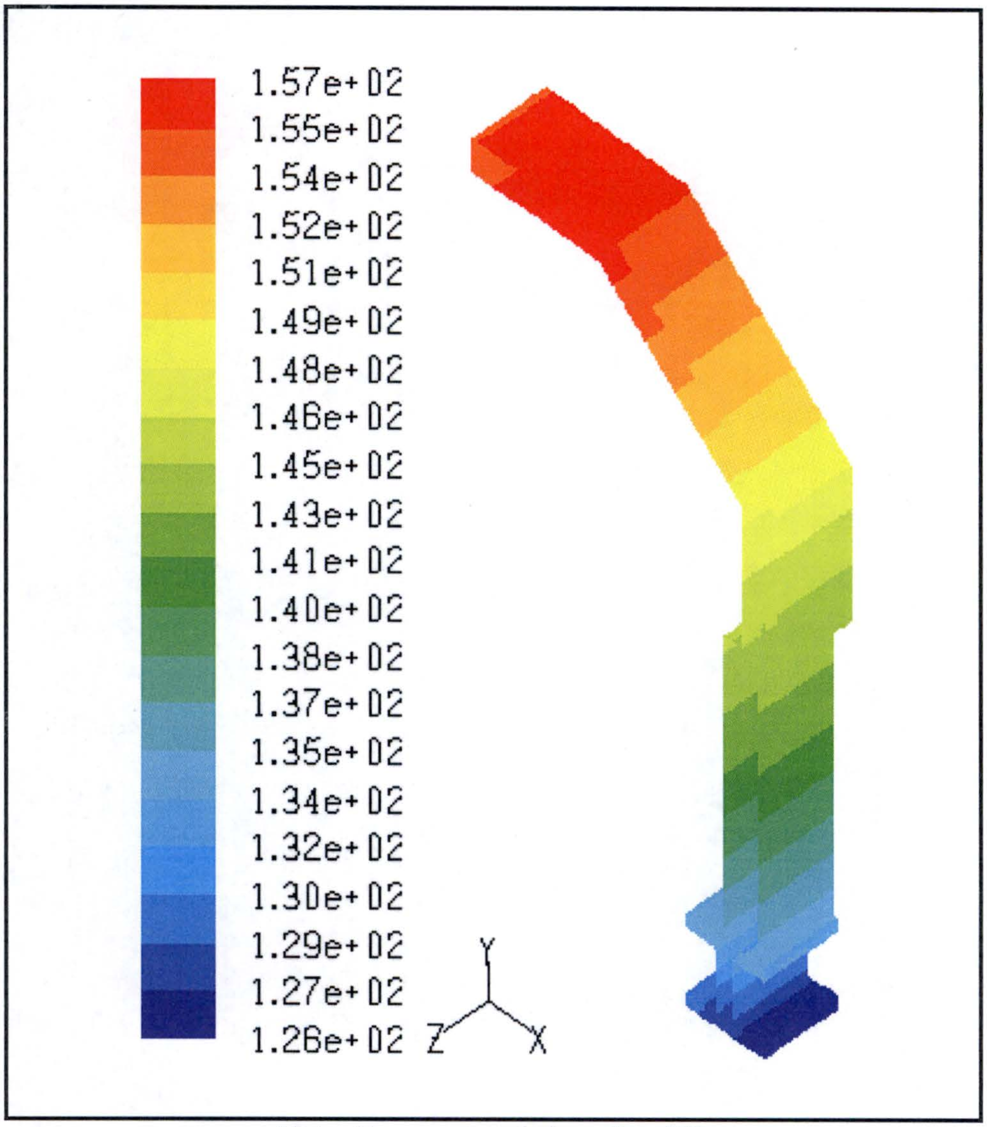
Figure W.4-7
Temperature Distribution over OS197L *TC* Exterior Shell
with *32PT DSC (13 kW)*, 100°F Ambient



Note: Temperature is in units of °F

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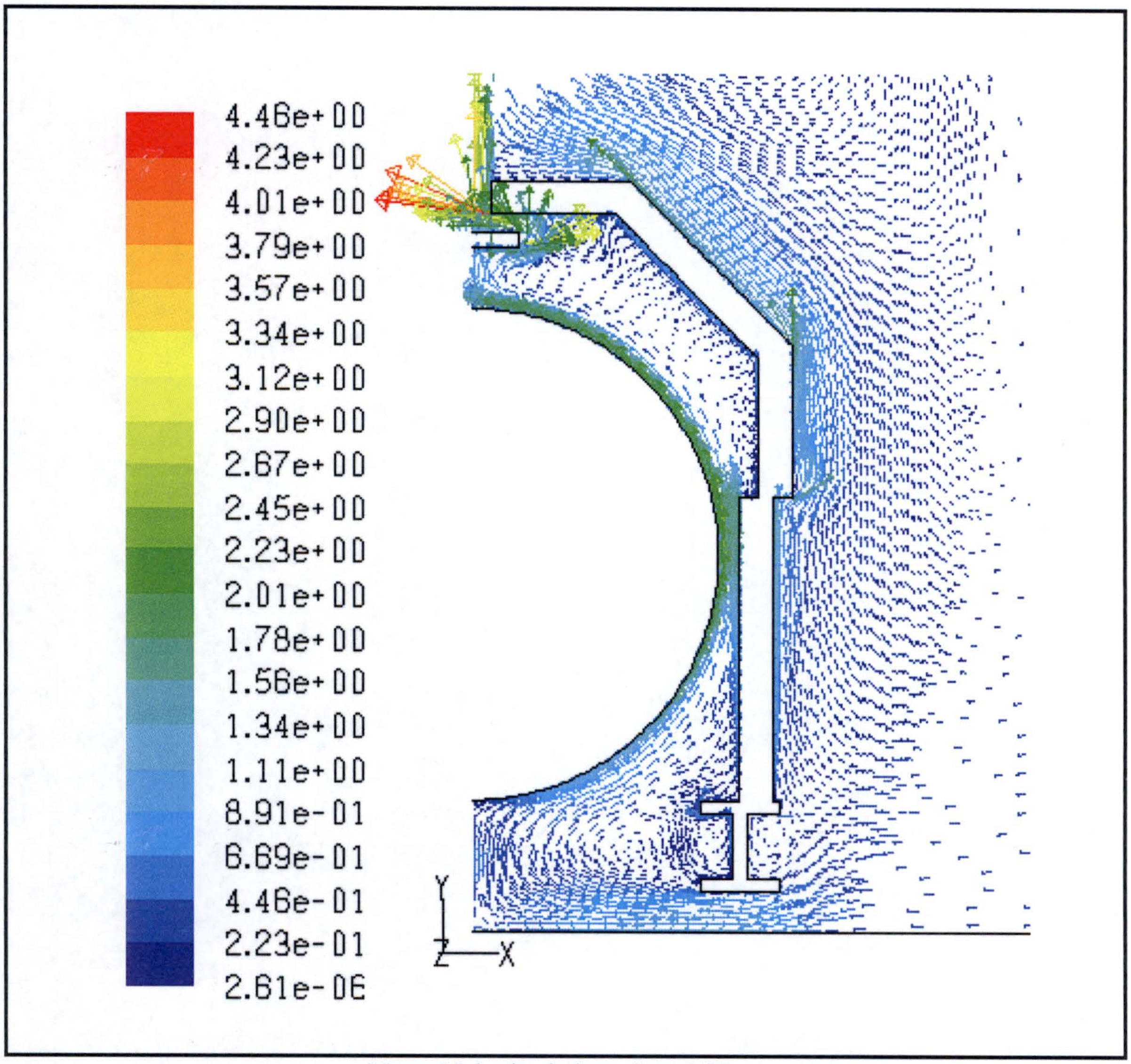
Figure W.4-8
Temperature Distribution for OS197L *TC* Transfer Skid Shields with *32PT DSC (13 kW)*, 100°F Ambient



Note: Velocity is in units of ft/sec

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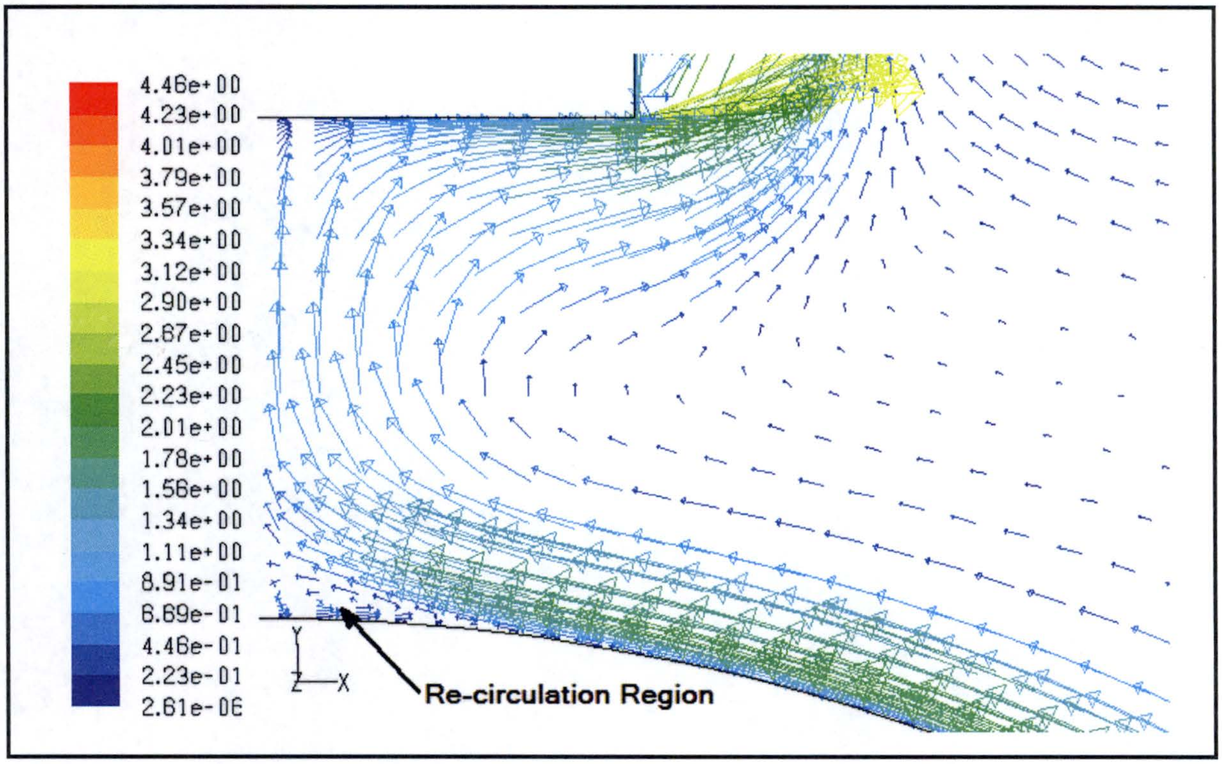
Figure W.4-9
Velocity Distribution at OS197L *TC* Model Centerline for *32PT DSC (13 kW)*, 100°F Ambient, Plan View



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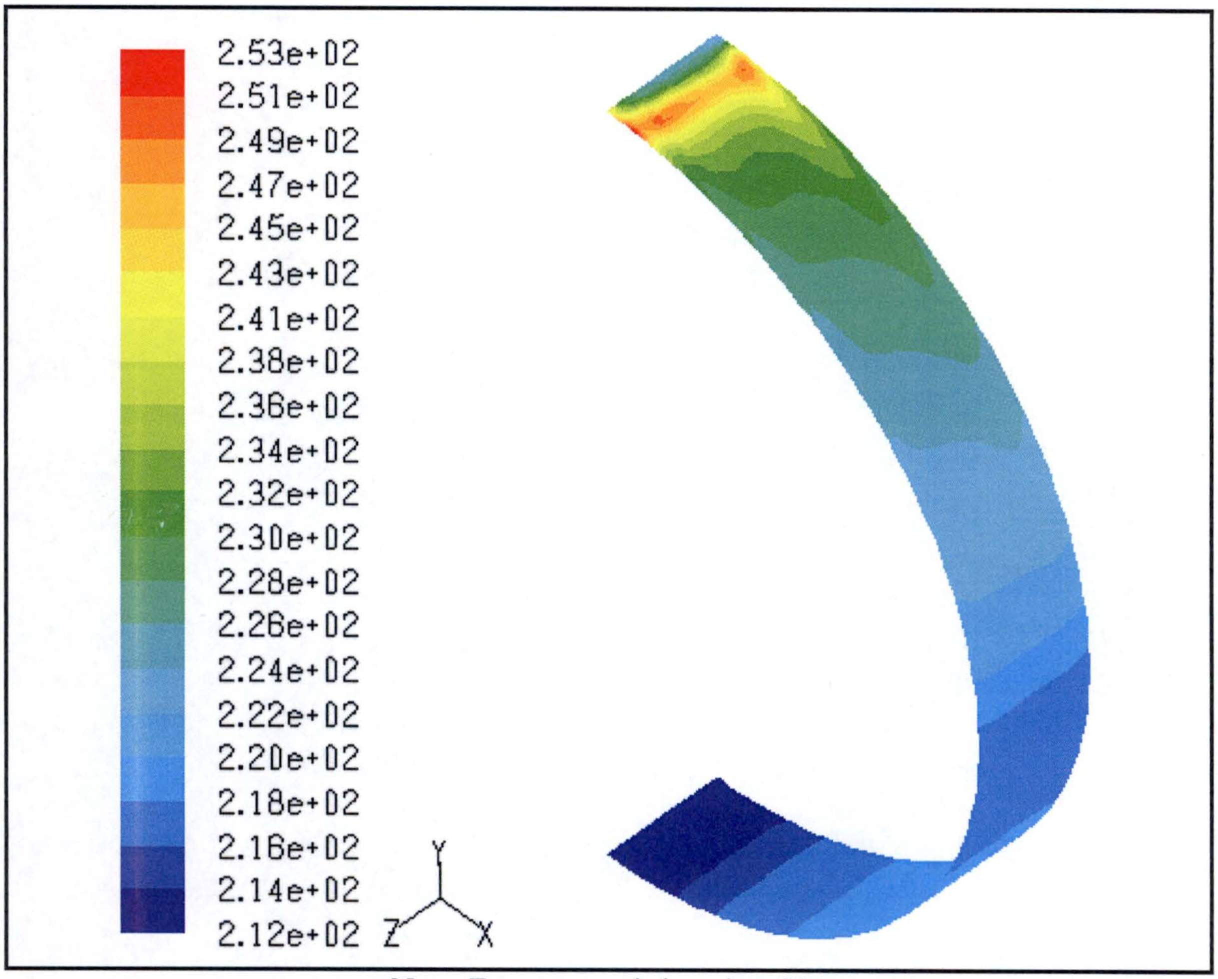
Note: Velocity is in units of ft/sec

Figure W.4-10
Velocity Distribution at OS197L *TC* Exit *with 32PT DSC (13kW)*, 100°F Ambient, Plan View



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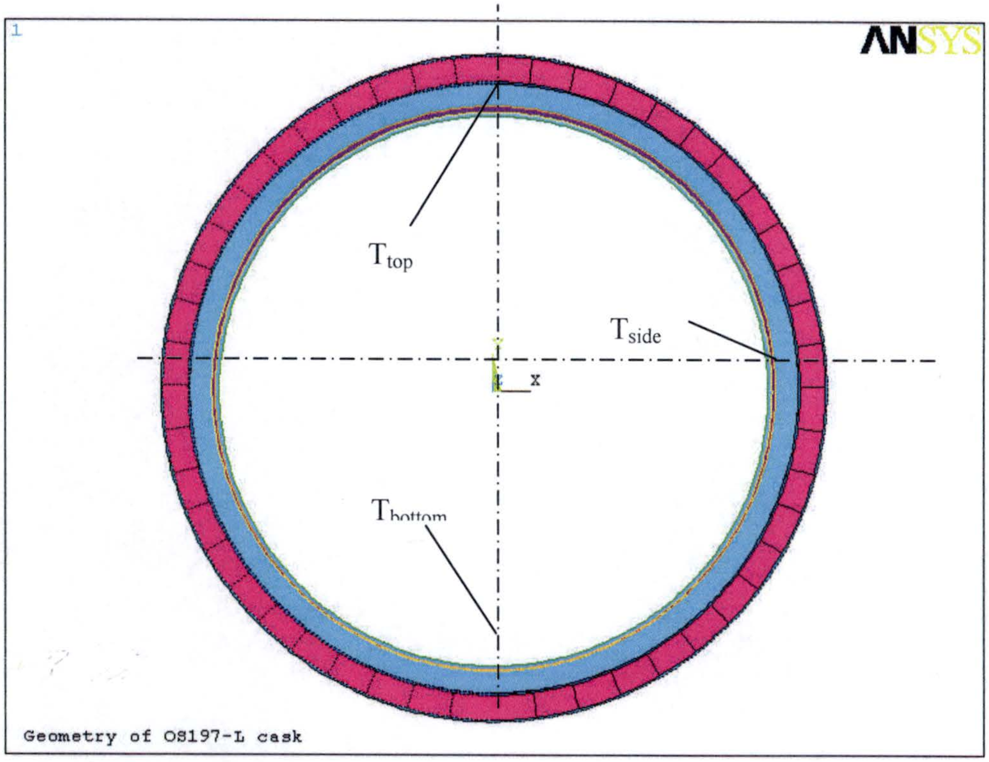
Note: Temperature is in units of °F

Figure W.4-11
Temperature Distribution over OS197L *TC* Exterior Shell with *32PT DSC (13 kW)*, 117°F Ambient



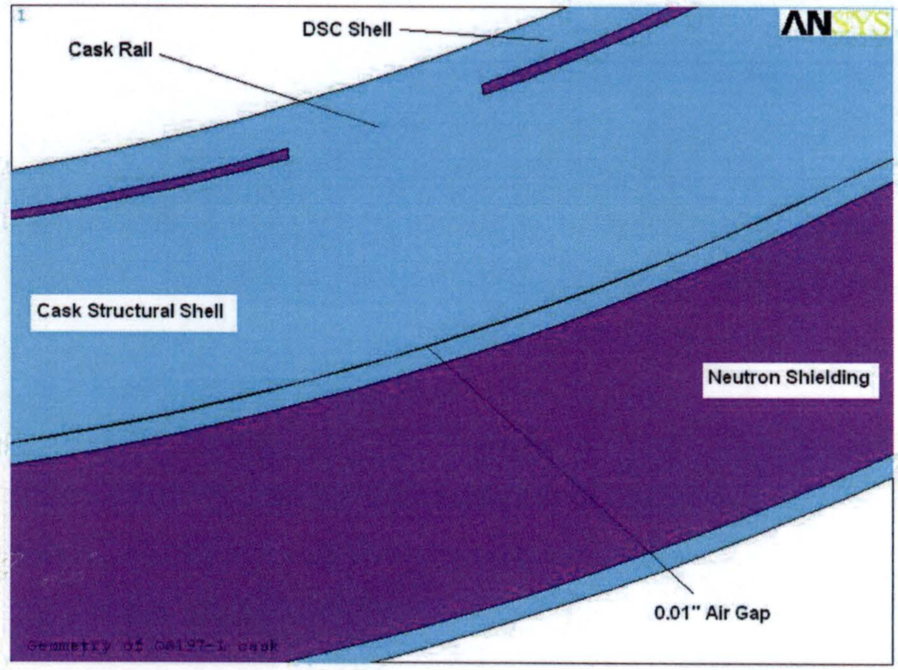


Figure W.4-12
OS197L TC ANSYS Model

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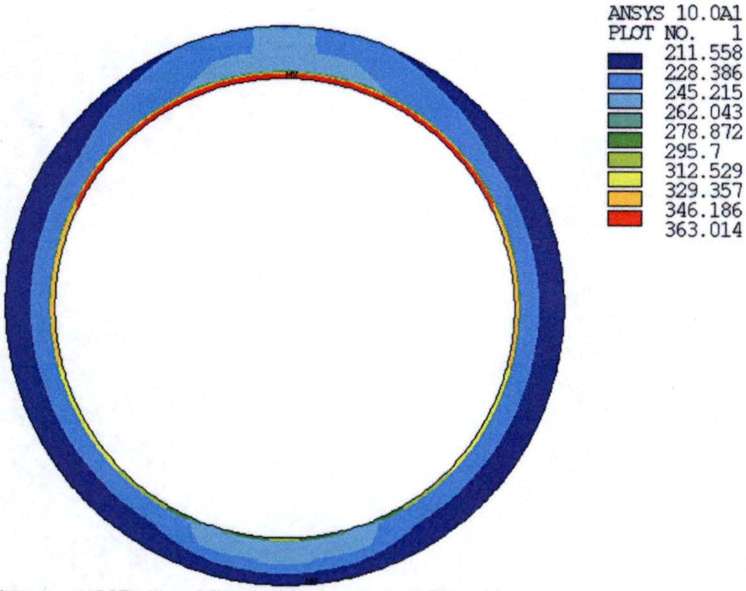
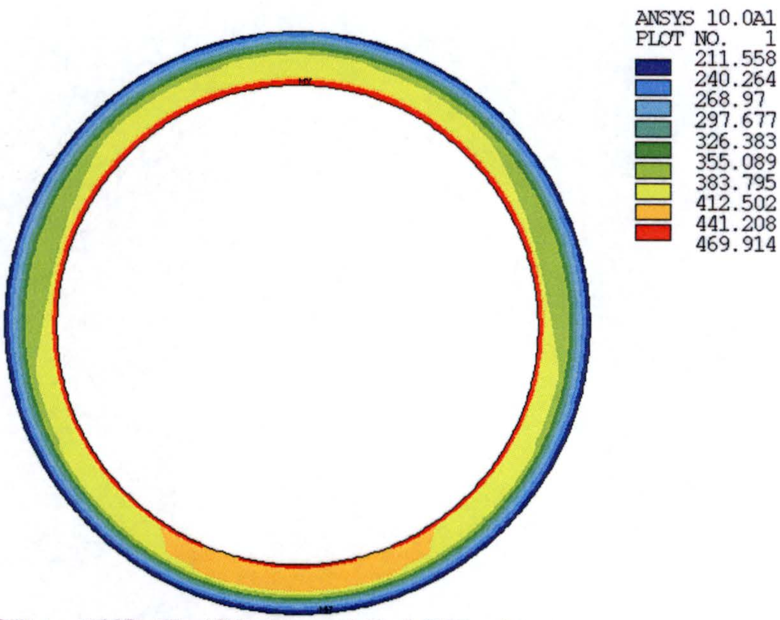


32PT in OS197L TC, 13kW, Off-Normal @ 117F ambient



32PT in OS197L TC, 13kW, Loss of NS @ 117F ambient

Figure W.4-13
Temperature Distribution on OS197L TC with *32PT DSC (13 kW)* and Supplemental Shielding, T_{amb}=117°F, Insolation

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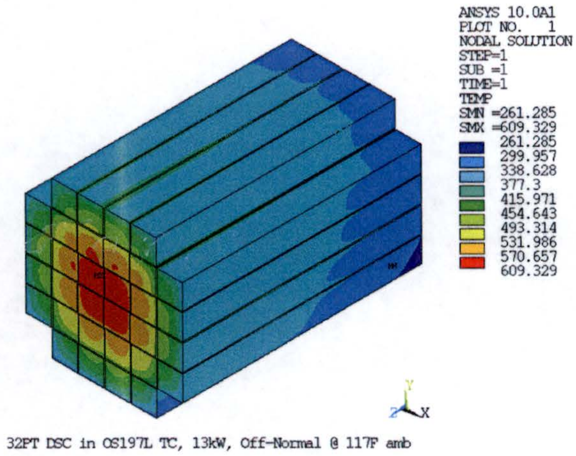
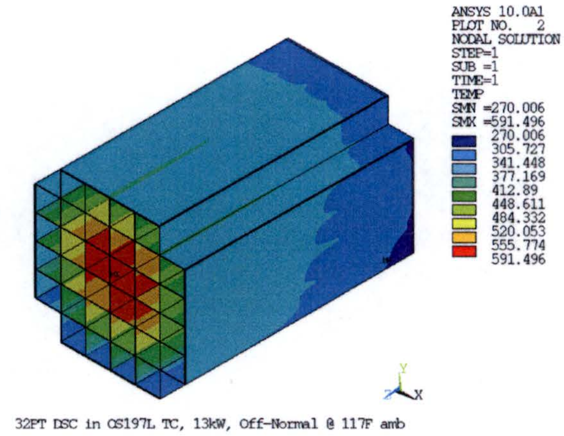
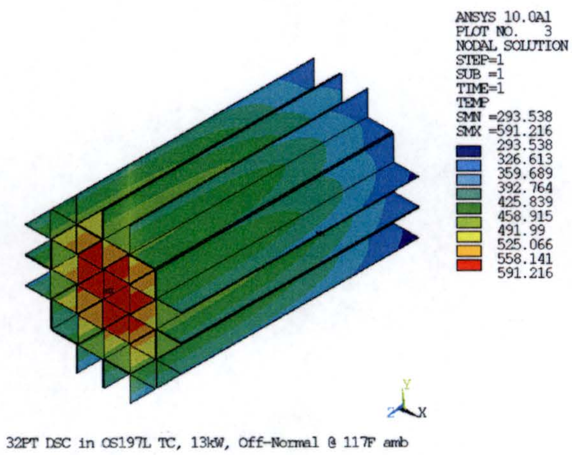


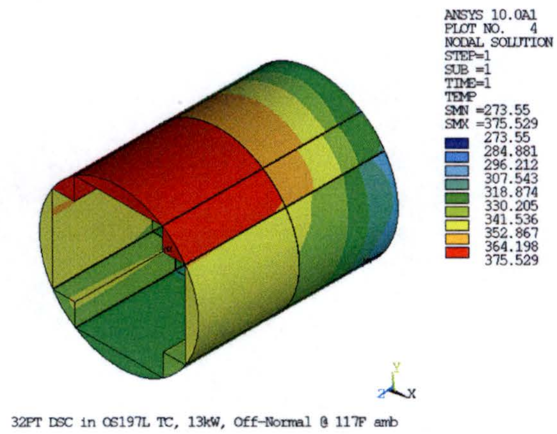
Fuel Assemblies



Basket Grid



Basket Aluminum/Poison



Basket Rails

Figure W.4-14
Temperature Plots for 32PT DSC (*13* kW) in OS197L TC with Supplemental Shielding, T_{amb}=117°F, *Off-Normal Transfer Conditions*

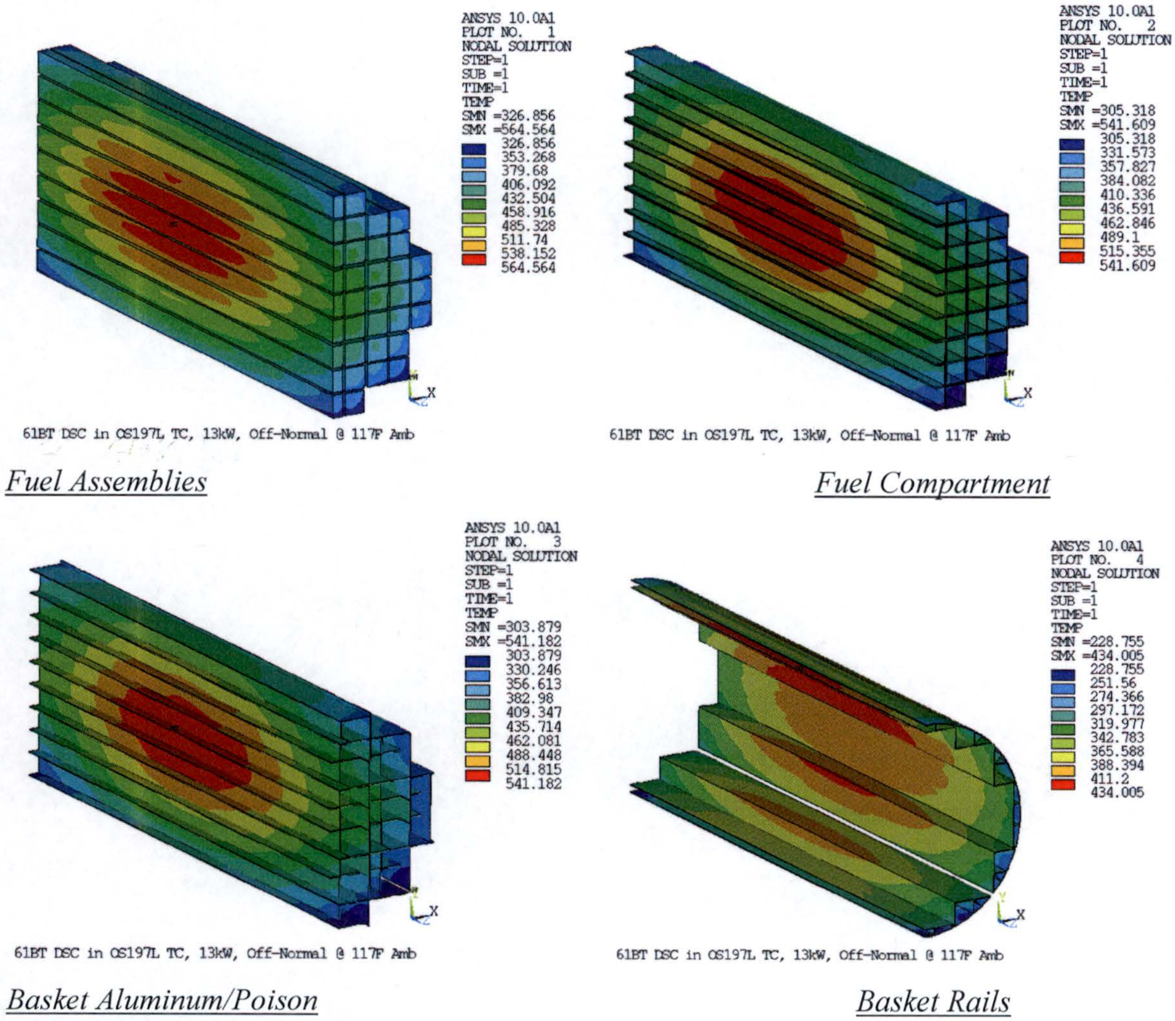


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Figure W.4-15
Temperature Plots for 61BT DSC (*13* kW) in OS197L TC with Supplemental Shielding, T_{amb}=117°F, *Off-Normal Transfer Conditions*

W.5 Shielding Evaluation

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This Appendix presents the shielding evaluation of the OS197L TC when used for fuel loading and transfer of *32PT and 61BT* DSCs.

The shielding analysis is performed for various configurations of the OS197L TC during loading and transfer operations containing a fully loaded 32PT or 61BT DSC with a maximum 13 kW/DSC heat load for PWR and BWR fuel assemblies, respectively. Bounding dose rates for the OS197L TC with the 32PT and 61BT DSC payloads are evaluated. The dose rates on and around the bare OS197L TC are dominated by primary gamma sources during normal, off-normal and accident conditions of loading and transfer. The results for normal operations demonstrate that exposures for OS197L TC activities with operational personnel present are bounded by OS197 TC exposures (remote crane operation is used and no personnel are present in the immediate vicinity of the OS197L TC while the cask is on the crane hook during normal operations and decontamination area shielding is used).

The physical characteristics of the intact and damaged spent fuel assemblies authorized for storage in the NUHOMS®-61BT DSC are as described in Appendix K.2, Tables K.2-1 and K.2-2 respectively. However, to minimize the dose consequences when using the OS197L TC for the loading and transfer of the 61BT DSC, the contents allowed shall meet the heat load zoning configuration requirements of Figure W.2-1 and the DSC heat load shall be limited to 12.0 kW or less. The intact and damaged BWR fuel assembly characteristics allowed for storage in the NUHOMS®-61BT when using the OS197L TC are summarized in Table W.2-2 and Table W.2-3, respectively. Tables W.2-4 and W.2-5 provide the fuel qualification tables (FQTs) for the Zone 1 and Zone 2 fuel assemblies, respectively.

The physical characteristics of the intact spent fuel assemblies, with or without control components (CCs), authorized for storage in the NUHOMS®-32PT DSC are as described in Appendix M, Table M.2-1. However, to minimize the dose consequences when using the OS197L TC for the loading and transfer of the 32PT DSC, the contents allowed shall meet the heat load zoning configuration requirements of Figure W.2-2 and the DSC heat load shall be limited to 13.0 kW or less. The intact PWR fuel assembly characteristics allowed for storage in the NUHOMS®-32PT when using the OS197L TC are summarized in Table W.2-6. Tables W.2-7 and W.2-8 provide the FQTs for the Zone 1 and Zone 2 fuel assemblies, respectively.

The bounding dose rates for normal and accident conditions of transfer are due to radiological sources specified in Section W.5.2. The sources are shown in Table W.5-1 and Table W.5-2. These tables represent radiological sources at burnup and enrichment combinations from 32PT and 61BT DSC transfer FQTs.

The FQTs for the 61BT DSC when using the OS197L TC are presented in Appendix W.2, Table W.2-4 and Table W.2-5. The FQTs for the 32PT DSC when using the OS197L TC are presented in Appendix W.2, Tables W.2-7 and W.2-8. These tables assure that the total dose rate during normal condition of transfer on the bare cask surface does not exceed the calculated maximum dose rates (shown in Tables W.5-8 and W.5-11) as well as assuring that the maximum decay heat load per DSC is not greater than 13 kW/DSC for the 32PT DSC and 12 kW/DSC for the 61BT

DSC. These tables also provide assurance that single fuel assembly decay heat limits shown in Appendix W.2, Figure W.2-1 and Figure W.2-2 are not exceeded.

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W.5.1 Discussion and Results

A summary of the bounding maximum dose rates on and around the OS197L TC with the 61BT or 32PT DSC during loading and transfer operations during normal and accident conditions are shown in Table W.5-3 and Table W.5-4, respectively. The bounding maximum dose rates for various shielding configurations of the OS197L TC with the 61BT or 32PT DSC during the various operational evolutions for normal, off-normal and accident conditions, at various locations are shown in Table W.5-6 through Table W.5-14. The total dose rates shown in these tables correspond to the maximum dose rates for the various shielding configurations. A brief description of the various shielding configurations evaluated herein for various loading and transfer operations is provided in Figure W.5-1 and in Section W.5.4.10.

A discussion of the method used to determine the bounding source terms for this evaluation is included in Section W.5.2. The shielding material densities are given in Section W.5.3. The model specification and the method used to determine the dose rates due to 32 PWR or 61 BWR allowed fuel assemblies in the various OS197L TC design configurations with 32PT DSC and 61BT DSC payload is provided in Section W.5.4. The radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the spent fuel contents. The shielding evaluation is performed with the MCNP5 [5.2] code with the ENDF/B-VI cross section library.

W.5.1.1 Dose Rates Near Shielding Configurations Containing 32PT and 61BT DSC

In general, the intensity and shape of the radiation field distribution around a shielding configuration is determined by two factors: (1) the intensity and spatial distribution of radiological sources, and (2) the shielding properties and spatial configuration of shielding materials. Also note that the maximum of gamma and neutron radiation dose rates may occur at different locations. Therefore the maximum of total dose rate is not necessarily equal to the sum of the maximum gamma and maximum neutron dose rates.

The bounding radiological source terms employed herein are shown in Table W.5-1 and Table W.5-2 for the 32PT DSC and 61BT DSC, respectively. Such sources result in bounding dose rates from 32PT and 61BT DSC contents of the cask when compared with other radiological sources at various burnup, enrichment, and cooling time combinations shown in the FQTs. Neutron radiation dose rates near the OS197L bare cask containing a 32PT DSC bounding sources are bounding for the neutron radiation dose rates from the cask containing the 61BT DSC bounding sources.

The data presented in Tables W.5-6 through Table W.5-14 are based on MCNP calculated dose rates. There is always a statistical uncertainty in the results obtained using a Monte Carlo method like MCNP.

Because of axial symmetry of the shielding materials distribution on the side of the cask and the DSCs and in the distribution of radiological sources, it is convenient to consider dose rate *distribution along the cask side in a cylindrical coordinate system with an axis that coincides with the axis of the cask/DSC. In general, dose rate is dependent on all three coordinates: axial, radial and angular. Dose rate distribution along the side of the cask has an angular symmetry. The 32PT DSC has R45 and R90 solid aluminum rails on its periphery which provide noticeable shielding at the periphery of the DSC where the rails are present. There is no such shielding on the periphery of the 61BT DSC. Therefore, the primary gamma radiation (PGR) dose rate on and near the side of the cask has a pronounced angular dependence with the 32PT DSC. Further, the 32PT DSC can contain stronger radiological sources. As a result, maximum values near the cask surface occur when the cask is loaded with a 32PT DSC. Aluminum does not affect the shape of the neutron radiation dose rate distribution as dramatically as it does the PGR.*

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Two sets of dose rates near the bare cask at various axial and radial distances are calculated for the OS197L TC for both the 32PT and 61BT DSCs at normal and accident conditions. Normal conditions are modeled with a dry DSC and DSC/TC annulus with a water-filled neutron shield and accident conditions are modeled with a dry neutron shield. These dose rates are presented in Table W.5-6 through Table W.5-11. For the other shielding configurations described in Section W.5.4.7 (to determine dose rate distributions during the various operational evolutions with the OS197L TC), only bounding dose rates at distances up to 10 meters are presented. They are from a 32PT DSC payload of the cask. The dose rates for the various operational evolutions at distances greater than 10 meters do not have pronounced axial and angular variations and are presented in Table W.5-6 and Table W.5-7 for the 32PT DSC, and Table W.5-9 and Table W.5-10 for the 61BT DSCs.

W.5.1.2 Bounding Dose Rates as a Function of Distance

Dose rates as a function of distance for various shielding configurations of the OS197L TC under normal conditions of transfer are plotted on Figure W.5-2.

Figure W.5-2 displays five dose rate versus radial distance curves from the side for 4 shielding configurations: (1) bare cask, (2) cask with 2.5" thick steel shell; (3) the cask on a trailer platform with 2.5" thick inner top support skid supplemental shielding and without 3" thick outer top steel shielding; and (4) the cask ready for transfer (i.e., cask with both inner and outer top trailer shielding installed). Dose rates shown in Figure W.5-2 are bounding for the OS197L containing either a 32PT or a 61BT DSC.

Note that the support skid supplemental shielding is referred to as trailer shielding or trailer area shielding throughout this chapter.

The bare cask dose rate curve (curve 1) as a function of distance shown in Figure W.5-2 is bounding for all other configurations. It is included to provide a comparative understanding of the shielding effectiveness of the trailer shielding. These dose rates are based on results shown in Table W.5-6.

The dose rate curve (curve 2), as a function of distance corresponding to the "Cask with Additional 2.5" thick Steel Shell Only" shown in Figure W.5-2, is based on an MCNP model that includes a 2.5" thick steel cylindrical shell around the OS197L TC (see description "Pre-

Transfer" in figure W.5-1). This configuration is a simplified representation of the OS197L TC prior to the placement of the outer top trailer shielding. These dose rates are based on results shown in Table W.5-12.

The dose rate curves for the three remaining configurations correspond to the cask on the transfer trailer platform. Geometry of the modeled shielding configuration (only the inner top trailer shielding is modeled) is depicted on Figure W.5-5. These plots display the maximum dose rates in the horizontal direction from the side of the transfer package. Note that there is no shielding underneath the cask in the computational model, except for the 0.25" thick steel plate representing the trailer platform.

The dose rate curve (curve 3) as a function of distance corresponding to the "Cask on Trailer with 2.5" thk. Inner Top Trailer Area Shielding" is for the configuration where the cask is on the trailer platform where the 2.5" inner top trailer shielding is installed and the 3" thick steel outer top trailer shielding is not yet installed. The dose rate maximum is near the interface between the 2.5" thick inner top trailer shielding over the cask side and the 5.5" thick side trailer shielding (side shielding plates in Figure W.5-5). These dose rates are nearly identical to those shown in curve 2 because the maximum dose rates occur around the 2.5" thick shielding.

The dose rate curves for the two remaining configurations correspond to calculated maximums at a vertical elevation between the ISFI concrete pad and the trunnions.

The dose rate curve (curve 4) as a function of distance corresponding to the "Cask on Trailer with 2.5" thick Inner and 3.0" thick Outer Trailer Top Shielding" is the maximum dose rate at a vertical elevation between the bottom of the 5.5" thick trailer shielding along the side of the cask and trunnions level. These dose rates are based on results shown in Table W.5-13.

These dose rates are bounding for the cask in the decontamination area for the following reasons. First, these dose rates are for 5.5" thick steel shielding along the side of the cask, while the decontamination area shield thickness is 6.0". Second, the cask is in a horizontal position on the trailer platform with no shielding underneath the trailer platform modeled when calculating the dose rates. This causes substantial radiation streaming and scattering from the ISFSI concrete pad surface which are accounted for in the calculated dose rates. On the other hand, the cask in the decontamination area is in a vertical position and entirely surrounded with a 6" thick steel cylindrical shell.

The dose rate curve (curve 5) as a function of distance corresponding to the "Below Cask Support Skid when Cask is on Trailer with Inner and Outer Trailer Top Shielding" is the maximum dose rate at the vertical elevation between the ISFSI concrete pad and the cask support skid. As expected, this distribution shows fairly high dose rates at short distances. This is because of direct radiation from underneath the cask, where no shielding is present, except for the 0.25" thick steel plate representing the trailer platform, and radiation scattered from concrete. The dose rates shown in Table W.5-14 are employed to plot this curve. The dose rate distribution at this elevation is not significantly dependent on the outer top trailer shielding and therefore, the dose rates shown in Table W.5-14 are applicable.

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The dose rate distributions (curve 2 through curve 5) shown in Figure W.5-2 indicate that the maximum dose rates decrease to approximately 100 mrem/hr at a distance of 2 m from the cask transfer trailer.

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W.5.1.3 Bounding Dose Rates as a Function of Axial Distance

For the dose rates on top of the cask, two radial regions need to be considered: within and beyond the cask radius. These radial regions are depicted in Figure W.5-3. Dose rates beyond the cask radius are due to radiation from the side of the cask and through shielding on its top. Because there is substantially less shielding and sources are stronger (because of the in-core region) on the side, the contribution by radiation from the side is dominating. On the other hand, dose rates just over the ends of the cask at radial distances within the cask radius are contributed due to radiation through shielding over the ends of fuel assemblies. Further, the contribution to the top end dose rates due to scattering through the radial shielding layers, including the water neutron shield, is minimized at distances within the radius of the cask. It is assumed that the water neutron shielding on the side of the cask is lost during the accident. Therefore, accident and normal condition dose rates on top of the cask are not different when radial distances less than the radius of the cask are under consideration. Hence, the dose rates on the TC axis (r=0) and at r<= TC radius are applicable for both normal and accident conditions at "short" or relatively "closer" axial distances. Here the axial distance is the distance from the cask end when dose rate distribution becomes uniform (i.e., maximum and average dose rates are approximately the same). For the axial dose rates, this occurs at distances greater than 10.0 meters. Because shielding properties on the top of the various configurations are identical, dose rates at these axial distances are the same for various shielding configurations when radial distances less than the cask radius are considered.

When the cask is ready for transfer to the ISFSI, there is at least 12.1" of steel and 2.0" of NS3 over the top of the fuel assemblies for the 32PT DSC. The shielding of steel on top of the 61BT DSC is 1.5" thinner. At the same time, primary gamma and neutron radiation sources (in the two energy groups that are the dominant contributors to the PGR dose rates, 1.00 to 1.33 and 1.33 to 1.66 MeV) are by a factor of ~3.1 and 4.8 stronger in the central compartments of the 32PT DSC, respectively. It is established in the shielding analysis that such a difference makes the TC dose rates with the 32PT DSC source, bounding. Also, dose rates in Chapter M.5, Table M.5-5 are calculated with a model having the geometry depicted in Figure M.5-24. The model uses 2.45" of NS3 but only 9.75" of steel. The model distributes 32PT DSC design basis (DB) radiological sources from 1.2 kW/FA assemblies (conservatively) in 16 peripheral fuel compartments instead of 8 peripheral compartments at the corners of the DSC basket structural grid.

This makes the total top dose rates predicted with the shielding model of UFSAR Appendix M very conservative. For the OS197L TC, the dose rates on top of the cask depicted on Figure M.5-24 and presented in Table M.5-5 are bounding for the dose rates on top of the OS197L cask containing a 32PT DSC when radial locations bounded by the bare cask radius are under consideration. Therefore, dose rates on the top of the reference shielding configuration presented in Table M.5-5 are bounding for dose rates on the top of the OS197L containing a 61BT or 32PT DSC with contents bounded by source terms presented in Table W.5-1 and Table W.5-2, respectively.

W.5.2 Source Specification

The radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The computational model of the DB PWR fuel from Appendix M.5 is directly utilized, as appropriate, to calculate the bounding radiological source terms due to assemblies in the 12 central (Zone 1) and 20 peripheral (Zone 2) fuel compartments of the 32PT DSC shown in Figure W.2-2. Similarly, the computational model of the DB BWR fuel from Appendix T, Chapter T.5, is directly utilized to calculate bounding radiological source terms for the assemblies in the 48 peripheral (Zone 2) fuel compartments of the 61BT DSC shown in Figure W.2-1. The sources for the central 13 fuel compartments (Zone 1) of the DSC are unchanged and are obtained directly from Appendix K.5, Table K.5-7.

FQTs for various decay heat values are presented in Appendix M.2, Appendix P.2 and Appendix U.2 for PWR fuel assemblies, and Appendix T for BWR fuel assemblies. The minimum required cooling times in order not to exceed the desired decay heat limits were determined using the SAS2H\ORIGEN-S models of the corresponding DB fuel assemblies. Also, radiological source terms were obtained as a "side" product during the calculation for all burnup, enrichment and cooling time (BECT) combinations evaluated for the OS197L TC. Important observations regarding the radiological source terms and their applicability to the OS197L TC provide the foundation for the determination of the bounding radiological sources. These observations are summarized below.

- *The dose rates along the side of the OS197L TC are mainly dominated by assemblies located in peripheral fuel compartments. This is especially true when all the fuel compartments contain the same radiological sources. Because of that, it is desirable to keep "hot" assemblies closer to the cask axis in order to minimize dose rates.*

- *The PGR dose rates are dominated by the radiological sources in the 1.0 to 3.0 MeV energy groups.*

- *^{106}Rh ($T_{1/2}$=29.8 second) and ^{144}Pr ($T_{1/2}$=17.3 minutes) isotopes are the major contributors to the intensity of the PGR source term in the 2.0 to 3.0 MeV energy group. The contribution to the PGR dose rates by the sources in this energy range is more prominent at cooling times below 3.0 years. These isotopes are products from ^{106}Ru ($T_{1/2}$=1.015 years) and ^{144}Ce ($T_{1/2}$= 0.7805 years) decay, respectively.*

- *The PGR dose rate is dominated by radiological sources in the 1.00 to 1.66 MeV energy range at cooling times greater than 5.0 years. The contribution is greater than 70%. The intensity of the PGR source in this energy range is mainly due to the ^{60}Co ($T_{1/2}$=5.27 years) isotope.*

- *Both the intensity of the PGR source in the 1.00 to 1.66 MeV energy range and its fraction in the total PGR source intensity are important for this purpose. Any of the BECT combinations may be selected to yield the bounding PGR source terms if the difference in the calculated dose rates falls within an acceptable band of 10%. This approach is adequate for selecting a bounding source for assemblies in a group of fuel compartments*

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whose contribution to the total dose rate is substantially less than assemblies in other compartments.

- *A description of shielding for the bare OS197L cask can be found in various sections of this chapter. Normal and accident conditions dose rate for such a shielding configuration is essentially dominated by PGR sources. Therefore, one can use, without introducing significant conservatism, the largest neutron radiation source among considered BECT combinations.*

The methodology to calculate the FQTs for the bounding BWR assemblies is described in Appendix K, Chapter K.5 and Appendix T, Chapter T.5. The methodology to calculate the FQTs for the bounding PWR assemblies is described in Appendix M, Chapter M.5; Appendix P, Chapter P.5; and Appendix U, Chapter U.5. The term "bounding assemblies" means that radiological and decay heat sources from such assemblies are bounding for qualified BWR and PWR classes of assemblies. However, these FQTs provide only minimum required cooling times at various burnup and enrichment combinations such that the resulting decay heat per fuel assembly is below a specified maximum. The difference between the reference FQT methodology and that employed for the OS197L TC is that one also needs to ensure that the resulting dose rates are below a specified maximum limit (bare cask maximum dose rate below 10 rem/hr). Note that radiological sources at various cooling times for each decay heat FQT burnup and enrichment combination are also determined as part of the FQT methodology. This implies that the results of the SAS2H/ORIGEN-S evaluations employed to determine the FQTs for the bounding BWR and PWR fuel assemblies for a variety of decay heat values and BECT combinations can be utilized for the OS197L TC evaluations. Additional observations can be made from radiological sources data related to the decay heat FQTs generated in these calculations and summarized as follows:

- *For all the burnup and enrichment combinations presented in a decay heat FQT, the bounding neutron radiation source occurs at a combination at the lowest enrichment and at the highest burnup. This statement can be easily verified by simply looking at a set of neutron radiation sources relevant to any FQT presented in the above referenced appendices.*

- *For a set of burnup and enrichment combinations with cooling times greater than a certain value in an FQT for a given decay heat restriction, the bounding PGR source occurs at the lowest enrichment, and the highest burnup and the lowest cooling time combination in that set. This is illustrated by the following example.*

For the 0.40 kW/PWR FA FQT presented in Appendix W.2, Table W.2-8, set the minimum threshold cooling time at 10.0 years. This ensures that all the entries in the FQT can be included in this example. Since the minimum cooling time is greater than 5 years, the contribution to the total dose rates due to PGR sources are dominated by the source terms in the 1.00 to 1.66 MeV energy range. The burnup and enrichment combinations with the minimum cooling time are selected for further evaluation thereby limiting the search to within those combinations with cooling times equal to 17.5 years (minimum cooling time to ensure that the dose rates are below the limits). For these combinations, the maximum burnup is 24 GWd/MTU and the minimum enrichment is 1.1 wt. % U-235. Therefore, the radiological source associated

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with 24 GWd/MTU and 1.1 wt. % can be considered as bounding because, (1) it has the strongest intensity in the 1.00 to 1.66 MeV energy range that contributes the most to the PGR dose rate on the surface of the OS197L TC, and (2) the intensity of the PGR source in the 1.66 to 3.0 MeV energy range is not substantially (greater than 19%) less than the intensity in the same energy groups at 1.2 and 1.3 wt. % enrichments. This also illustrates that when cooling times are greater than 15 years (approximately 3 half-lives for Co-60), the variation in the source strength with enrichment is not as pronounced as it is at lower cooling times, particularly at low to moderate burnups.

All the observations summarized herein are part of the methodology to determine a bounding radiological source and FQTs for on-site transfer. Since the bare cask dose rate is nearly completely dominated by the PGR source, the BECT combination resulting in the bounding radiological source can be identified as that resulting in the highest PGR source intensity in the 1.00 to 1.66 MeV range. By placing a restriction on the dose rate at the side of the cask, one can search for a desired combination through successive iterations by considering the sources at cooling times equal to some threshold value. Based on the discussion in the previous section, the bounding combination will be the highest burnup and the lowest enrichment at a cooling time equal to the threshold value. The iteration is terminated when the combination resulting in the maximum dose rate less than or equal to the dose rate limit is obtained for all the entries in the FQT. Since the intensity of a gamma radiation source in the 1.00 to 1.66 MeV energy range at cooling times greater than 5.0 years is mostly dominated by ^{60}Co ($T_{1/2}$=5.27 years) and the majority of the dose rate on the side of the cask is from assemblies in peripheral fuel compartments, the educated guess for a minimum threshold time can be easily obtained after calculating the maximum dose rate from any radiological source in a set of peripheral fuel compartments at cooling time greater than 5.0 years that is generated as a "side product" when determining a decay heat FQT in the various appendices referenced above.

W.5.2.1 Methodology for Determination of Bounding Radiological Source Terms

The evaluation to obtain the bounding radiological sources can be started with identifying BECT combination(s) in the decay heat FQTs that result in the bare cask normal condition dose rates less than or equal to a predetermined limiting value of 10.0 rem/hr. After that a trial and error process is used to prove that either the calculated BECT(s) combination results in the bounding dose rate or adjust cooling times in the decay heat FQTs in a manner that assures that the radiological sources from the BECT become bounding, as shown in the following steps:

1. *Set an MCNP model for the OS197L bare cask containing 32PT and 61BT DSCs. Continue with the next 10 steps starting from the 32PT DSC payload of the cask. A description of the MCNP models is provided in Section W.5.4.7.*

2. *The models should allow one to determine the contribution to the maximum dose rate on the side of the cask due to sources in certain groups of fuel compartments. Because of the axial symmetry of the canister, it is convenient to group the compartments within certain radial zones (in this case, two radial zones).*

3. *Obtain the maximum of PGR dose rate on the side of the cask when all the fuel compartments contain the same radiological sources. Start from radiological sources that*

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result in the maximum PGR dose rate greater than the desired limit. Determine groups of fuel compartments from which the contribution to the maximum dose rate is not significant in comparison with the sources in peripheral compartments. Quantify their contribution. Such compartments can be designated to hold the "hotter" radiological sources.

4. *A contribution to the maximum dose rate on side of the cask due to sources in peripheral fuel compartments is also identified by this point. Estimate the intensity of the PGR source term strength in the 1.00 to 1.66 MeV range that is needed in order not to exceed the desired dose rate limit. Use an exponential decay formula with a decay constant relevant to ^{60}Co ($T_{1/2}$=5.27 years) to estimate the change in source term intensity as a function of cooling time.*

5. *Find a burnup enrichment and cooling time combination in existing decay heat FQTs that have intensity of the PGR source in the 1.00 to 1.66 MeV range that is the closest and bounded by the intensity evaluated in step 4.*

6. *Include the source terms in the MCNP models set up in step 1. Calculate the maximum dose rate value.*

7. *If the maximum dose rate obtained in step 6 is greater or substantially lower than the desired limit, repeat steps 4 through 6. Otherwise, proceed with the next step (step 8).*

8. *An acceptable BECT combination for assemblies in the peripheral fuel compartments generating sources resulting in dose rates on the side of the cask below the desired limit at the normal conditions is achieved by this step. Since the dose rate near the OS179L bare cask is nearly entirely dominated by the PGR source, it is conservative to use the absolute bounding neutron radiation source. That source occurs at the lowest enrichment and the highest burnup when considering BECT combinations relevant to decay heat FQTs of the appendices mentioned in the introduction to Section W.5.2.*

9. *Utilize the neutron and the PGR source obtained so far and repeat steps 6 and 7 until the maximum of the total (due to neutron and gamma radiation) dose rate on the side of the cask is below the desired limit. Then, proceed with the next step (step 10). Note, when dealing with the OS197L TC containing a 61BT DSC, ensure that the maximum dose rate is bounded by that obtained while performing a similar qualification of the radiological sources for the OS197L TC containing a 32PT DSC.*

10. *The intensity of the PGR source in the 1.00 to 1.66 MeV energy range at the bounding BECT combination for assemblies in peripheral fuel compartments that result in (when combined with the bounding neutron source) dose rates on the side of the cask below the desired limit is established at this stage.*

11. *Consider the BECT combinations that are intended for assemblies in peripheral fuel compartments. Adjust, if necessary, cooling times for burnup and enrichment combinations that generate greater intensity of the PGR source in the 1.00 to 1.66 MeV energy range than the bounding source obtained by step 10.*

All changes on this page are AMD 11

A similar sequence of steps is used for the OS197L TC bare cask configuration containing a 61BT DSC.

Finishing step 11 completes an adjustment of cooling times in select decay heat FQTs. The adjusted FQTs are referred to as the transfer FQTs. They assure that both the decay heat limit and a restriction for the maximum dose rate on the side of the OS197L bare cask at normal condition of transfer are satisfied. The decay heat FQTs are adjusted in such a manner that the maximum of the total dose rate on the surface of the OS197L bare cask containing a 32PT DSC is bounding for the cask containing a 61BT DSC at normal conditions of transfer. Therefore, the payload of the cask for the bounding shielding evaluation and the bounding radiological sources are identified. Also, normal conditions of transfer dose rates for the OS197L bare cask shielding configuration are calculated for 32PT and 61BT payloads after completion of step 9.

Note, in general, that there are plenty of options for an arrangement of radiological sources within fuel compartments of the DSCs contained in the cask that would result in the dose rates below the desired limit. This shielding analysis investigates the arrangement of the sources within fuel compartments grouped in two radial zones. It is assumed that loading of the radial zones is uniform, i.e., all the decay heat and radiological sources within each zone are the same.

Numerous iterations employing the MCNP models mentioned in step 1 above determined that the arrangement of fuel assemblies as shown in Chapter W.2, Figure W.2-1 for the 61BT DSC and Chapter W.2, Figure W.2-2 for the 32PT DSC result in maintaining the maximum dose rate on the side of the bare cask below the 10 mrem/hr limit under normal conditions. The fuel assemblies shown as Zone 1 in these figures are limited to a maximum decay heat of 0.60 kW/FA for the 32PT DSC and 0.30 kW/FA for the 61BT DSC. The fuel assemblies shown as Zone 2 in these figures are limited to a maximum decay heat of 0.40 kW/FA for the 32PT DSC and 0.17 kW/FA for the 61BT DSC.

The minimum required cooling times shown in Table W.2-4 ensure that the decay heat from Zone 1 FAs does not exceed the limit of 0.30 kW/FA for the 61BT DSC. The minimum required cooling times shown in Table W.2-5 ensure that the decay heat from Zone 2 FAs does not exceed the limit of 0.17 kW/FA for the 61BT DSC.

The minimum required cooling times shown in Table W.2-7 ensure that the decay heat from Zone 1 FAs does not exceed the limit of 0.60 kW/FA for the 32PT DSC. The minimum required cooling times shown in Table W.2-8 ensure that the decay heat from Zone 2 FAs does not exceed the limit of 0.40 kW/FA for the 32PT DSC.

Note that neutron shielding material on the side of the cask is assumed lost during accident conditions. Since implementation of the methodological steps in the current section results in bounding primary gamma and bounding neutron radiation sources, these are also bounding for accident conditions of transfer.

All changes on this page are AMD 11

W.5.2.2 Primary Gamma and Neutron Source Terms

The major contributors to the PGR source due to LWR fuel assemblies as well as uncertainties in calculating the intensity of the source are identified in Section P.5.2.1.3, Section T.5.2.1.2 and Section U.5.2.1.3.

W.5.2.2.1 Bounding Radiological Sources for FAs in 12 Zone 1 Fuel Compartments of the 32PT DSC

The radiological source terms are calculated when preparing the 0.60 kW/FA decay heat FQT in Table W.2-7. These radiological sources at cooling times greater than or equal to 5.0 years are employed for further analysis. Based on the discussion in the introduction to Section W.5.2 it is identified that the bounding source occurs at a burnup of 20 GWd/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 5.0 years. As can be observed from intensities, PGR source term from such a BECT combination are bounding because they yield the highest intensity in the 1.66 to 3.0 MeV range, and intensities in the 0.8 to 1.66 MeV range are approximately 3% lower than the absolute maximum of the PGR source intensity in that energy range. As stated in Section W.5.2, bullet item 4, the PGR source in the 1.00 to 1.66 MeV range contributes significantly to the PGR dose rate at cooling times greater than 5.0 years. The "weight" of the 1.0 to 1.66 MeV energy range in the total intensity of the PGR source is the largest at a burnup of 25 GWd/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 6.4 years. The absolute maximum of the total intensity of the PGR source, 3.33e+15 gammas per second per assembly, is at a burnup of 25 GWd/MTU, an enrichment of 1.6 wt. % U-235 and a cooling time of 6.0 years. The intensities of the PGR source in the most important energy groups are not very different for assemblies in the burnup range of 20 to 25 GWd/MTU,enrichment range of 1.1 to 1.9 wt. % U-235 and cooling times in the range of 5.0 to 6.5 years. Sensitivity MCNP runs showed that Zone 1 compartments with these sources contribute about 500 mrem/hr to the maximum dose rate on the surface of the bare OS197L TC containing a 32PT DSC. Radiological sources from 0.60 kW/FA assemblies are intended for Zone 1 fuel compartments in the 32PT DSC. Therefore, the choice of the exact BECT combinations to determine the radiological source terms for Zone 1 locations does not significantly affect the dose rates. Therefore, a hybrid source is utilized for Zone 1 locations. Intensities and spectrum of the PGR source in the bottom nozzle, plenum, and top nozzle are due to fuel assemblies with a burnup of 25 GWd/MTU, an enrichment of 1.6 wt. % U-235 and a cooling time of 6.0 years. Intensity of the source in the in-core region is also due to the same combination but the spectrum is conservatively "hardened" by assigning the fraction of the total source intensity at 2.0–2.5 MeV energy range obtained from a fuel assembly with a burnup of 20 GWd/MTU, an enrichment of 1.9 wt. % U-235 and a cooling time of 5.0 years. When used in MCNP calculations, this hardening decreases the frequency of the radiological source spectrum sampling in the most important energy range, 1.00–1.66 MeV, by 0.03% while increasing the sampling frequency of the 2.0–2.5 MeV energy range by 5 times. Therefore, the "hardened" spectrum for the incore region, the region which essentially dominates dose rates at various radial distances from the side of the cask, is bounding.

The combined intensity of the PGR "hybrid" source from all the axial exposure regions is 3.34e+15 gammas per second per assembly. As mentioned in Section W.5.2, bullet item 8, the bounding neutron source is due to a BECT combination at the lowest enrichment and the highest

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burnup when considering a set of the BECTs represented in the decay heat FQT. This source is due to a fuel assembly with a burnup of 45 GWd/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 27.2 years. Its intensity is 6.86e+8 neutrons per second per FA. The source terms are summarized in Table W.5-1. These source terms are bounding for the 32PT DSC DB fuel assemblies for loading in Zone 1 (Figure W.2-2) with an initial loading of 0.475 MTU.

The decay heat due to the fuel assembly parameters employed in the PGR source term calculations is 571 watts. The decay heat due to the fuel assembly parameters employed in the neutron source term calculations is 597 watts.

All changes on this page are AMD 11

W.5.2.2.2 Bounding Radiological Sources for FAs in Zone 2 Fuel Compartments of the 32PT DSC

The radiological source terms were generated when preparing the 0.40 kW/FA decay heat FQT in Table W.2-8. An iterative process was employed to determine the minimum cooling time of 17.5 years for the Zone 2 fuel assemblies. This ensures that the maximum total dose rate on the side of the OS197L TC containing a 32PT DSC is below the limit of 10 rem/hr.

The fuel assembly with a burnup of 29 GWd/MTU, an enrichment of 5.0 wt. % U-235 and a cooling time of 17.5 years results in source terms with the largest PGR source intensity and the largest fraction of the PGR source intensity in the 1.66 to 2.50 MeV energy range. The fuel assembly with a burnup of 29 GWd/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 19.3 years results in source terms with the largest fraction of the PGR source intensity in the 1.00 to 1.66 MeV energy range. Even though the total intensity of the PGR source intensity of this BECT combination is 21% lower than that of the previous combination, the intensity in the 1.00 to 1.66 MeV energy range is 35% higher. Therefore, this combination can be considered as resulting in the bounding PGR source for FAs in the peripheral fuel compartments. To provide more assurance that this combination provides the bounding PGR source, the spectrum in the in-core region is artificially hardened by employing the larger fraction of the total source intensity in the 1.66 to 2.5 MeV energy range from the 29 GWd/MTU, 5.0 wt. % and 17.5 years combination. When used in MCNP calculations, this hardening decreases the frequency of radiological source spectrum sampling in the most important energy range, 1.00 to 1.66 MeV, by 0.002% while doubling the sampling frequency in the 1.66 to 2.5 MeV energy range. Therefore, the "hardened" spectrum for the in-core region, the region which essentially dominates dose rates at various radial distances from side of the cask, is bounding.

The combined intensity of the PGR "hybrid" source from all the axial regions is 1.724e+15 gammas per second per assembly. As mentioned in Section W.5.2, bullet item 7, the bounding neutron source is due to a BECT combination at the lowest enrichment and the highest burnup when considering a set of the BECTs represented in the decay heat FQT. This source is due to a fuel assembly with a burnup of 45 GWd/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 47.3 years with an intensity of 3.34e+8 neutrons per second. However, it is slightly, ~1%, lower than the intensity (3.37e+8 neutrons per second) of the fuel assembly at a burnup of 44 GWd/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 45.1 years. This difference is within the uncertainty of the SAS2H\ORIGEN-S sequence of the SCALE [5.1] package for calculation of radiological source terms, however the higher source term is employed as bounding.

The source terms for the DB fuel assembly for the 32PT DSC with an initial loading of 0.475 MTU are summarized in Table W.5-1. As stated in Section W.5.2, bullet item 4, the dose rate near the OS197L TC is largely dominated by the PGR source and the dose rate due to that source is mostly contributed to by the 1.00–1.66 MeV energy group. The intensity of the PGR source at 29 GWd/MTU, 1.1 wt. % and 19.3 years combination BECT in this energy group is 4.572E+13 gammas per second per FA. To assure that sources in Table W.5-1 are bounding for assemblies designated as Zone 2 in Figure W.2-1, cooling times for 0.40 kW/FA decay heat FQT presented in Table W.2-8 are adjusted such that the intensity of the PGR source terms in the 1.00–1.66 MeV energy range for all the BECT combinations does not exceed 4.572E+13 gammas per second per FA. Such an adjustment in the cooling times affects combinations with burnups less than or equal to 28 GWd/MTU. Adjusted cooling times are shown in Table W.2-8.

All changes on this page are AMD 11

In summary, the bounding PGR source for the 20 Zone 2 locations within the 32PT DSC is based on a 32PT DB FAs with 0.475 MTU at a burnup of 29 GWd/MTU, an enrichment of 1.1 wt. % U-235, with a cooling time of 19.3 years generating 0.398 kW/FA of decay heat. However, the spectrum in the in-core region is artificially hardened by assigning a larger fraction to the intensity in the 1.66 to 2.5 MeV energy range (corresponding to a fuel assembly with a burnup of 29 GWd/MTU, an enrichment of 5.0 wt. % U-235, and a cooling time of 17.5 years). The SAS2H input file for the in-core region at a burnup of 29 GWd/MTU and an enrichment of 1.1 wt. % U-235 is included in Section W.5.6.1.

W.5.2.3 <u>*Bounding Radiological Sources for FAs in Fuel Compartments of 61BT DSC*</u>

As mentioned in Section W.5.2.1, the dose rates on the side of the bare OS197L TC do not exceed the limit of 10 rem/hr when the central fuel compartments (designated as Zone 1 in Figure W.2-1) contain 61BT DSC DB sources if the peripheral fuel compartments (designated as Zone 2 in Figure W.2-1) contain radiological sources from assemblies generating less than 0.17 kW/FA.

The DB gamma radiation source terms are based on 27 GWd/MTU burnup, an initial enrichment of 2.00 wt. % and a 5 years cooled 61BT DB BWR assembly. The source terms are shown in Table K.5-7 of the UFSAR and replicated in Table W.5-2. The DB neutron source term strength is 1.427e+8 neutrons per second per assembly. It is due to 35 GWd/MTU burnup, an initial enrichment of 2.65 wt. % and an 8 years cooled DB fuel assembly. The MCNP built-in Cm-244 fission spectrum is utilized in the MCNP models for the current analysis.

Fuel qualification for 61BT FQT for Zone 1 assemblies is provided in Chapter W.2, Table W.2-4 (identical to that shown in Appendix K, Table K.2-11). It was determined through multiple iterations that radiological sources in fuel compartments designated as Zone 2 in Figure W.2-1 due to 40 GWd/MTU, 1.8 wt. % and 37.0 years cooled (DB BECT) assemblies result in dose rates on the side of the OS197L TC below 10.0 rem/hr when the Zone 1 fuel compartments contain 61BT design basis radiological source terms. On the other hand, the largest cooling time in the 61BT DSC FQT based on the decay heat restriction is 16.0 years and is due to a fuel assembly with a burnup of 40 GWd/MTU and an enrichment of 1.8 wt. % U-235. To assure that radiological source from DB BECT is bounding for the 48 Zone 2 fuel compartments of the 61BT DSC, cooling times in Table W.2-4 are adjusted. The adjustment is implemented in such a manner that the intensity of the PGR source in the 1.0–1.66 MeV energy range is bounded by the *corresponding intensity at DB BECT, 3.118E+12 gamma/(sec*FA). Adjusted cooling times are shown in Chapter W.2, Table W.2-5.*

The bounding radiological source terms for the Zone 2 locations of the 61BT DSC used in the shielding evaluation are shown in Table W.5-2. Note that the neutron radiation source corresponds to 35.0 years of cooling, not 37.0 years. Since the dose rate is dominated by the PGR source, this adds additional conservatism in the evaluated dose rates.

The SAS2H input file for the in-core region at a burnup of 40 GWd/MTU and an enrichment of 1.8 wt. % U-235 is included in Section W.5.6.2.

W.5.2.4 <u>Spectral Distributions of Neutron Source Terms and the Main Contributors</u>

The major contributors to the neutron radiation source from LWR FAs as well as uncertainties in calculating the intensity of the source are identified in Section P.5.2.2, Section T.5.2.2 and Section U.5.2.2.

The fixed source spectrum in MCNP is assumed to follow a ^{244}Cm spontaneous fission spectrum for all of the calculations in this chapter. This approach to neutron dose rate calculations is identical to that employed in Appendix P, Section P.5.2 for the MCNP dose rate evaluation of the 24PTH DSC. It is based on the following relationship:

$$p(E) \sim \exp(-E/a)\sinh(bE)^{1/2}$$

The input parameters: a=0.906 MeV and b=3.848 MeV^{-1}, as given in the MCNP manual Volume I (Appendix H) [5.2].

W.5.2.5 <u>Axial Peaking</u>

The axial peaking factors for both neutron and gamma sources in PWR fuel utilized herein are directly obtained from those utilized in the 32PT DSC shielding evaluation. These factors are shown in Appendix M, Table M.5-15.

The axial peaking factors for both neutron and gamma sources in BWR fuel are provided as a function of active fuel height in Appendix K, Section K.5.2.3. The same peaking factors are used in the MCNP analysis presented herein. These factors are directly applied to MCNP source input for the fuel region. More details about treatment of the peaking factors in the MCNP models are provided in Section T.5.2.3.

W.5.3 <u>Material Densities</u>

The material masses given in Appendix M, Table M.5-6 for the fuel are used to calculate material densities for in-core, plenum, top, and bottom regions of fuel assemblies inside the 32PT DSC. The material densities utilized in the MCNP calculations are shown in Appendix M, Table M.5-19.

All changes on this page are AMD 11

The material masses given in Appendix T, Table T.5-6 for the fuel are used to calculate material densities for in-core, plenum, top, and bottom regions of fuel assemblies inside the 61BT DSC. The material densities utilized in the MCNP calculations are shown in Appendix T, Table T.5-19.

Densities for miscellaneous materials like air, water, aluminum, carbon steel, stainless steel, etc. are obtained from Appendix M, Table M.5-16.

W.5.4 *Shielding Evaluation*

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from fuel assemblies in 32PT and 61BT DSCs within the OS197L TC are calculated with the MCNP Code [5.2] at various locations on and around the various evaluated shielding configurations.

The following shielding evaluation discussion specifically addresses the NUHOMS® 32PT and 61BT DSC in an OS197L TC using the design-basis source terms described in Section W.5-2.

W.5.4.1 *Computer Program*

MCNP [5.2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP very versatile and easy to use include a powerful general source, an extensive collection of cross-section data, and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems. MCNP was employed to take advantage of its mesh tallies capabilities in calculating dose rates distributed over the surface of the TC.

W.5.4.2 *Spatial Source Distribution*

The source components are:

- *The neutron sources due to the active fuel region,*
- *The gamma source due to the active fuel region,*
- *The gamma source due to the plenum,*
- *The gamma source due to the top region, and*
- *The gamma source due to the bottom region.*

Axial peaking is accounted for in the active fuel region by inputting an axial shape, as discussed in Section W.5.2.5.

All changes on this page are AMD 11

W.5.4.3 Cross-Section Data

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP code [5.2]. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made to account for secondary gamma (n,γ) radiation. All of the OS197L TC dose rate calculations account for the dose rate due to secondary gamma radiation.

W.5.4.4 Flux-to-Dose-Rate Conversion and MCNP Tallies

The flux distribution calculated by the MCNP code is converted to dose rates using flux-to-dose rate conversion factors from ANSI/ANS-6.1.1-1977 [5.4] given in Appendix P, Table P.5-19. The same flux-to-dose rate conversion factors have been employed in the 32PT and 61BT shielding analysis with the OS197 TC documented in Appendix M.5 and K.5, respectively.

Dose rates are computed at various distances from the cask in various shielding configurations described in Section W.5.1.2. Mesh tallies calculate neutron, primary and secondary gamma radiation dose rate distributions at various distances from the side and ends of the configurations. Cylindrical and rectangular mesh types are used. Locations of mesh nodes are defined either in cylindrical or rectangular Cartesian coordinate systems.

The rectangular mesh tallies are employed to obtain dose rate distributions at various horizontal distances from the side of the shielding configurations corresponding to the cask on a trailer platform. The Z-axis of the rectangular coordinate system is along the cask axis. The X-axis is on an imaginary plane through the cask axis and trunnions, perpendicular to the cask axis. The XZ plane is a horizontal plane and the Y axis runs in vertical elevation when the cask is in the transfer position. Rectangular (Z-Y) mesh tallies are used to calculate spatial distributions over 5.5" thick shielding plates on the trailer skid side and down to the ground below the trailer. Y=0 corresponds to the TC axis. The surface of the ISFSI concrete pad or ground level is at Y= -263.91 cm. Y=-140.08 cm corresponds to the bottom of the 5.5" thick shield plate along the TC side on the trailer and Y=-157.23 cm is related to the top of the 0.25" thick plate on top of the trailer. There are three distinct Y regions for segmentation: (1) [0 -140.08] cm- along the 5.5" thick plate on the side of the trailer skid; (2) [-140.08, -157.23] cm, just over the 6.75" wide gap between the bottom of the 5.5" thick plate along the TC side and the top of the 0.25" thick plate on the top of the trailer platform (see Figure W.5-5); (3) [-157.23, 263.91] cm slightly over the 42" wide clearance between the top of the trailer platform and the surface of the ISFSI concrete pad.

Because the dose rate around the cask is the highest along the cask side, the cylindrical (angular-axial) mesh tallies along the side of the cask between its ends were also employed. The cylindrical (angular-axial) mesh is used for determining the dose rate distribution along the cask side between the ends at various radial distances from the side. The Z axis of the cylindrical coordinate system coincides with the cask axis. The axial coordinates of the mesh nodes are measured from the bottom end. The axial distance between nodes of the cylindrical mesh is about 30 cm. The angular coordinate is measured in counter-clockwise direction from an imaginary plane through the cask axis and trunnions, which is the XZ plane in the rectangular coordinate system.

All changes on this page are AMD 11

A set of mesh tallies with various segmentation is used. The fine segmentation is used for dose rates at close distances and the regions where radiation streaming is expected. A segment size for the fine segmentation is about 30x30 cm but even finer Y segmentation, about 8 cm, is used for the Y region just over the 6.75" wide gap between the bottom of the 5.5" thick plate along the TC side and the top of the 0.25" thick plate on top of the trailer platform. When the distance from the side increases, the dose rate distributions become uniform. One can use coarse or no segmentation at all. This is true for distances beyond 50.8 meters.

W.5.4.5 Methodology

The MCNP computer code was utilized to analyze shielding performance of the cask in various shielding configurations. MCNP allows for explicit 3-D modeling of any shielding and source configuration. The methodology used herein is summarized below.

1. *Sources are developed for all fuel regions using the source term data described in Section W.5.2. Source regions include the active fuel region, bottom end fitting (including all materials below the active fuel region), plenum, and top end fitting (including all materials above the plenum region).*

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

3. *The 3-D Monte Carlo code MCNP is used to calculate dose rates on and around the OS197L TC loaded with the bounding, from a shielding standpoint, fuel and DSC designs. The MCNP code is selected because of its ability to handle thick, multi-layered shields and account for streaming through the TC/DSC annulus and TC neutron shielding using 3-D geometry.*

4. *MCNP results are used to calculate occupational and offsite exposures (see Chapter W.10).*

5. *MCNP models are also generated to determine the effects of off-normal and accident scenarios, such as loss of cask neutron shield and the support skid supplemental shielding for the OS197L TC.*

6. *The relative error in the total dose rate is calculated using the square root of the sum of the squares method as shown below:*

   $\sigma_{(n+\gamma)} = [(\sigma_n * D_n)^2 + (\sigma_\gamma * D_\gamma)^2]^{1/2} / (D_n + D_\gamma)$

   where

   σ_n *and* σ_γ *are the MCNP calculated relative errors in the neutron and gamma dose rates, respectively, and* $\sigma_{(n+\gamma)}$ *is the relative error in the total dose rate*

   D_n *and* D_γ *are the MCNP calculated neutron and gamma dose rates, respectively, and* $(D_n + D_\gamma)$ *is the calculated total dose rate*

All changes on this page are AMD 11

All changes on this page are AMD 11

W.5.4.6 Assumptions in MCNP Calculations

The following general assumptions are used in the analyses. Some of these assumptions are generic in nature and are similar to those employed to calculate the dose rates with systems in Appendix P, T and U.

W.5.4.6.1 Source Term Assumptions

- *The primary neutron source in LWR spent fuel is the spontaneous fission of* ^{244}Cm*. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables,* ^{244}Cm *represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein and is assumed to follow the* ^{244}Cm *fission spectrum provided in Section W.5.2.4.*

- *Due to large cooling times from a fuel qualification standpoint, the BECT combinations with bounding intensity of the PGR source in the 1.00–1.66 MeV energy range provide for the design basis source terms.*

- *Surface gamma dose rates are calculated for the TC surfaces using the actual photon spectrum applicable for each case.*

W.5.4.6.2 OS197L TC Dose Rate Analysis Assumptions

- *The 32PT and 61BT DSC models in MCNP include features like the basket structure and fuel compartments of the 61BT and 32PT DSCs and solid aluminum peripheral rails for the 32PT DSC.*

- *The borated neutron absorber sheets are modeled as aluminum.*

- *Axial peaking factors assumed as described in Section W.5.2.5.*

- *Fuel is homogenized within the fuel assembly perimeter, although the baskets of the DSCs are modeled explicitly.*

- *Axial dose rates during normal conditions of operation are discussed in Section W.5.1.3. The results with the OS197 TC are bounding to the OS197L TC because there is no change in the axial shielding design in the OS197L TC compared to the OS197 TC. In addition, the axial dose rates calculated with the OS197 TC are due to stronger radiological sources and include conservatism in the determination of OS197 TC dose rates using the 32PT and 61BT DSCs.*

- *All normal condition operations require water to be present in the neutron shielding. All calculations under normal conditions are performed with a water-filled neutron shield. Dose rate results (if any) calculated with a dry neutron shield are therefore conservative.*

- *For the design basis accident case, the cask neutron shield (water) along with the support skid supplemental shielding is assumed to be lost.*

W.5.4.7 Summary of the Calculational MCNP Models

All changes on this page are AMD 11

Explicit computational models containing a 32PT DSC for various shielding configurations are utilized. A computational model for the bare cask containing a 61BT DSC is also developed. The MCNP model of the 61BT DSC is taken directly from the MCNP model used for the bounding shielding evaluation in Section T.5. The numbering convention for cells, surfaces, and materials was modified to that used in MCNP models for the 32PT DSC. Dose rates from the bare cask containing the 32PT DSC were also calculated. Because dose rates at radial distances greater than 10.0 meters are low, dose rates from these two sets can be used as a conservative representation of the radiation field around various shielding configurations described in this chapter. The radial dose rate results for the bare cask configuration containing the 32PT DSC are bounding for the 61BT DSC except at locations near the top of the TC. This is inferred by comparing the dose rates shown in Tables W.5-8 and W.5-11 at axial distances greater than 500 cm. This is due to the differences in the basket design between the 61BT and the 32PT DSCs. However, for all configurations where the maximum dose rates are employed for comparison, the dose rates calculated with the 32PT remain bounding. Where appropriate, the results of shielding calculations with both the 61BT and 32PT DSCs are described.

The following is a summary of the various MCNP models utilized to obtain the results for the various loading and transfer configurations. All the models employ quarter-symmetry except those which model the OS197L TC in the trailer prior to placement of the top trailer shield where a half-symmetry model is employed. A brief description of the various shielding configurations evaluated herein is provided in Figure W.5-1. The basket and source layout geometry for the quarter-symmetry MCNP models are based on sketches shown in Figure W.5-1, Figure W.2-1 and Figure W.2-2. The bare cask model includes models of both 32PT and 61BT DSCs. Bounding radial dose rates for other shielding configurations are calculated using considerations presented at the end of Section W.5.1.1. All the dose rate results shown are radial dose rates unless explicitly specified as axial. For most configurations of loading and transfer the axial (ends of the TC) dose rate results from the OS197 calculations are directly applicable. The dose rate results as a function of distance for various shielding configurations of the OS197L TC are also shown in Figure W.5-2. The various shielding configurations for which the dose rates are determined are described below.

1. *OS197 TC under normal conditions with water in the neutron shield using the MCNP calculational methodology. The results for this case are shown in Table W.5-3 in the row corresponding to the OS197 TC transfer cask configuration (Case #3-2). Dose rates are due to 32PT DSC design basis radiological sources. The results for this configuration illustrate the conservatism inherent in the results documented for the 32PT DSC in the UFSAR (Case #3-1).*

2. *OS197L TC without any supplemental shielding and with water in the neutron shield. The DSC is assumed to be dry. This configuration is expected during the remote handling operations when the cask is lowered into the transfer trailer from the decontamination area cask shield after the DSC welding and sealing operations. The summary results are shown in Table W.5-3 in the row corresponding to the OS197L TC bare cask transfer cask*

configuration (Case #3-3). Detailed results for this case are shown in Table W.5-6 for the 32PT DSC and Table W.5-9 for the 61BT DSC.

3. *OS197L TC without any supplemental shielding and without water in the neutron shield. This configuration conservatively bounds that expected during the remote handling operations when the cask is lowered into the transfer trailer during accident conditions. The summary results are shown in Table W.5-4 in the row corresponding to the OS197L TC bare cask transfer cask configuration (Case #4-5). The radial dose rate results for this case are shown in Table W.5-7 for the 32PT DSC and Table W.5-10 for the 61BT DSC. The axial dose rate results for this case are discussed in Section W.5.1.3 and geometric locations for axial dose rate calculations are shown in Figure W.5-3.*

4. *Configuration is similar to 3, above, except that both the inner and outer liners of the neutron shielding are absent. This configuration provides for the worst case accident for shielding purposes. The results for this case can be inferred from Table W.5-4, using the first note under the table.*

5. *OS197L TC with 2.5 inches of supplemental shielding and no water in the neutron shield during accident conditions. The maximum of neutron, gamma and total dose rate on the side surface of this configuration are bounded by 1466, 540 and 1543 mrem/hr, respectively. The summary results are presented in Table W.5-4 in the row corresponding to the OS197L TC (with supplemental inner top trailer shielding only) transfer cask configuration (Case #4-4). The axial dose rates for this configuration are discussed in Section W.5.1.3. Geometric locations for axial dose rate calculations are shown in Figure W.5-3.*

6. *OS197L TC with the inner top supplemental trailer shielding (2.5 inches of shielding in the top and 5.5 inches of shielding in the side of the trailer) only and water in the neutron shield (see the "Section B-B" view of Figure W.5-5). The calculational model does not consider any shielding beneath the cask in the horizontal orientation except for the trailer platform (see notes 3 and 4 for Table W.5-3). This configuration is expected prior to the installation of the additional 3 inches of the outer top supplemental trailer shielding prior to the commencement of transfer operations. The MCNP model details including geometrical and material descriptions are shown in Figure W.5-5. The summary results are shown in Table W.5-3 in the row corresponding to the OS197L TC (without the outer top supplemental trailer shielding) transfer cask configuration (Case #3-5). The results that bound this case are shown in Table W.5-12. The bounding axial dose rate results applicable for this configuration are discussed in Section W.5.1.3.*

7. *OS197L TC with 5.5 inches of supplemental trailer shielding and with water in the neutron shield. This is the transfer configuration under normal conditions and bounds that during decontamination operations since credit is taken for 5.5 inches of supplemental shielding instead of 6.0 inches for the decontamination area cask shield. The summary results are shown in Table W.5-3 in the row corresponding to the OS197L TC (with decontamination area cask or supplemental trailer shielding) transfer cask configuration (Case #3-4). The results for this case are shown in Table W.5-13.*

All changes on this page are AMD 11

8. *Configuration is similar to 7, above, except that there is no water in the neutron shield. This represents a loss of neutron shielding accident during transfer operations with the support skid supplemental shielding present. The maximum of neutron, gamma and total dose rate on side surface of this configuration are bounded by 727, 134 and 791 mrem/hr, respectively. The summary results are presented in Table W.5-4 in the row corresponding to the OS197L TC (with supplemental inner and outer trailer shielding) transfer cask configuration (Case #4-3).*

The MCNP model to determine the dose rates shown in Table W.5-13 and Table W.5-14 are based on a geometry shown in the "Section B-B" view of Figure W.5-5. Three different dose rate distributions are obtained depending on the location of the dose rate tallies as described in Section W.5.4.4.

Radial dose rates calculated at an elevation near the cask axis provide an estimate of the dose rate distribution representative of the configuration with the inner top trailer shielding installed. These dose rates are also compared to the dose rates from a simplified model (results shown in Table W.5-12) in Figure W.2-1 (curve #2 and curve #3). This comparison indicates that the simplified model is sufficient for this purpose.

Radial dose rates calculated at an elevation below the cask axis and above the bottom of the skid platform provide an estimate of the dose rate distribution representative of the configuration with the inner and outer top trailer shielding installed. These dose rates are shown in Table W.5-13.

Radial dose rates calculated at an elevation between the top of the concrete ISFSI pad and the bottom of the skid platform provide an estimate of the dose rate distribution representative of the configuration with the inner and outer top trailer shielding installed. The dose rate distribution at this elevation is not significantly dependent on the outer top trailer shielding. Therefore, the MCNP model employed for this purpose is appropriate. These dose rates are shown in Table W.5-14.

Note that the bounding radial dose rates for the shielding configurations with water in the neutron shield are also plotted on Figure W.5-2 and discussed in Section W.5.1.2. The MCNP neutron input file for the 61BT DSC in the bare cask configuration is included in section W.5.6.3. The MCNP gamma input file for the 32PT DSC in the transfer trailer configuration is included in Section W.5.6.4.

W.5.4.8 Normal Condition Models

Various MCNP models are developed to perform the shielding evaluation of the 32PT DSC in the OS197L TC. Normal conditions include the 32PT DSC, the OS197L TC with the water filled neutron shield and 5.5 inches of supplemental trailer shielding. Due to the capability of modeling complex geometry in 3-D, several modeling conservatisms, originally employed in the 2-D DORT calculations (Section M.5 for 32PT DSC) are eliminated. However, the conservative estimates for axial end dose rates are still employed as discussed in Section W.5.1.3. The resulting MCNP model is also employed to determine the dose rates on and around the OS197 TC for comparison. These results are shown in Table W.5-3 (Case #3-1); see rows marked

All changes on this page are AMD 11

"UFSAR (Table M.5-5 and Section M.11.2.5.3)" and "OS197 TC." The computational models with the OS197L TC are described in the subsequent sections.

The neutron shield remains filled at all times and therefore any configuration involving an empty neutron shield is considered an accident condition.

W.5.4.8.1 <u>32PT DSC in OS197L TC</u>

Two three-dimensional MCNP models with quarter-symmetry are employed for shielding analyses of the 32PT DSC within an OS197L TC, one model for neutrons and the other for gammas. The z-axis in the MCNP models coincides with the axis of rotation of the cask and the 32PT DSC. The MCNP geometry of the DSC basket structure and the source representation is shown in Figure W.5-4. Note, the lattice cell (2 1 0) and (1 2 0) are loaded with 1.2 kW/FA assemblies and the other cells are loaded with 0.6 kW/FA assemblies when determining "OS197 TC" dose rates presented in Table W.5-3 (Case #3-2) and Table W.5-4 (Case #4-6). Those dose rates are due to 32PT DSC design basis radiological sources that are used in Appendix M, Section M.5 of the UFSAR. Select features within the cask and on its surface are neglected because they produce only localized effects and have minimal impact on operational dose rates. Examples of neglected features include the relief valves, clevises, and eyebolts. The TC trunnions are not explicitly modeled, however, a sensitivity evaluation on the effect of trunnion design is discussed in Section W.5.4.8.2. For the purpose of this evaluation, the trunnions represent areas of increased gamma shielding and are located in regions of relatively low importance for shielding.

In addition, a separate set of 3D MCNP models, similar to the calculational 2D DORT models (Appendix M, Section M.5 of the UFSAR) for the 32PT DSC in the OS197 TC, are also employed for comparison. These models provide a measure of the amount of conservatism present in the DORT models.

Design features relevant to the shielding analysis of the OS197L TC and 32PT DSC are modeled in MCNP. The cask shell is modeled with a thickness of 2.68", the neutron shield inner and outer shells are modeled with thicknesses of 0.26" and 0.19", respectively with a liquid water neutron shield thickness of 3.00". The effect of the two-piece neutron shield on the normal condition dose rates is discussed in Section W.5.4.8.3.

The supplemental trailer shielding is modeled as a half cylinder with full design thickness to determine transfer condition dose rates. The bounding results for this configuration are provided in Table W.5-13. In addition, calculations are performed to determine the dose rates as a function of distance using a transfer configuration model with only the inner top trailer shielding installed. The MCNP model description for this configuration is provided in Figure W.5-5. The bounding results for this configuration are shown in Table W.5-12.

W.5.4.8.2 <u>Solid One-Piece Trunnion Dose Rate Evaluation</u>

Analyses are performed to compare the effect of the solid steel trunnion design to the original trunnion design (multiple pieces) which used NS-3 neutron absorber to reduce neutron dose rates. The result of this analysis indicates that this change does result in an increase in neutron

All changes on this page are AMD 11

dose rate; however, since the majority of the dose rate contribution is gamma, the overall dose rate is reduced in the solid steel trunnion configuration. A comparison of the dose rates is provided in Table W.5-5.

In summary, the use of a one-piece trunnion reduces the total calculated dose rate by a factor greater than ten, thus providing a beneficial impact on occupational exposures.

W.5.4.8.3 Removable Two-Piece Neutron Shield Dose Rate Evaluation

The two-piece neutron shield provides the same level of shielding as the OS197 TC neutron shield. The water cavity thickness is unchanged. The outer shell of the OS197L TC neutron shield is slightly thicker than that used in the OS197 TC (0.25" versus 0.18"). The addition of the seam between the two halves would reduce gamma dose rate in the vicinity of the seam but would increase neutron dose rate due to less water in the vicinity. As discussed for the trunnion modification above, the cask surface dose rate is dominated by gamma. The presence of steel will result in a net reduction in the total dose rate in the vicinity of the seams.

The seam between the two halves of the neutron shield is 1.5 inches wide and is "filled" with 3 inches of steel instead of water. The calculational MCNP model did not explicitly include the seam between the two halves of the neutron shield. Instead, the neutron shield shell is modeled as if it was continuous. This is conservative as the region around the weld seams represents an area of dose rate depression due to superior gamma shielding. The justification for such a representation is provided below.

The maximum dose rate at the surface of the OS197L cask with water in the neutron shield, from Table W.5-3 (Case #3-3), is approximately 9,840 mrem/hr (320 mrem/hr neutron and 9520 mrem/hr gamma). The maximum dose rate at the surface of the OS197L cask in the vicinity of the seams, from Table W.5-4 (Case #4-4), is approximately 1550 mrem/hr (1470 mrem/hr neutron and 540 mrem/hr gamma). Note that the dose rate values shown herein are rounded up from those shown in Table W.5-3 and Table W.5-4. This dose rate is not calculated using an explicit model of the weld seams but calculated with an equivalent (conservative) model. For this model, the OS197L cask model includes no water in the annulus and neutron shield shell and also includes a 2.50" thick steel shell. This is a conservative representation of the weld seam region with no water and a thickness of 3.00" of steel. These results are conservative since the thickness of the seam is 3.00" instead of 2.50" employed in the model. These results demonstrate that there is a substantial dose rate reduction in the vicinity of the seams since the dose rate distribution on and around the cask is dominated (>95%) by gamma sources.

W.5.4.9 Accident Models

Accident condition models are those where the OS197L cask and its contents (32PT or 61BT DSC with design basis fuel) are modeled with loss of shielding arising out of hypothetical accident conditions. Loss of water in the neutron shielding is the most common consequence of these accidents. The accident condition MCNP models are similar to the normal condition MCNP models except that the water in the neutron shield is replaced with air.

All changes on this page are AMD 11

The radial dose rates as a function of distance for selected distances are summarized in Table W.5-4. The bounding axial dose rates are discussed in Section W.5.1.3. These accident configurations are described below:

- *The first configuration involves the OS197L cask in the supplemental trailer shielding with loss of water in the neutron shield. Dose rates at certain radial distances for this configuration are shown in Table W.5-4 (Case #4-3) (see data related to OS197L TC (with supplemental inner & outer trailer shielding)).*

- *The second configuration involves the OS197L cask in the supplemental trailer shielding without the outer top supplemental trailer shielding and without water in the neutron shield. Dose rates at certain radial distances for this configuration are shown in Table W.5-4 (Case #4-4) (see data related to OS197L TC (with supplemental inner trailer shielding only)).*

- *The third configuration involves the OS197L bare cask with loss of supplemental trailer shielding and loss water in the neutron shielding. This configuration can only occur during the handling of a bare cask with loaded fuel and is not likely to occur during transfer operations. Dose rates at certain radial distances for this configuration are shown in Table W.5-4 (Case #4-5) (see data related to OS197L TC (bare cask)). Detailed radial dose rate results for this configuration are obtained from the results shown in Table W.5-7 (for 32PT DSC) and Table W.5-10 (for 61BT DSC). The maximum dose rates (neutron, gamma, and total) from these tables are reported in Table W.5-4.*

- *The fourth configuration involves the loss of inner and outer neutron steel shells (liners) in addition to the loss of supplemental trailer shielding and without water in the neutron shielding as described above. This configuration has not been evaluated in an accident dose rate calculation with any other transfer cask design and is evaluated herein for conservatism. The bounding radial dose rate results for this configuration can be inferred from Table W.5-4, per the first note below the table. Dose rates associated with this configuration have not been used for the evaluation discussed in Chapters W.10 or W.11.*

The accident condition dose rate results are compared to those for the 32PT DSC/OS197 TC calculations documented in Appendix M, Chapter M.5, Table M.5-3. Note that these dose rates are also shown in Appendix M, Chapter M.11, Table M.11-2. These dose rates are also shown in Table W.5-4 as Case #4-1 for the 32PT DSC (see data related to UFSAR). The accident condition dose rate results are compared to those for the 61BT DSC/OS197 TC calculations documented in Appendix K, Chapter K.5 Table K.5-2. Note that these dose rates are also shown in Appendix K, Chapter K.11, Table K.11-4. These dose rates are also shown in Table W.5-4 as Case #4-2 for the 61BT DSC (see data related to UFSAR).

W.5.4.10 <u>OS197L TC Models During Fuel Loading and Transfer Operations</u>

MCNP models are developed for the various operational evolutions during fuel loading and transfer using the 32PT or 61BT DSC. For most of the operational sequences, water is always present in the DSC/TC annulus, however, the dose rates are calculated using models that,

All changes on this page are AMD 11

conservatively, do not credit the presence of water in the annulus. These operational sequences and their modeling are described below.

All changes on this page are AMD 11

W.5.4.10.1 <u>TC Loading and Placement in Decontamination Area</u>

This operation involves the loading of fuel in the DSC and the movement of the loaded DSC in the OS197L TC to the decontamination area that houses the 6" supplemental decontamination area cask shielding. The decontamination area cask shielding is a two-piece shielding structure where the upper portion (shield bell) is placed after the OS197L with the loaded DSC is placed into the lower portion of the shielding. The DSC cavity, DSC/TC annulus and the TC neutron shield are filled with water. The actual lifting and transfer operations are performed using remote crane operation using a laser/optical targeting system and cameras for confirmation of the cask location without the need for personnel in the vicinity of the cask. Should a failure of the crane occur during these operations, procedures will be in place to either repair the crane using proper ALARA practices and resume remote operations, or manually position the load in a safe, shielded location. Therefore, the dose received by operations personnel resulting from this high dose operation will be minimal as these operations are short duration and are performed remotely with no personnel in the vicinity. The applicable bounding dose rate distributions in the radial direction for estimating the dose rates for ALARA planning of repair and recovery operations during malfunctions are shown in Tables W.5-7 and W.5-8 for the 32PT DSC and Tables W.5-10 and W.5-11 for the 61BT DSC. One can also conservatively apply those dose rates to axial locations at radial distances beyond the perimeter (as R=TC Radius on Figure W.5-3). Bounding axial dose rates at R< TC radius are discussed in Section W.5.1.3.

W.5.4.10.2 <u>Cask Decontamination</u>

The DSC and the OS197L TC are placed inside the decontamination area shield. The top shield plug is assumed to be in place and the DSC/TC annulus and the neutron shielding are filled with water. This is identical to the decontamination operation documented in Appendix M, Chapter M.5 for the 32PT DSC/OS197 TC in the axial direction. This is also identical to the decontamination operation documented in Appendix K, Chapter K.5 for the 61BT DSC/OS197 TC in the axial direction. Therefore, the axial dose rate results from Appendix M, Figure M.5-26 or Appendix K, Figure K.5-13 can be conservatively applied. The top end dose rate for the 32PT DSC with the OS197 TC is slightly less than 7000 mrem/hr at the DSC axis and is less than 10,500 mrem/hr at the DSC periphery and DSC/TC annulus. The top end dose rate for the 61BT DSC with the OS197 TC is slightly less than 7000 mrem/hr at the DSC axis and DSC periphery and is less than 11,000 mrem/hr at the DSC/TC annulus.

For the OS197L TC, the qualification of fuel to be loaded within the 32PT or the 61BT DSC is limited to fuel that will result in significantly lower dose rates than calculated with the OS197 TC. Further, the results in Table W.5-8 and Table W.5-11 show that the maximum radial dose rates near the axial ends (approximately 500 cm from the cask bottom) are less than 3000 mrem/hr and are clearly bounded by the 10,000 mrem/hr dose rate results from the Appendix K and Appendix M calculations. Therefore, the axial dose rates calculated with the OS197 TC remain bounding for the OS197L TC.

Cask decontamination operations are described in SAR Section W.8.1.3.

The only additional step that is not evaluated is the operation involving the "inspection" of the upper and lower openings of the decontamination area cask shielding for blockage. The maximum radial surface dose rates for normal conditions at the axial locations of these openings are conservatively utilized to determine the dose rates for this operation. The radial dose rates as a function of axial height for the bare OS197L cask (with and without water in the neutron shield) are shown in Table W.5-8 for 32PT DSC and Table W.5-11 for 61BT DSC and are utilized to determine the dose rates at the upper and lower openings of the decontamination area shield.

The decontamination area cask shielding completely covers the DSC cavity. The upper and lower openings of the decontamination area cask shielding are at axial locations of approximately 10 inches from the top and bottom of the TC. A separate MCNP model is not developed to determine the actual dose rate distribution for the purpose of decontamination, in particular, the dose rate distribution near the openings of the decontamination area cask shielding. The dose rates at these locations initially increase with radial distance (due to contribution from the middle regions). Therefore, the 1m dose rate results shown in Table W.5-8 and W.5-11 are employed to determine the maximum dose rates near these openings. From Table W.5-11, the maximum dose rate is approximately 3000 mrem/hr. Note that this is based on a very conservative estimate of a bare cask configuration. With the decontamination area cask shielding in place, the maximum dose rate from Table W.5-3 (Case #3-4) is 61 mrem/hr. To calculate an average dose rate for the purpose of occupational exposure, a simplified calculation is performed. A scaling factor to determine the effect of the presence of the decontamination area cask shielding on the dose rates in the vicinity of the openings is estimated. Considering the fact that the maximum dose rate reduces by approximately a factor of 150 due to the presence of the decontamination area cask shielding, a scaling factor of 4.5 (volumetric scaling) is considered conservative. Therefore, it is estimated that the average dose rate is less than 700 mrem/hr.

W.5.4.10.3 Welding Operations

The 32PT DSC and the OS197L TC are still inside the decontamination area cask shielding area. The 32PT DSC top shield plug and inner top cover plate are assumed to be in place for inner top cover welding operation. The DSC cavity is assumed to be dry (for modeling simplicity) and the DSC/TC annulus and the neutron shielding are filled with water. Temporary shielding consisting of three inches of NS-3 and one inch of steel is assumed to cover the 32PT DSC inner top cover plate. In addition, the DSC outer top cover plate is not present. This is identical to the inner top cover welding operation documented in Appendix M, Chapter M.5 for the 32PT DSC/ OS197 TC in the axial direction. This is also identical to the decontamination operation documented in Appendix K, Chapter K.5 for the 61BT DSC/OS197 TC in the axial direction. Therefore, the axial dose rate results from Appendix M, Figure M.5-27 or Appendix K, Figure K.5-14 can be conservatively applied. The maximum top end dose rate at the DSC/TC annulus for both the 32PT DSC and the 61BT DSC is approximately 5000 mrem/hr. Since these dose rates are also calculated using design basis source terms, they can be utilized conservatively for the OS197L TC.

All changes on this page are AMD 11

W.5.4.10.4 *TC Placement in the Transfer Trailer*

Following welding and sealing of the DSC, the TC is lifted and placed horizontally on the transfer trailer containing the supplemental trailer shielding. The supplemental trailer shielding consists of a 5.5" side shielding attached to the trailer where the TC is lowered. Subsequently, the 2.5" "inner" top trailer shielding encloses the TC and prepares the system for transfer. An MCNP model, employing half-symmetry, representing this configuration is utilized to determine the dose rates as a function of distance. This model also considers the absence of shielding beneath the cask except for the 0.25" thick plate representing the trailer platform as shown in Figure W.5-5. The vents in the trailer shielding are not modeled explicitly; however, the details in the MCNP model are sufficient to obtain a conservative calculation of the radial dose rates. The MCNP model details including geometrical and material descriptions are shown in Figure W.5-5. Additional description is provided in Section W.5.4.7. The bounding radial dose rate results for this configuration are shown in Table W.5-12 and also plotted on Figure W.5-2 (described as curve 3 in Section W.5.1.2). The dose rates below the cask support skid are expected to be maximized, in particular at closer distances, because of the absence of shielding between the trailer deck and trailer shield. The results demonstrate that this effect diminishes with distance and at distances greater than 2m the dose rate peaking is minimized. The maximum radial neutron, gamma and total dose rate for this configuration below the cask support skid are bounded by 1466, 540, and 1543 mrem/hr, respectively. The bounding dose rates for this configuration when the neutron shield is filled with water are plotted on Figure W.5-2 (described as curve 5 in Section W.5.1.2). The maximum dose rate at 100 m prior to the installation of the outer top trailer shielding is about 0.1 mrem/hr thereby ensuring that off-site doses are not significantly affected during the placement of the outer top trailer shielding.

It is expected that the actual lifting and transfer operations are performed using remote crane operation using cameras for confirmation of the cask location without the need for personnel in the vicinity of the cask. Should a failure of the crane occur during these operations, procedures will be in place to either repair the crane using proper ALARA practices and resume remote operations, or manually position the load in a safe, shielded, location. Therefore, the dose received by operations personnel resulting from this high dose operation will be minimal as these operations are short duration and are performed remotely with no personnel in the vicinity. The applicable dose rate distributions for estimating the dose rates for ALARA planning of repair and recovery operations during malfunctions are shown in Table W.5-6 and Table W.5-8 in the radial direction for the 32PT DSC and in Table W.5-9 and Table W.5-11 in the radial direction for the 61BT DSC. Geometric locations for axial dose rate calculations are shown in Figure W.5-3. The dose rate distribution for accidental configurations during these operations bound those during decontamination. The results of these calculations are shown in Table W.5-14. As discussed previously, the maximum radial neutron, gamma and total dose rate for this configuration with empty neutron shield below the cask support skid are bounded by 1466, 540, and 1543 mrem/hr, respectively.

W.5.4.10.5 *Cask Transfer to ISFSI Operations*

These operations are performed outside the fuel building when the DSC is actually transferred to the HSM. For this purpose, the neutron shielding is filled with water and the cask lid is in place. The additional 3" of "outer" top shielding is also in place. The loss of neutron shielding

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accident or the loss of "outer" shielding accidents are bounded by the dose rates calculated for the "pre-transfer" operations described in Section W.5.4.10.4. The dose rates are calculated assuming that the OS197L cask is completely enclosed by supplemental trailer shielding. The results for this configuration with water in the neutron shield are shown in Table W.5-13. Dose rates at certain radial distances from the same shielding configuration but without water in the neutron shield are provided in Table W.5-4 (Case #4-3), see data relevant to OS197L TC (with supplemental inner & outer trailer shielding). Applicable bounding axial dose rates are discussed in Section W.5.1.3. Dose rates shown in Table W.5-6 and Table W.5-9 can be conservatively applied to determine the axial dose rates for this configuration with water in the neutron shield at radial distances beyond the edge of the transfer trailer. Dose rates shown in Table W.5-7 and Table W.5-10 can be conservatively applied to determine the axial dose rates for this configuration without water in the neutron shield at radial distances beyond the edge of the transfer trailer.

The dose rates at the axial ends of the OS197L TC within the radius of the TC are bounded by those for the transfer operation documented in Appendix M, Chapter M.5 for the 32PT DSC/OS197 TC. This is also identical to the transfer configuration documented in Appendix K, Chapter K.5 for the 61BT DSC/OS197 TC in the axial direction. Therefore, the axial dose rate results from Appendix M, Table M.5-5 or Appendix K, Table K.5-4 can be conservatively applied. The maximum top end dose rate within the radius of the TC for the 32PT DSC is 107 mrem/hr and that for the 61BT DSC is 132 mrem/hr. The maximum bottom end dose rate within the radius of the TC for the 32PT DSC is 1710 mrem/hr and that for the 61BT DSC is 2540 mrem/hr. Since these dose rates are also calculated using design basis source terms, they can be utilized conservatively for the OS197L TC.

All changes on this page are AMD 11

W.5.5 References

5.1 *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.*

5.2 *A General Monte Carlo N-Particle Transport Code, Version 5, Volume I: Overview and Theory, LA-UR-03-1987 and Volume II: User's Guide, LA-CP-03-0245, 2005.*

5.3 *CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P3, Cross Sections for Shipping Cask Analysis," DLC-23, Oak Ridge National Laboratory, RSIC Data Library Collection, June 1987.*

5.4 *"American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, LaGrange Park, Illinois, March 1977.*

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W.5.6 *List of Input Files*

The list of input files begins on the next page.

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Table W.5-1
Bounding Radiological Source for Fuel Assemblies in the 32PT DSC within the OS197L TC

PGR Spectrum			*Normalized Spectrum and Total Strengths per Assembly for PGR in Each Axial Region. Total Neutron Source Strength per Assembly.*							
			12 Zone 1 Fuel Compartments				*20 Zone 2 Fuel Compartments*			
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle	Bottom Nozzle	In-core	Plenum	Top Nozzle
0.00	to	0.05	2.072e-2	2.880e-1	3.299e-2	2.073e-2	2.172e-2	3.254e-1	2.435e-2	2.178e-2
0.05	to	0.10	2.601e-3	5.674e-2	2.613e-3	2.601e-3	2.608e-3	6.637e-2	2.610e-3	2.608e-3
0.10	to	0.20	6.275e-4	4.338e-2	1.615e-3	6.277e-4	6.295e-4	4.203e-2	8.209e-4	6.296e-4
0.20	to	0.30	3.118e-5	1.242e-2	9.036e-5	3.120e-5	3.140e-5	1.259e-2	4.302e-5	3.142e-5
0.30	to	0.40	4.086e-5	8.460e-3	2.765e-4	4.088e-5	4.092e-5	8.216e-3	8.637e-5	4.092e-5
0.40	to	0.60	2.584e-6	8.636e-2	5.267e-3	2.584e-6	2.584e-6	8.855e-3	1.018e-3	2.584e-6
0.60	to	0.80	7.701e-5	4.027e-1	2.882e-3	8.684e-5	4.293e-4	5.043e-1	1.414e-3	4.845e-4
0.80	to	1.00	2.502e-3	3.746e-2	4.090e-4	2.239e-3	4.472e-4	5.700e-3	8.895e-4	5.004e-4
1.00	to	1.33	7.590e-1	5.019e-2	7.438e-1	7.592e-1	7.596e-1	2.190e-2	7.555e-1	7.594e-1
1.33	to	1.66	2.144e-1	1.367e-2	2.100e-1	2.144e-1	2.145e-1	4.628e-3	2.133e-1	2.145e-1
1.66	to	2.00	1.997e-12	1.228e-4	6.120e-12	1.988e-12	1.392E-15	2.825e-5	2.258E-11	1.572E-15
2.00	to	2.50	5.087e-6	4.640e-4	4.985e-6	5.088e-6	5.091e-6	1.519e-6	5.063e-6	5.090e-6
2.50	to	3.00	7.887e-9	9.451e-6	7.729e-9	7.890e-9	7.894e-9	6.128e-8	7.850e-9	7.892e-9
3.00	to	4.00	9.082e-24	1.192e-6	1.848e-23	1.020e-23	5.792e-24	1.381e-8	1.217e-23	6.279e-24
4.00	to	5.00	0.0	2.393e-9	0.0	0.0	0.0	4.529e-9	0.0	0.0
5.00	to	6.50	0.0	9.604e-10	0.0	0.0	0.0	1.817e-9	0.0	0.0
6.50	to	8.00	0.0	1.884e-10	0.0	0.0	0.0	3.565e-10	0.0	0.0
8.00	to	10.0	0.0	4.000e-11	0.0	0.0	0.0	7.569e-11	0.0	0.0
Total Gamma, g/(sec*FA)			1.2862e+13	3.3097e+15	5.0346e+12	7.9104e+12	2.9087e+12	1.7183e+15	1.1213e+12	1.7895e+12
Total Neutrons, n/(sec*FA)			6.86e+8				3.37e+8			

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Table W.5-2
Bounding Radiological Source for Fuel Assemblies in the 61BT DSC within the OS197L TC

Spectrum and Total Strengths per Assembly for PGR in Each Axial Region. Total Neutron Source Strength per Assembly.											
48 Peripheral Fuel Compartments							*[(1)] 13 Inner Fuel Compartments*				
PGR Spectrum, Total Source Strength per Assembly											
E_{mim} MeV	*to*	*E_{max} MeV*	*Bottom Nozzle*	*In-core*	*Plenum*	*Top Nozzle*	*PGR Spectrum, E_{ave} MeV*	*Bottom Nozzle*	*In-core*	*Plenum*	*Top Nozzle*
0.00	to	0.05	2.8811e+9	2.1727e+14	8.1557e+8	2.0220e+9	0.01	1.113e+11	4.864e+14	3.892e+10	8.060e+10
0.05	to	0.10	2.6683e+8	4.6909e+13	7.9633e+7	1.9827e+8	0.025	3.023e+10	1.220e+14	6.659e+10	2.145e+10
0.10	to	0.20	6.4317e+7	2.6703e+13	2.1269e+7	4.8066e+7	0.0375	1.342e+10	1.270e+14	1.825e+10	9.680e+9
0.20	to	0.30	3.1990e+6	8.3024e+12	1.0856e+6	2.4071e+6	0.0575	1.190e+10	9.809e+13	3.894e+9	8.759e+9
0.30	to	0.40	4.2356e+6	5.6236e+12	1.7628e+6	3.1695e+6	0.085	4.683e+ 9	6.835e+13	1.556e+9	3.447e+9
0.40	to	0.60	1.7739e+6	3.9877e+12	1.1541e+7	1.6655e+6	0.125	1.866e+9	6.506e+13	9.486e+8	1.369e+9
0.60	to	0.80	1.0805e+7	3.2911e+14	5.9619e+6	5.0905e+7	0.225	1.614e+9	5.372e+13	5.482e+9	1.118e+9
0.80	to	1.00	1.3025e+7	1.6177e+12	1.0281e+6	5.0851e+7	0.375	6.267e+9	3.555e+13	3.160e+10	4.193e+9
1.00	to	1.33	7.7394e+10	2.6000e+12	2.3057e+10	5.7562e+10	0.575	7.853e+9	8.468e+14	4.055e+10	5.240e+9
1.33	to	1.66	2.1856e+10	3.1550e+11	6.5113e+9	1.6256e+10	0.85	2.222e+10	1.923e+14	6.666e+9	6.970e+9
1.66	to	2.00	2.2526e+0	1.4279e+10	1.8180e+1	2.3428e+0	1.25	4.000e+12	6.670e+13	1.152e+12	2.947e+12
2.00	to	2.50	5.1867e+5	7.4338e+8	1.5452e+5	3.8577e+5	1.75	2.511e+2	1.395e+12	8.200e+1	1.896e+2
2.50	to	3.00	8.0426e+2	6.0989e+7	2.3960e+2	5.9817e+2	2.25	2.120e+7	6.823e+11	6.103e+6	1.562e+7
3.00	to	4.00	2.2255e-12	9.3055e+6	9.4410e-16	1.4295e-11	2.75	6.560e+4	2.634e+10	1.889e+4	4.833e+4
4.00	to	5.00	0.0	3.1409e+6	0.0	0.0	3.5	9.733e-15	3.386e+9	1.944e-18	2.559e-14
5.00	to	6.50	0.0	1.2604e+6	0.0	0.0	5.0	0.0	4.127e+6	0.0	0.0
6.50	to	8.00	0.0	2.4723e+5	0.0	0.0	7.0	0.0	4.759e+5	0.0	0.0
8.00	to	10.0	0.0	5.2487e+4	0.0	0.0	9.5	0.0	5.468e+4	0.0	0.0
Total Gamma, g/(sec*FA)			**1.0250e+11**	**6.4245e+14**	**3.0506e+10**	**7.6196e+10**	**Total PGR**	**4.211e+12**	**2.160e+15**	**1.366e+12**	**3.090e+12**
Total Neutrons, n/(sec*FA)			9.86e+7				**Total Neutrons**	1.427e+8			

[(1)] *This is a 61BT design basis radiological source from Table K.5-7 from the UFSAR. It is replicated here for convenience.*

Table W.5-3
Summary of OS197L TC Normal Condition Bounding Dose Rates

Case #	Transfer Cask Configuration	Dose Rate[5] Component	Dose Rates (mrem/hr) at Different Distances from Side Surface Normal Condition–Water in Neutron Shield: On Side Surface	4.57 meters (15')	100 meters	609.9 meters (2000')
3-1	UFSAR (Table M.5-5 and Section M.11.2.5.3)	Neutron	261	Not calc.	Not calc.	Not calc.
		Gamma	784	Not calc.	Not calc.	Not calc.
		Total	950	Not calc.	Not calc.	0.01
3-2	OS197 TC[1] Results are directly shown in this table.	Neutron	102	7.20	0.006	7.09e-6
		Gamma	248	20.3	0.03	5.29e-5
		Total	346	25.9	0.03	5.67e-5
3-3	OS197L TC bare cask (Maximum from Table W.5-6 and Table W.5-9)	Neutron	323	22..4	0.022	1.51e-5
		Gamma	9,521	824	1.41	1.45e-3
		Total	9,835	845	1.42	1.46e-3
3-4	OS197L TC with decontamination area cask or supplemental trailer shielding[2,4] Table W.5-13	Neutron	27.9	4.00	0.02	1.32e-5
		Gamma	39.5	25.0	0.08	1.83e-4
		Total	60.6	29.0	0.10	1.96e-4
3-5	OS197L TC without the outer top supplemental trailer shielding[3,4] Table W.5-12	Neutron	58.6	4.30	0.02	8.20e-5
		Gamma	336	36.7	0.17	6.32e-4
		Total	394	40.8	0.20	7.14e-4

(1) Dose rates are due to 32PT DSC design basis radiological sources. These are calculated to compare against those shown for the OS197 TC in the UFSAR, shown in this table as Case #3-1.

(2) The dose rates are also applicable to the cask on trailer at vertical elevations above the trailer support skid. These are dose rates a person could potentially be exposed when doing manual operations at altitudes above the trunnions level; for example, traversing the crane bridge above the OS197L. Use data in Table W.5-14 for dose rates below the trailer support skid.

(3) The dose rates are applicable for radial locations over the 2.5" thick inner top supplemental trailer shielding prior to the installation of the outer top trailer shielding. These dose rates do not reflect those at radial distances from the side of the trailer. Use data in Table W.5-14 for dose rates below the trailer support skid.

(4) Table W.5-14 dose rates below the trailer support skid do not account for shielding from the trailer gear boxes, wheels assembly that may provide substantial shielding at certain locations near the trailer platform. Therefore, they represent a conservative estimate.

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

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Table W.5-4
Summary of OS197L TC Accident Condition Bounding Dose Rates

Case #	*Transfer Cask Configuration*	*Dose Rate[5] Component*	*Dose Rates (mrem/hr) at Different Distances from Side Surface Accident Condition–No Water in Neutron Shield*			
			On Side Surface	*4.57 meters (15')*	*100 meters*	*609.9 meters (2000')*
4-1	*UFSAR (Table M.5-3, Table M.11-2)*	*Neutron*	*3,780*	*Not calc.*	*Not calc.*	*Not calc.*
		Gamma	*1,070*	*Not calc.*	*Not calc.*	*Not calc.*
		Total	*4,640*	*Not calc.*	*Not calc.*	*< 0.02*
4-2	*UFSAR (Table K.5-2, Table K.11-4)*	*Neutron*	*3,700*	*Not calc.*	*Not calc.*	*Not calc.*
		Gamma	*4,820*	*Not calc.*	*Not calc.*	*Not calc*
		Total	*8,520*	*Not calc*	*Not calc*	*< 0.02*
4-3	*OS197L TC[1, 3] (with supplemental inner & outer trailer shielding) Results are directly shown in this table.*	*Neutron*	*727*	*79.0*	*0.32*	*2.31e-4*
		Gamma	*134*	*40.0*	*0.14*	*3.04e-4*
		Total	*791*	*104*	*0.46*	*5.34e-4*
4-4	*OS197L TC[1, 4] (with supplemental inner trailer shielding only) Results are directly shown in this table.*	*Neutron*	*1466*	*87.0*	*0.48*	*7.36e-4*
		Gamma	*540*	*59.0*	*0.29*	*1.07e-3*
		Total	*1543*	*129*	*0.77*	*1.81e-3*
4-5	*OS197L TC[1] (bare cask) (Maximum from Table W.5-7 and Table W.5-10)*	*Neutron*	*4,194*	*175*	*0.20*	*4.97e-5*
		Gamma	*15,305*	*1,332*	*2.30*	*2.43e-3*
		Total	*18,210*	*1,438*	*2.48*	*2.46e-3*
4-6	*OS197 TC[2] Results are directly shown in this table.*	*Neutron*	*1,282*	*66.0*	*0.07*	*1.87e-5*

*[1] 0.19" thick neutron shield shell(s) are credited in the calculations. To obtain a rough and conservative estimates for dose rates without the neutron shield shells one can scale the dose rates by the factor of exp(ln(2)*0.19/0.85)=1.17, where 0.85" is a half layer thickness of steel for Co-60 radiation.*

[2] Dose rates are due to 32PT DSC design basis radiological sources. Those sources would result in nearly 87 rem/hr maximum dose rate on the side of the bare OS197L cask without water in the neutron shield. These are calculated to compare against those shown for the OS197 TC, shown in this table as Case #4-1.

[3] The dose rates are also applicable to the cask on the trailer at vertical elevations above the trailer support skid.

[4] The dose rates are applicable for radial locations around the 2.5" thick inner top supplemental trailer shielding prior to the installation of the outer top supplemental trailer shielding. These dose rates do not reflect those at radial distances from the side of the trailer.

[5] Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the maximum total dose rate (as reported) is not always the sum of the maximum gamma plus maximum neutron dose rate.

Table W.5-5

Dose Rate Results for Two Trunnion Designs (mrem/hr)

Trunnion Type	***Neutron Dose Rate***	***Gamma Dose Rate***	***Total Dose Rate***
Original upper	*0.20*	*621*	***621***
Solid steel upper	*51.1*	*.14*	***51.2***
Original lower	*1.00*	*1702*	***1703***
Solid steel lower	*79.5*	*1.30*	***80.8***

Note that the dose rates presented are due to 32PT DSC design basis sources and are conservative for this purpose.

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Table W.5-6
Bounding Radial Dose Rates for the Bare OS197L TC with 32PT DSC
(Normal Condition, Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	323	0.010	9,521	0.004	9,835	0.004
1	136	0.003	4,022	0.002	4,158	0.002
2	71.8	0.003	2,306	0.002	2,378	0.002
3	42.7	0.003	1,463	0.002	1,506	0.003
4.57 (15')	22.4	0.005	824	0.003	845	0.007
10	5.63	0.005	217	0.003	223	0.010
50.8 (2000")	0.16	0.010	7.52	0.010	7.67	0.010
100	0.02	0.020	0.98	0.010	1.01	0.010
200	2.30E-03	0.070	0.11	0.020	0.11	0.020
300	5.02E-04	0.130	0.02	0.030	0.02	0.030
609.6 (2000')	1.51E-05	0.290	6.37E-04	0.060	6.45E-04	0.060

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

Table W.5-7
Bounding Radial Dose Rates for the Bare OS197L TC with 32PT DSC
(Accident Condition, No Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	2,904	0.003	15,305	0.010	18,210	0.008
1	968	0.002	6,522	0.004	7,491	0.003
2	470	0.002	3,755	0.004	4,225	0.004
3	271	0.002	2,378	0.004	2,649	0.010
4.57 (15')	141	0.003	1,333	0.010	1,430	0.011
10	34.6	0.003	351	0.010	386	0.010
50.8 (2000")	0.99	0.010	12.2	0.030	13.2	0.030
100	0.14	0.010	1.56	0.020	1.70	0.020
200	1.22E-02	0.020	0.19	0.080	0.21	0.076
300	2.02E-03	0.030	0.04	0.040	0.04	0.040
609.6 (2000')	3.85E-05	0.120	9.53E-04	0.100	9.88E-04	0.100

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

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Table W.5-8
Bounding Radial Dose Rates for the OS197L TC with 32PT DSC as a Function of Axial Height

Axial Distance from TC Bottom, (cm)	Bare Cask Dose Rates (mrem/hour) Normal Condition with Water in Neutron Shield		Bare Cask Dose Rates (mrem/hour) Accident Condition without Water in Neutron Shield	
	Surface	1 M from Surface	Surface	1 M from Surface
2.5	73	568	91	1,039
16.5	60	747	108	1,336
39.5	252	1,134	539	1,982
62.5	3,142	1,649	4,904	2,874
85.5	5,148	2,196	8,393	3,797
108.5	6,824	2,740	11,563	4,722
131.5	8,745	3,202	15,251	5,558
154.5	9,505	3,554	17,077	6,207
177.5	9,809	3,825	17,787	6,704
200.5	9,809	3,978	18,046	7,030
223.5	9,742	4,095	18,123	7,264
246.5	9,835	4,140	18,128	7,375
269.5	9,811	4,158	18,210	7,416
292.5	9,782	4,129	18,180	7,371
315.5	9,781	4,069	18,184	7,223
338.5	9,828	3,961	17,976	7,002
361.5	9,723	3,794	17,608	6,611
384.5	9,130	3,525	16,100	6,082
407.5	7,832	3,201	13,408	5,463
430.5	5,607	2,809	9,559	4,750
453.5	5,900	2,399	8,991	4,028
476.5	5,610	1,984	8,380	3,284
499.5	3,807	1,574	5,761	2,584
522.5	911	1,208	1,498	1,994
545.5	50	860	140	1,448
568.5	38	560	63	959
Maximum	9,835	4,158	18,210	7,416
Average	6,183	2,695	10,926	4,696

Note: Bottom of the TC is at 0 cm. Top of the TC is at 500 cm.

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Table W.5-9
Bounding Radial Dose Rates for the Bare OS197L TC with 61BT DSC (Normal Condition, Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	311	0.010	8,817	0.010	9,129	0.010
1	116	<0.01	3,566	<0.01	3,682	0.010
2	56.5	<0.01	2,030	<0.01	2,083	0.010
3	32.6	<0.01	1,326	<0.01	1,357	0.010
4.57 (15')	14.3	0.006	621	0.010	641	0.013
10	4.16	0.010	222	0.010	226	0.010
50.8 (2000")	0.12	0.020	8.00	0.020	8.12	0.020
100	0.02	0.040	1.41	0.020	1.42	0.020
200	2.05E-03	0.120	0.18	0.050	0.18	0.050
300	3.98E-04	0.090	0.04	0.080	0.04	0.078
609.6 (2000')	1.50E-05	0.280	1.45E-03	0.140	1.46E-03	0.140

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

Table W.5-10
Bounding Radial Dose Rates for the Bare OS197L TC with 61BT DSC (Accident Condition, No Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	4,194	0.004	14,816	0.010	18,141	0.008
1	1279	0.003	6,080	0.003	7,359	0.003
2	596	0.003	3,444	0.003	4,038	0.003
3	339	0.003	2,228	0.004	2,559	0.010
4.57 (15')	175	0.005	1,304	0.010	1,438	0.010
10	42.7	0.005	366	0.010	408	0.010
50.8 (2000")	1.26	0.010	13.4	0.020	14.7	0.020
100	0.20	0.010	2.30	0.030	2.48	0.030
200	1.68E-02	0.030	0.28	0.050	0.29	0.050
300	2.89E-03	0.040	0.06	0.080	0.07	0.080
609.6 (2000')	4.97E-05	0.140	2.43E-03	0.190	2.46E-03	0.190

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

Table W.5-11
Bounding Radial Dose Rates for the OS197L TC with 61BT DSC as a Function of Axial Height

Axial Distance from TC Bottom, (cm)	Bare Cask Dose Rates (mrem/hour) Normal Condition with Water in Neutron Shield		Bare Cask Dose Rates (mrem/hour) Accident Condition without Water in Neutron Shield	
	Surface	1 M from Surface	Surface	1 M from Surface
2.5	29	379	72	835
16.5	26	489	84	1,053
39.5	68	753	260	1,561
62.5	736	1,120	1,527	2,273
85.5	2,330	1,589	4,478	3,160
108.5	5,403	2,129	9,988	4,169
131.5	7,589	2,625	14,231	5,146
154.5	8,458	3,039	16,226	5,925
177.5	8,843	3,334	17,329	6,490
200.5	9,044	3,542	17,866	6,919
223.5	9,129	3,657	18,141	7,136
246.5	9,103	3,682	18,021	7,207
269.5	8,994	3,666	17,899	7,138
292.5	8,747	3,600	17,086	6,994
315.5	8,346	3,488	16,219	6,729
338.5	8,077	3,337	15,453	6,361
361.5	7,676	3,168	14,318	5,951
384.5	6,701	2,976	12,220	5,482
407.5	5,345	2,885	9,535	5,124
430.5	3,360	2,931	5,848	4,978
453.5	3,134	3,141	6,503	4,994
476.5	3,393	3,404	4,917	5,179
499.5	3,405	3,414	4,862	5,119
522.5	3,378	3,386	4,741	4,979
545.5	2,687	2,696	3,778	3,989
568.5	1,612	1,617	2,234	2,392
Maximum	9,129	3,682	18,141	7,207
Average	5,216	2,694	9,763	4,896

Note: Bottom of the TC is at 0 cm. Top of the TC is at 500 cm.

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Table W.5-12
OS197L TC Radial Dose Rates over 2.5" Inner Top Trailer Area Shielding (Normal Condition, Water in Neutron Shield)

Distance from Inner Top Trailer Area Shielding Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	58.6	0.010	336	0.010	394	0.010
1	22.2	0.010	156	0.004	178	0.004
2	12.0	0.010	94.4	0.004	106	0.004
3	7.60	0.010	62.9	0.005	70.2	0.005
4.57 (15')	4.30	0.010	36.7	0.010	40.8	0.010
10	1.30	0.010	11.0	0.010	12.2	0.010

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

Table W.5-13
OS197L TC Radial Dose Rates with 5.5" Trailer Area Shielding (Normal Condition, Water in Neutron Shield)

Distance from Outer Top Trailer Area Shielding Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	27.9	0.010	39.5	0.004	60.6	0.010
1	17.4	0.010	81.9	0.010	97.3	0.009
2	10.6	0.010	52.6	0.010	63.0	0.009
3	6.90	0.010	39.3	0.010	46.2	0.010
4.57 (15')	4.00	0.020	25.0	0.010	29.0	0.010
10	1.10	0.030	7.10	0.020	8.20	0.020

Note: Dose rates presented are bounding for OS197 L TC radial dose rates above cask support skid. There is no shielding underneath in the computation model used for determination of the dose rates presented in the tables except for only 0.25" thick steel plate on top of the trailer platform. Geometry of the model is depicted on sketches of Figure W.5-5. Contribution of scattered radiation is pronounced at short (less than 2 meters) radial distances. The presented maximum values of the dose rate are determined at vertical elevations above the cask support skid and they account for radiation scattered from concrete at grade level. Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the maximum total dose rate (as reported) is not always the sum of the maximum gamma plus maximum neutron dose rate.

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Table **W.5-14**
OS197L TC Radial Dose Rates below Cask Support Skid (Prior to Installation of Outer Top Trailer Area Shielding, Water in Neutron Shielding)

Distance in horizontal direction from side of 5.5" thick Side Shielding Plates, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	56.8	0.010	1878	0.006	1934.3	0.01
1	20.1	0.020	475	0.010	494.0	0.01
2	8.50	0.010	79.3	0.010	87.7	0.01
3	5.50	0.010	31.2	0.010	36.3	0.01
4.57 (15')	3.20	0.020	14.2	0.010	17.3	0.01
10	1.00	0.030	4.50	0.020	5.50	0.02

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

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Security-Related Information
Figure Withheld Under 10 CFR 2.390.

Figure W.5-1
OS197L TC MCNP Shielding Configuration Description

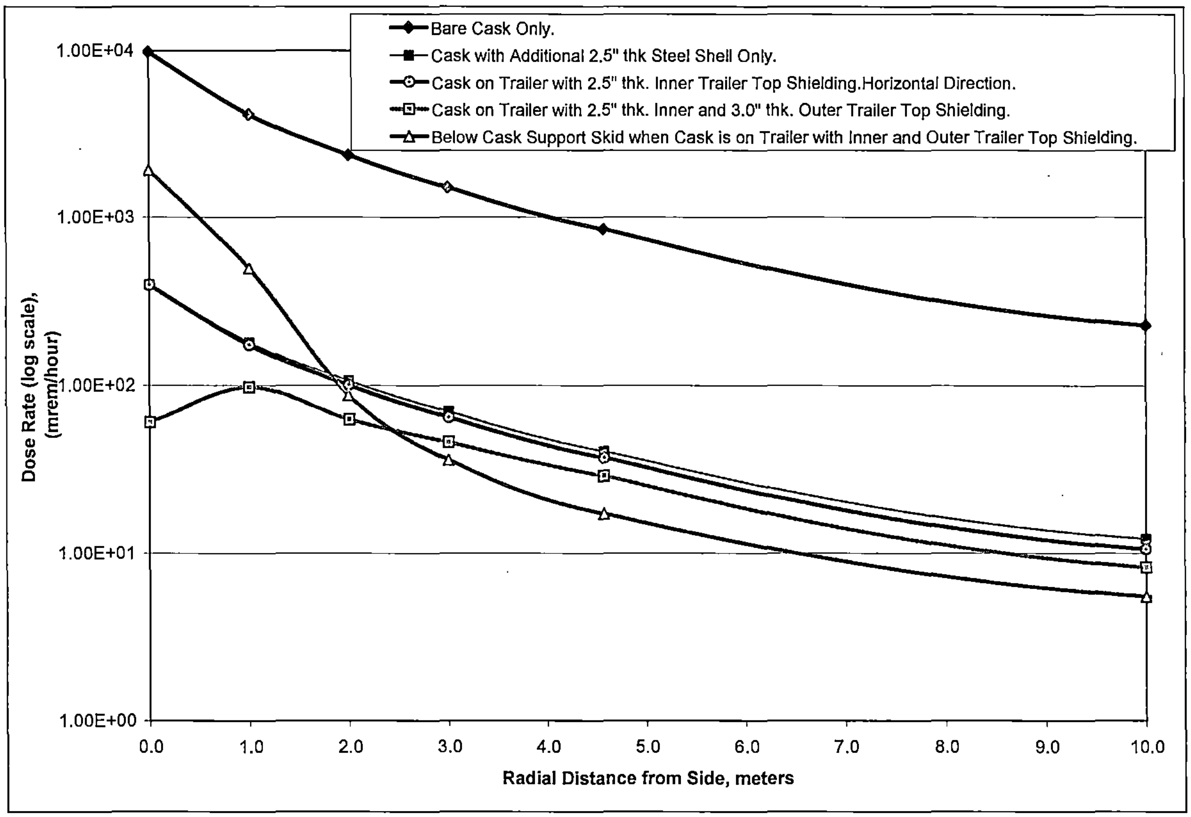

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Figure W.5-2
Bounding Radial Dose Rate Results for the Various OS197L TC Shielding Configurations

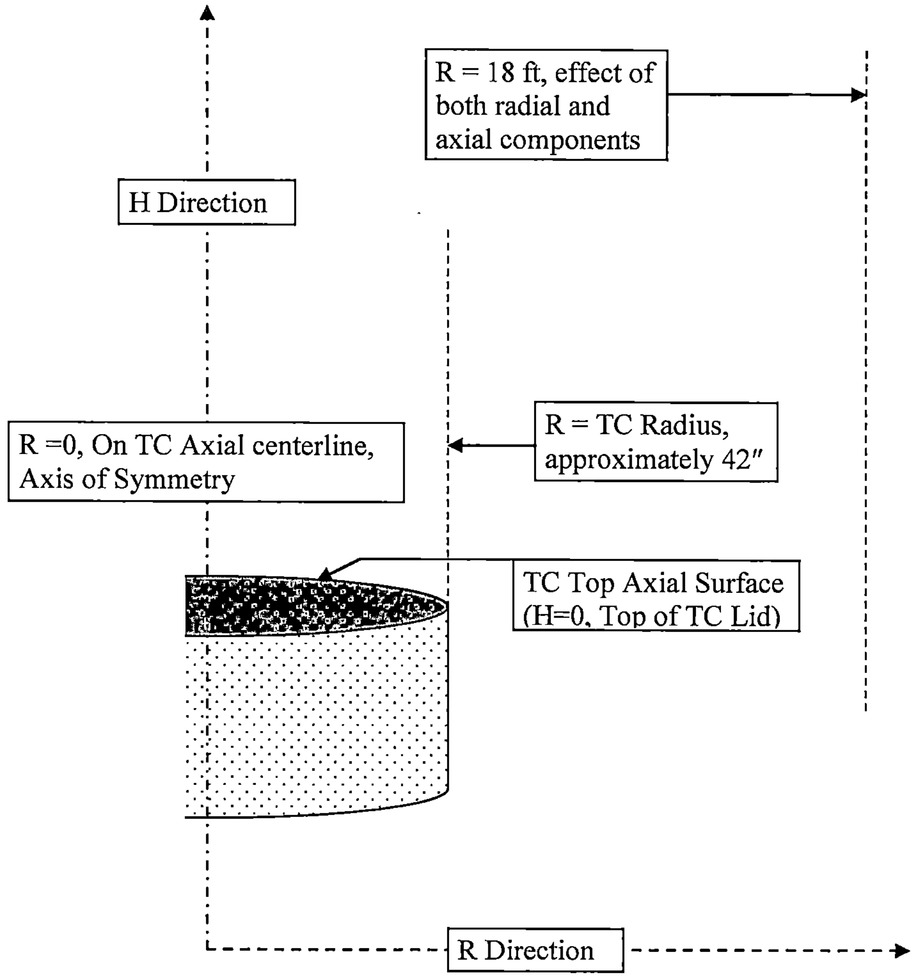

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***Figure* W.5-3**
Geometrical Layout for OS197L Axial Dose Rate Calculations

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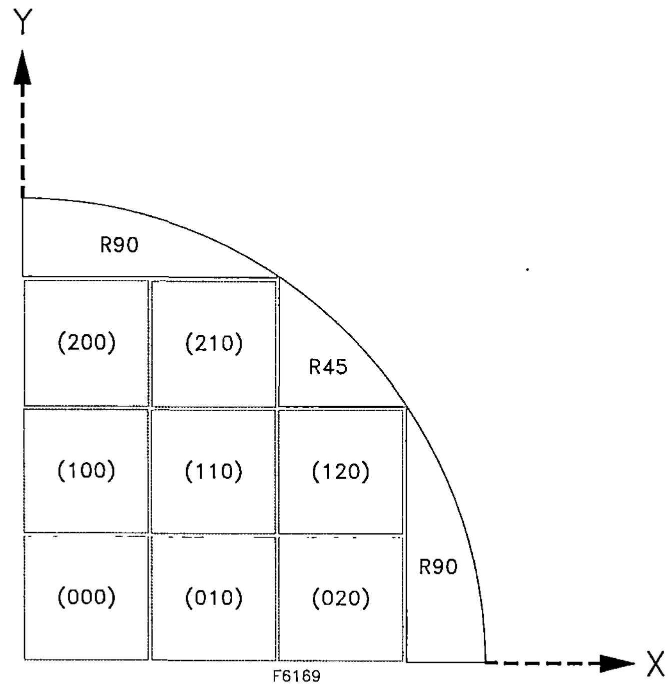

Figure **W.5-4**
MCNP Geometry of the 32PT DSC Basket Structure and Source Region

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Security-Related Information Figure Withheld Under 10 CFR 2.390.

***Figure* W.5-5**
Description of the Trailer Area Shielding Calculational Model

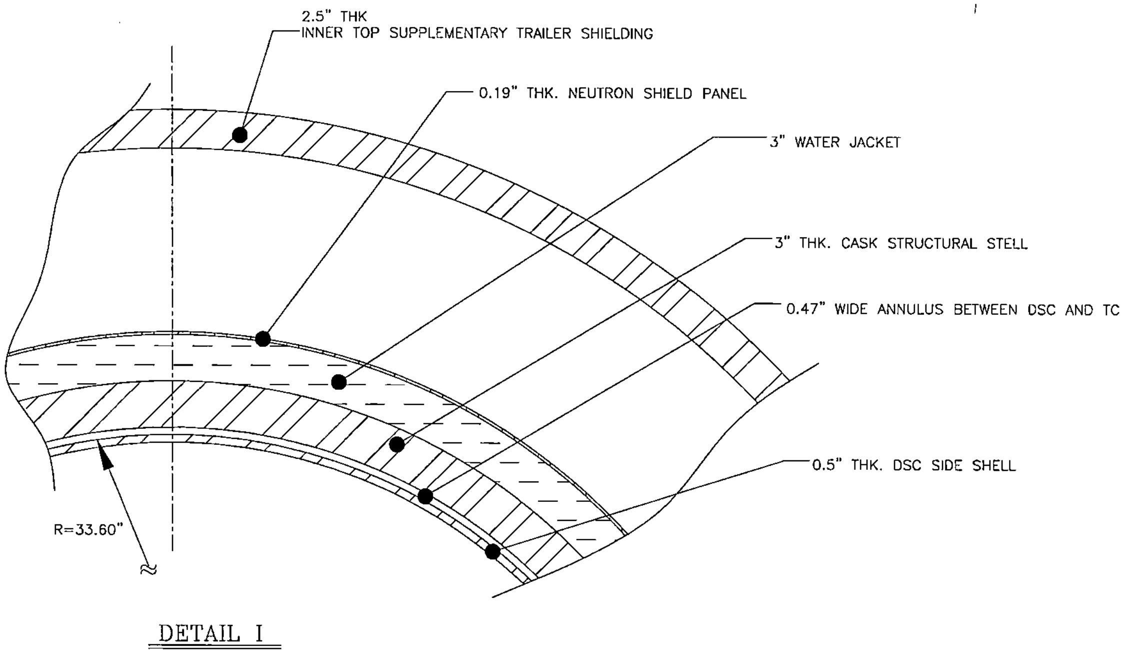

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Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)

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Security-Related Information
Figure Withheld Under 10 CFR 2.390.

Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)

All changes on this page are AMD 11

Security-Related Information Figure Withheld Under 10 CFR 2.390.

Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)

All changes on this page are AMD 11

Security-Related Information
Figure Withheld Under 10 CFR 2.390.

Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)

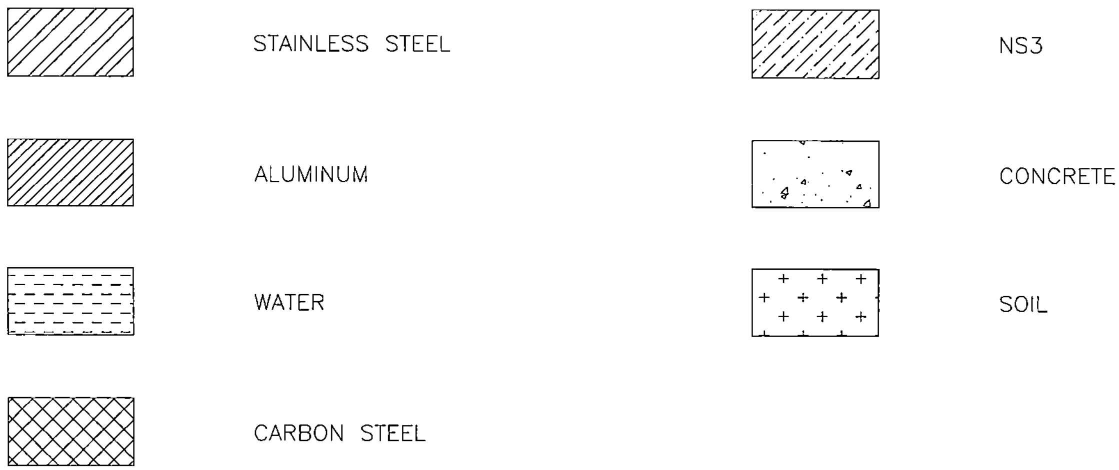

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Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(concluded)

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W.8 Operating *Systems*

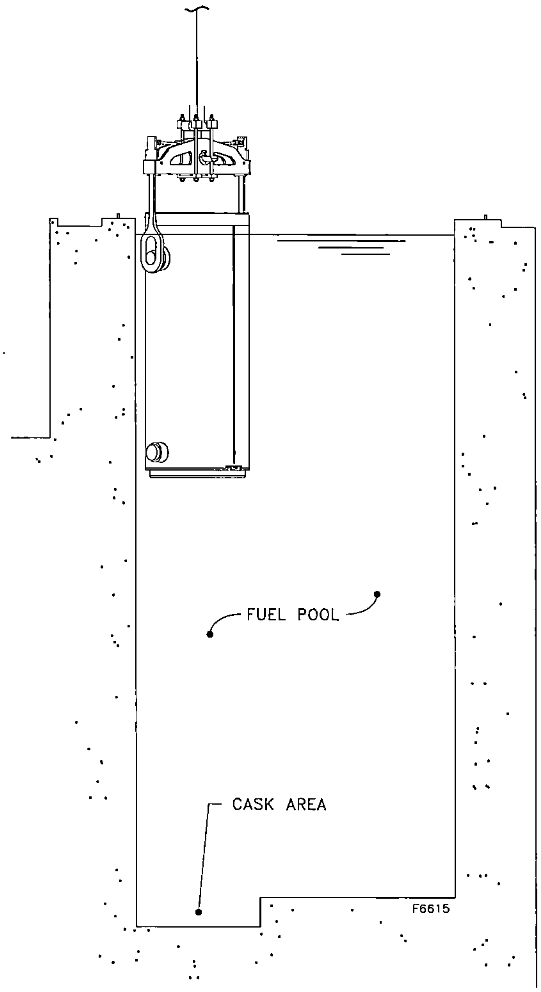
The following is a description of the operational sequences for use of the OS197L TC. In general, the steps are similar to those for the OS197 TC, described in detail in *Appendix K.8 or M.8* of the UFSAR. This chapter *consolidates these procedures and includes* the differences in operational steps when using OS197L TC relative to the OS197 TC. Figures are provided to illustrate *the differences in operational* steps.

Notes: A general licensee shall meet the requirements of applicable Technical Specifications *(such as 4.4.1 – 4.4.4)* prior to the use of OS197L TC for onsite transfer of an authorized payload.

The generic term "DSC" used throughout this chapter may be the 61BT or 32PT DSC. The term "cask" or "TC" is used for the OS197L TC.

Discussion of Similarities and Differences Between Use of the OS197 TC and OS197L TC Systems:

Placement of the DSC into the OS197L TC and preparations for placement of the TC into the fuel pool are the same as for the OS197 TC. The DSC/TC annulus is filled with clean water and sealed with the annulus seal. The TC neutron shield is also filled with clean water. *T*he DSC may be filled with fuel pool water *either* prior to lowering *OS197L TC* into the pool, or the OS197L TC lowered to within a few feet of submergence and the DSC filled at that time. The OS197L TC with DSC is then lowered to the fuel pool bottom and landed, and the yoke removed. Sequence 1 below shows the cask as it enters the pool.



(1) OS197L IS LOWERED INTO FUEL POOL

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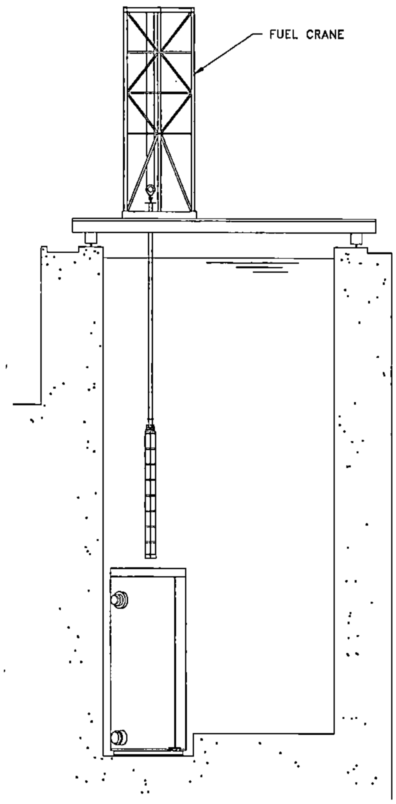
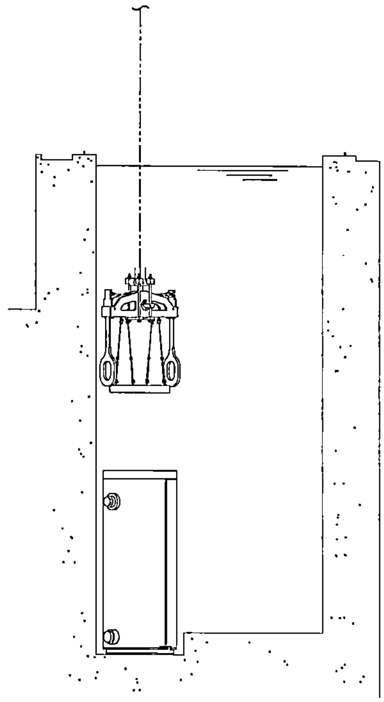
Selected *f*uel *a*ssemblies (FAs) are then placed into the DSC. Following fuel verification, the top shield plug is lowered into place and set. *This is shown as Sequences 2 and 3.* The yoke is then lowered and connected to the OS197L TC.



(2) FUEL ASSEMBLIES ARE LOADED INTO OS197L



(3) TOP SHIELD PLUG IS LOWERED INTO DSC

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The cask is then lifted until the cask top just breaks the surface of the fuel pool.

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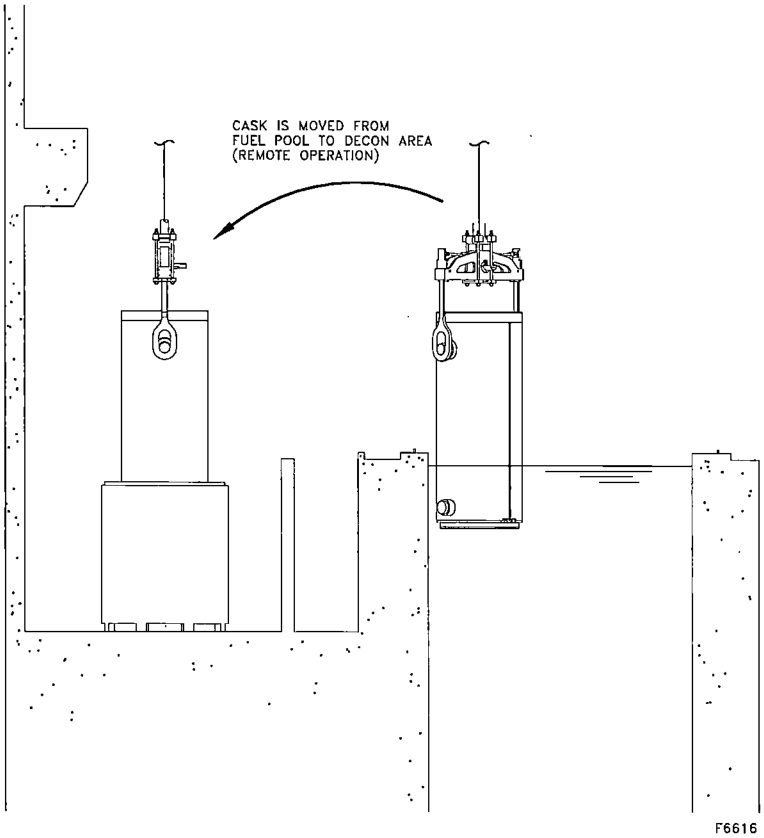
CAUTION: Prior to performing the next step of lifting OS197L TC from the pool, the licensee shall meet the specific radiation protection program requirements of applicable Technical Specification*s* associated with the use of OS197L TC and remote monitoring devices. The licensee shall develop appropriate measures to keep the dose rates ALARA during recovery from a potential malfunction of these devices, *such as cameras for monitoring, targeting devices, remote controls, etc.*

CAUTION: The dose rates during the movement of a bare loaded TC from the spent fuel pool to the decontamination area supplemental shielding may be as high as 9.8 rem/hr.

*T*he OS197L TC *is next* lifted from the fuel pool to the decontamination area. The *TC* itself has significantly reduced shielding. However, the OS197L TC *system utilizes supplementary* shielding and *additional operational* measures to achieve *acceptable* shielding capacity. *T*he OS197L TC system consists of the bare cask and the upper and lower cask shielding utilized in the decontamination area and the additional shielding provided on the cask support skid. The bare cask is in this reduced shielding configuration ONLY during the movement of the cask from the fuel pool to the decontamination area and from the decontamination area to the transfer trailer. Both of these operations are of short time duration (i.e. minutes).

During bare cask movement from the fuel pool to the decontamination area and from the decontamination area to the trailer, remote crane operation and an optical targeting system with remote camera monitoring *is* used to minimize personnel exposure to the reduced shielding configuration. This remote operation is shown in Sequence *4*.



4 FROM FUEL POOL TO DECONTAMINATION AREA, OS197L IS PLACED IN SHIELDING SLEEVE (PART OF OS197L CASK SYSTEM)

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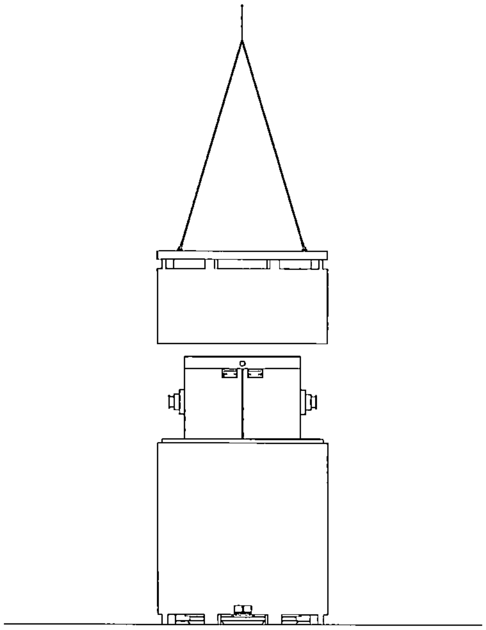
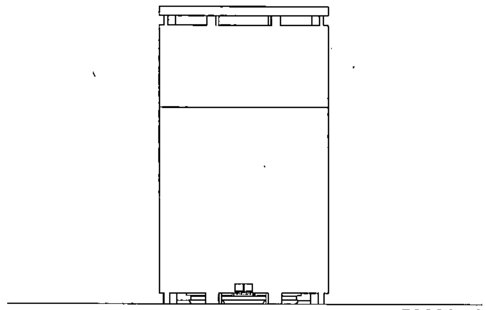
In the decontamination area, the bare cask is placed in a shielding sleeve (lower cask shield) which provides shielding below the trunnions. An upper cask shield (shielding bell) is then placed on top of the shielding sleeve to shield the upper section of the cask. The shielding sleeve and shield bell are nominally 6" thick carbon steel. Placement of the cask in the shielding sleeve and placement of the shielding bell on the cask is performed using remote crane operation and an optical targeting system with remote camera monitoring. The OS197L TC system configuration of the cask, shielding sleeve and bell is shown as Sequences *4* and *5*.

All changes on this page are AMD 11



5 TOP SHIELDING BELL COMPONENT OF OS197L IS PLACED (REMOTE OPERATION)

While in the shielding sleeve and bell, the canister is vacuum dried, helium backfilled, and all top covers welded in place. During these operations the DSC/TC annulus nearly remains full (approximately 12" drained from the top of the DSC) similar to OS197 TC operations. The OS197L TC neutron shield will remain filled and vented, similar to OS197 TC operations, during these steps. During these operations, the cask and the shielding sleeve and bell provide occupational radiation shielding for personnel necessary to perform the canister closure operations. These operations are essentially unchanged from those listed in *Appendix K.8 and M.8 for the 61BT and 32PT DSCs, respectively*. The shielding sleeve and the bell are designed to not interfere with the NUHOMS® AWS system or other equipment of the canister sealing operations. This is shown in Sequence *6*.



6 CANISTER IS PREPARED FOR CLOSURE OPERATIONS

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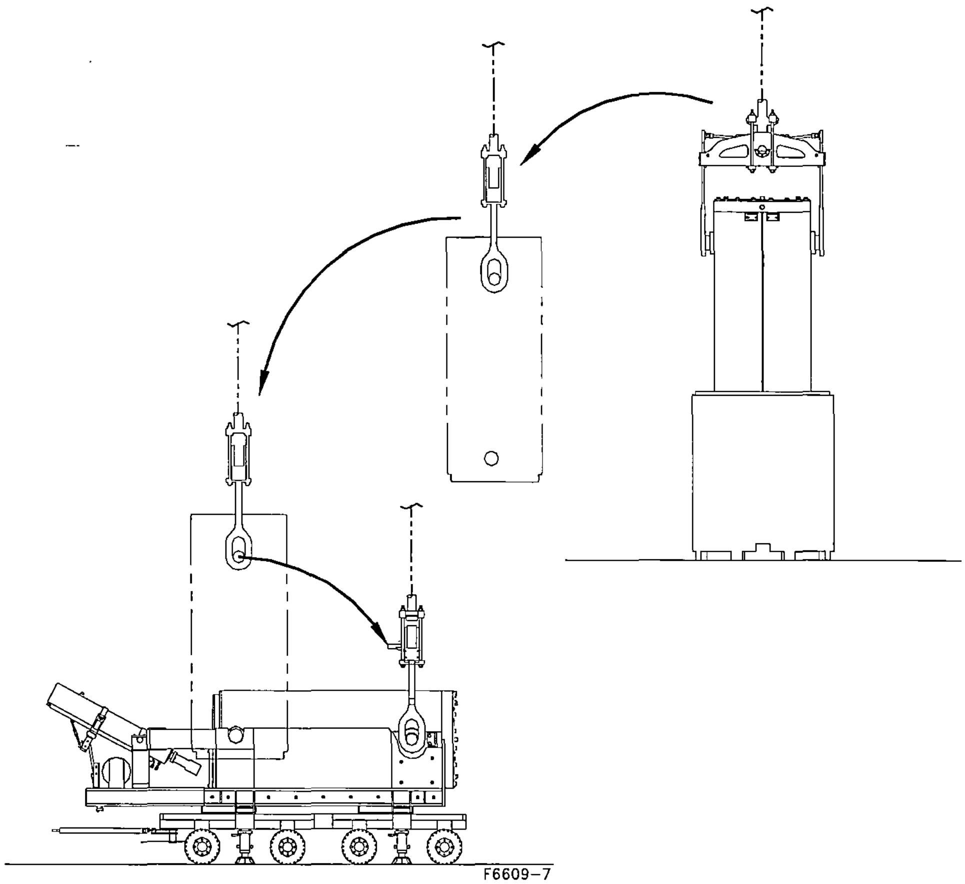
Once the DSC is sealed, the DSC/TC annulus will be drained and the cask top cover installed prior to downending onto the transfer trailer.

CAUTION: The dose rates during the movement of the bare TC from the decontamination area to the transfer trailer may be as high as 9.8 rem/hr.

During the downending process, the bare OS197L TC movement is of short time duration and is performed using remote crane operation and/or an optical targeting system with remote camera monitoring. This remote operation is shown in Sequence 7.



7 CASK IS MOVED FROM DECON AREA TO TRANSFER TRAILER (REMOTE OPERATION)

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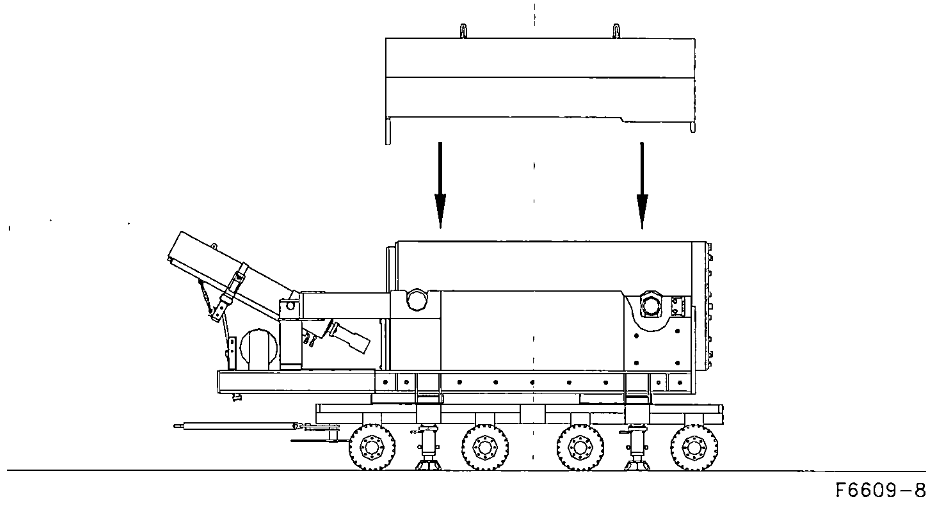
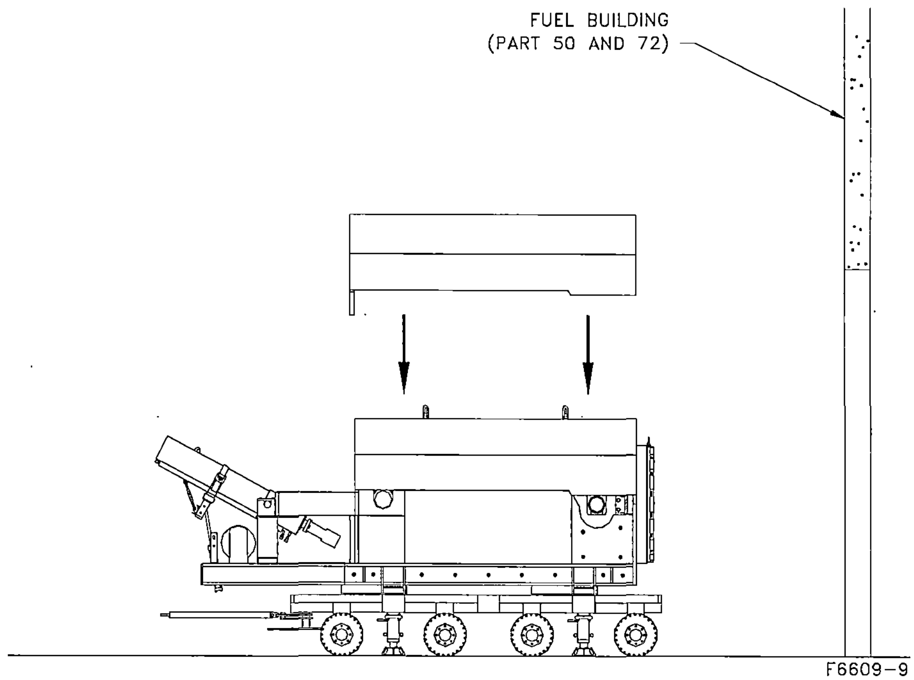
Once on the transfer trailer, the skid provides 5.5" of carbon steel shielding to the sides of the cask up to the trunnions. A 2.5" thick carbon steel shield will be placed over the cask/skid inside the fuel building, after which a 3" thick carbon steel shield will be placed over the 2.5" thick shield providing a total of 5.5" of shielding on the skid. These shields may be placed on the skid inside the fuel handling building, or if load limits exist within the building, the 3" outer shield may be placed on the skid *immediately after* the trailer exits the building. Placement of the inner shields and outer shields on the skid inside the fuel handling building will be performed in accordance with the plants heavy loads procedures, and is evaluated within the plant 72.212 (50.59) for the dry fuel loading process. Sequence *8* shows this remote operation.



8 INSTALLATION OF SUPPORT SKID INNER TOP SHIELDING

As previously indicated, if the fuel building weight limits preclude placement of the outer top skid shielding inside the fuel building, the OS197L TC may be moved out of the fuel building and the outer top skid shielding installed outside the fuel building. This is shown in Sequence 9.

All changes on this page are AMD 11



9 INSTALLATION OF PART 72 SUPPORT SKID OUTER TOP SHIELDING

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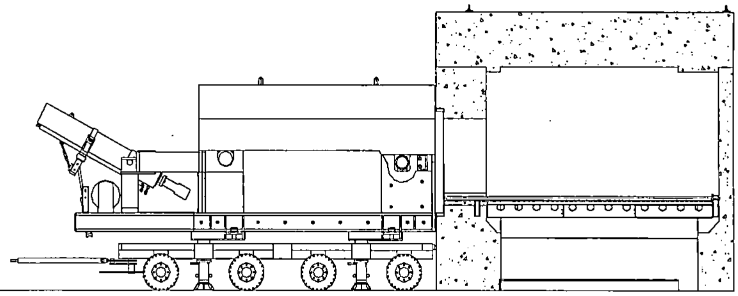
CAUTION: Visually monitor the outer top trailer *skid* vents and the openings around the cask ends for any sign of steaming which may indicate leakage of water from the cask neutron shield. If steaming is determined to be due to leakage of neutron shield water and not due to any rain or snow or other ambient conditions, then Licensee must take appropriate corrective actions including terminating the transfer operation and returning the loaded cask to the fuel handling building for further assessment.

The transfer trailer, with loaded OS197L TC including the supplemental shielding, is then moved to the ISFSI and the cask docked with the HSM. The DSC is then inserted into the HSM using the same methods as the OS197 TC. This is shown in Sequence *10*.



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(10) TRANSFER TRAILER IS DOCKED TO HSM AND CANISTER IS TRANSFERRED

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W.8.1 Procedures for Loading the Cask

Prior to transfer of the loaded OS197L TC from the fuel pool to the decontamination area or from the decontamination area to the transfer trailer, procedures for recovery from a crane malfunction and failure of other remote operations equipment such as the optical targeting system, cameras, etc. shall be developed and in place. The procedures should be developed based on the existing site emergency operating and response procedures which provide guidance for any probable set of events including failure of crane components.

The procedures include preparation of the DSC and fuel loading, closure of the DSC, transfer to the ISFSI using the OS197L TC, DSC transfer into the HSM, monitoring operations, and DSC retrieval from the HSM. The NUHOMS® OS197L transfer equipment, and the existing plant systems and equipment are used to accomplish these operations. Procedures are delineated here to describe how these operations are to be performed and are not intended to be limiting. Standard fuel and cask handling operations performed under the plant's 10CFR50 operating license are described in less detail. Existing operational procedures may be revised by the licensee and new ones may be developed according to the requirements of the plant, provided that the limiting conditions of operation specified in Technical Specifications and the Functional and Operating Limits of the NUHOMS® CoC are not exceeded.

The following sections outline the typical operating procedures for the NUHOMS® OS197L TC system. These generic NUHOMS® procedures have been developed to minimize the amount of time required to complete the subject operations, to minimize personnel exposure, and to assure that all operations required for DSC loading, closure, transfer, and storage are performed safely. Plant specific ISFSI procedures are to be developed by each licensee in accordance with the requirements of 10CFR72.212 (b) and the guidance of Regulatory Guide 3.61 [8.1]. The generic procedures presented here are provided as a guide for the preparation of plant specific procedures and serve to point out how the NUHOMS® OS197L system operations are to be *accomplished. They are not intended to be limiting, in that the licensee may evaluate that alternate acceptable means are available to accomplish the same operational objective.*

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W.8.1.1 Preparation of the TC and DSC

1. *Prior to placement in dry storage, the candidate intact and damaged fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification 2.1.*

2. *Prior to being placed in service, the TC is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification 5.2.4.d.*

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

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

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

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

7. *Record the DSC serial number which is located on the grapple ring. Verify the correct DSC type, basket type and poison material types against the DSC serial number. Verify that the DSC is appropriate for the specific fuel loading campaign per the criteria specified in Technical Specification 2.1.*

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

9. *If damaged fuel assemblies are included in a specific loading campaign, place the required number of bottom end caps provided into the cell locations per Technical Specification 2.1. Optionally, this step may be performed at any prior time.*

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

    For the 32PT DSC, fill the DSC cavity with water from the fuel pool, or an equivalent source which meets the requirements of Technical Specification 3.2.1.

11. *For the 61BT DSC, fill the DSC cavity with water from the fuel pool, an equivalent source, or demineralized water. (Note: this step may be accomplished in the fuel pool).*

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

12. *If not done previously, place the top shield plug onto the DSC and examine the top shield plug to ensure a proper fit. The top shield plug, once fitted is removed and disconnected from the yoke.*

13. *Position the cask lifting yoke above the transfer cask and engage the cask lifting trunnions and the rigging cables to the DSC top shield plug. Adjust the rigging cables as necessary to obtain even cable tension.*

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

15. *Provide for later connection to the vacuum drying system (VDS) or an optional water draining/pumping device to the siphon port of the DSC and position any connecting hose such that the hose will not interfere with loading (yoke, fuel, shield plug, rigging, etc.). A flowmeter or other suitable means for measuring the amount of water removed must be installed at a suitable location as part of this water removal system.*

16. *Move the scaffolding away from the cask as necessary.*

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

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

    b. Fill the TC liquid neutron shield. This step may be completed at any time prior to immersion of the TC/DSC into the pool.

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

W.8.1.2 *<u>DSC Fuel Loading</u>*

Note: The licensee shall verify that the lifting device used for handling the OS197L TC meets the requirements of the sites lifting program. Licensee shall use remote operations and optical targeting system and other mitigating ALARA practices when handling the bare OS197L TC when loaded with fuel as required by the sites ALARA program and the Radiation Protection Program requirements of Technical Specification 5.2.4.a.

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

All changes on this page are AMD 11

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

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

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

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

    - *A cask/DSC loading plan is developed to verify that the damaged and/or intact fuel assemblies and CCs, if applicable, meet the burnup, enrichment, and cooling time parameters of Technical Specification 2.1.*

    - *The loading plan is independently verified and approved before the fuel load.*

    - *A fuel movement plan is then written, verified, and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance of the fuel movement plan.*

    - *For 61BT DSC only, if loading damaged fuel assemblies, verify that the required number of bottom end caps is installed in appropriate fuel compartment tube locations of 61BT DSC before fuel load.*

6. *Prior to loading of a spent fuel assembly (and CCs, if applicable) into the DSC, the identity of the assembly (and CCs, if applicable) is to be verified by two individuals using an underwater video camera or other means. Verification of CC identification is optional if the CC has not been moved from the host fuel assembly since its last verification. Read and record the identification number from the fuel assembly (and CCs, if applicable) and check this identification number against the DSC loading plan which indicates which fuel assemblies (and CCs, if applicable) are acceptable for dry storage.*

7. *Move a candidate fuel assembly from a fuel rack in accordance with the plant's 10CFR50 fuel handling procedures.*

8. *Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Step 5 through 7 for each SFA loaded into the DSC. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly and CCs, if applicable, in the DSC. If loading damaged fuel assemblies, place top end caps over each damaged fuel assembly placed into the basket.*

9. *After all the SFAs and CCs, if applicable, have been placed into the DSC and their identities verified, place the hold down ring (61BT only), position the lifting yoke and the*

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top shield plug and lower the shield plug onto the DSC. Note that separate rigging may be used to install the shield plug prior to engaging the trunnions with the lifting yoke.

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

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

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

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

13. *Inspect the top shield plug to verify that it is properly seated onto the DSC. If not, lower the cask and reposition the top shield plug. Repeat Steps 9 through 13 as necessary.*

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

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

16. *Take a preliminary measurement of the OS197L TC dose rates at 3 feet from the top of the cask with the shield plug installed and water in the DSC cavity.*

16a. *Disconnect the top shield plug rigging from the shield plug.*

CAUTION: Prior to the next step, evacuate personnel from the area, as specified by plant's ALARA practices, due to the high cask dose rates. Crane operations shall be performed remotely and an optical targeting system with remote camera monitoring shall be used to minimize personnel exposure.

CAUTION: The surface dose rates during the movement of a bare loaded TC from the spent fuel pool to the decontamination area supplemental shielding may be as high as 9.8 rem/hr.

17. *Lift the TC from the fuel pool.*

18. *Move the TC with loaded DSC to the cask decon area and carefully place it in the decontamination area shielding sleeve.*

18a. *Disengage the lifting yoke from the trunnions and position it clear of the cask.*

19. *Place the decontamination area shielding bell over the side of the cask above the upper trunnions. Placement of the shielding bell shall be performed in accordance with the plant's heavy load procedures. The shielding sleeve and bell provide the additional shielding to produce similar shielding as the OS197 TC.*

All changes on this page are AMD 11

20. If applicable to keep the occupational exposure ALARA, temporary shielding may be installed as necessary to minimize personnel exposure. Install cask seismic restraints if required by Technical Specification 4.3.3 7. (Required only on plant specific basis).

21. Remove the lifting lugs from the top shield plug.

W.8.1.3 <u>DSC Drying and Backfilling</u>

CAUTION: During performance of steps listed in Section W.8.1.3, monitor the TC/DSC annulus water level and replenish if necessary until drained.

CAUTION: During performance of steps listed in Section W.8.1.3, the opening at the top and bottom of the decontamination area shielding shall be monitored (visual inspection) to assure no significant blockage of openings. Although blockage is improbable as all 16 openings would require sealing, personnel shall perform visual inspection of shielding sleeve and bell openings during the operations when DSC is in the sleeve.

1. Check the radiation levels along the perimeter of the cask/shields. The cask exterior surface should be decontaminated by providing spray mechanisms on the inside of the shield bell or other methods, using good ALARA practices. Water mixed with a commercial decontamination agent may be used for decontamination. Install additional temporary shielding as necessary to minimize personnel exposure.

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

3. Decontaminate the exposed surfaces of the DSC shell perimeter above the TC/DSC annulus seal. Remove the inflatable TC/DSC annulus seal.

3a. In accordance with Technical Specification 5.2.4.a, verify that the neutron shield (NS) is filled before the draining operation in Step 4 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.

4. Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately twelve inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with Technical Specification 5.2.4.d, taking corrective actions in accordance with that technical specification, if required, potentially involving removal of the fuel assemblies, removal of the DSC from the TC, and decontamination of the entire length of the DSC outer surface.

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

4a. In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 5 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.

All changes on this page are AMD 11

5. *Drain approximately the number of gallons of water shown in the table below (as indicated on a flowmeter) from the DSC back into the fuel pool or other suitable location. Consistent with ISG-22 [8.5] guidance and Technical Specification 3.1.1, helium at 1-3 psig is used to backfill the DSC with an inert gas (helium) as water is being removed from the DSC. Only helium may be used to assist in the removal of water.*

DSC	*Gallons of Water*
32PT	*750*
61BT	*1100*

Alternatively, if a slow helium purge is used while monitoring for hydrogen, less than these amounts of water may be drained, because this approach will prevent buildup of flammable gas to a flammability limit.

6. *Disconnect hose from DSC siphon port.*

7. *Install the automatic welding machine onto the inner top cover plate and place the inner top cover plate with the automatic welding machine onto the DSC. Optionally, the inner top cover plate and the automatic welding machine can be placed separately. Verify proper fit-up of the inner top cover plate with the DSC shell.*

8. *Check radiation levels along surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.*

9. *Insert tubing of sufficient length and adequate temperature resistance through the vent port such that it terminates just below the DSC shield plug. Connect the tubing to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner cover plate, in compliance with Technical Specification 5.2.6. Optionally, other methods may be used for continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner top cover plate, to comply with the Technical Specification.*

10. *Cover the cask/DSC annulus to prevent debris and weld splatter from entering the annulus.*

11. *Ready the automatic welding machine and tack weld the inner top cover plate to the DSC shell. Install the inner top cover plate weldment and remove the automatic welding machine.*

    CAUTION: Continuously monitor the hydrogen concentration in the DSC cavity using the arrangement or other alternate methods described in Step 9 during the inner top cover plate cutting/welding operations. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [8.2 and 8.3]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with approximately 2-3 psig helium via the tubing to reduce the hydrogen concentration safely below the 2.4% limit.

All changes on this page are AMD 11

All changes on this page are AMD 11

12. *Perform dye penetrant weld examination of the inner top cover plate weld in accordance with the Technical Specification 5.2.4.b requirements.*

13. *Remove purge lines and connect the VDS to the DSC siphon and vent ports.*

14. *Install temporary shielding to minimize personnel exposure throughout the subsequent welding operations as required.*

15. *Install Strongback on 61BT. (Strongback is optional for 32PT.)*

    a. *Place strongback so that it sits on the inner top cover plate and is oriented such that:*

    - *the DSC siphon and vent ports are accessible*
    - *the strongback stud holes line up with the TC lid bolt holes.*

    b. *Lubricate the studs and, using a cross pattern, adjust the strongback studs to snug tight ensuring approximately even pressure on the cover plate.*

16. a. *In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in this step is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.*

    b. *If using blowdown method to remove water, engage helium supply (up to 15 psig) and open the valve on the vent port and allow helium to force the water from the DSC cavity through the siphon port. Use of helium is required per Technical Specification 3.1.1.*

    c. *If using water pumps to remove water without blowdown, pump water from DSC. Use 1 to 3 psig of helium to backfill the DSC with helium per ISG-22 [8.5] guidance and Technical Specification 3.1.1 as water is being removed from the DSC cavity.*

17. *Once the water stops flowing from the DSC, close the DSC siphon port and disengage the helium source and/or turn off the suction pump, as applicable. Verify that the TC axial surface dose rates are compliant with limits specified in Technical Specification 5.2.4.e. The configuration for determining the TC axial surface dose rates shall be in accordance with Technical Specification 5.2.4.e.*

    CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

18. *Connect the hose from the vent port and the siphon port to the intake of the vacuum pump. Connect a hose from the discharge side of the VDS to the plant's radioactive waste system or spent fuel pool. Connect the VDS to a helium source.*

    Note: Proceed cautiously when evacuating the DSC to avoid freezing consequences.

19. *Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the DSC cavity. The cavity pressure should be reduced in steps of approximately 100 mm Hg,*

50 mm Hg, 25 mm Hg, 15 mm Hg, 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level (these levels are optional), the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg absolute or less as specified in Technical Specification 3.1.1.

All changes on this page are AMD 11

Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with Technical Specification 3.1.1 requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draws a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC.

20. *Open the valve to the vent port and allow the helium to flow into the DSC cavity.*

    Note: Steps 21 through 25 which follow are optional steps that provide an initial check of the inner top cover plate weld.

21. *Pressurize the DSC with helium up to about 24 psia not to exceed 34 psia.*

22. *Helium leak test the inner top cover plate weld to a leak rate of 1×10^{-5} atm cm^3 / sec.*

23. *If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.*

24. *Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.*

25. *Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored level (these levels are optional). When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg absolute or less in accordance with Technical Specification 3.1.1 limits.*

    CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

    Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with Technical Specification 3.1.1 requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draws a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to *atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC.*

26. *Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC as shown in the table below (Value A) and hold for 10 min. Depressurize the DSC cavity by releasing the helium through the VDS to the plant spent fuel pool or radioactive waste system to value shown in table below (Value B) in accordance with Technical Specification 3.1.2 limits.*

DSC	*Value A (psig)*	*Value B (psig)*
32PT	*16.5 – 18.0*	*2.5 ±1.0*
61BT	*11.0 – 12.5*	*2.5 ±1.0*

27. *Close the valves on the helium source.*

28. *Remove the strongback, decontaminate as necessary, and store (if used).*

W.8.1.4 *DSC Sealing Operations*

CAUTION: During performance of steps listed in Section W.8.1.4, monitor the cask/DSC annulus water level and replenish as necessary to maintain cooling.

CAUTION: During performance of steps listed in Section W.8.1.4, the opening at the top and bottom of the decontamination area shielding shall be monitored (visual inspection) to assure no significant blockage of openings. Although blockage is improbable as all 16 openings would require sealing, personnel shall perform visual inspection of shielding sleeve and bell openings during the operations when DSC is in the sleeve.

1. *Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports. Inject helium into blind space just prior to completing welding, and perform a dye penetrant weld examination in accordance with the Technical Specification 5.2.4.b requirements.*

2. *Temporary shielding may be installed as necessary to minimize personnel exposure. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Optionally, outer top cover plate may be installed separately from the welding machine. Verify proper fit up of the outer top cover plate with the DSC shell.*

3. *Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.*

4. *Perform a helium leak test of the DSC inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical*

All changes on this page are AMD 11

Specification 5.2.4.c limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.4]. Alternatively this can be done with a test head in step 1 of Section W.8.1.4.

5. *If a leak is found, remove the outer cover plate root pass (if not using test head), the vent and siphon port plugs and repair the inner cover plate welds. Repeat procedure steps from W.8.1.3 Step 20.*

6. *Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification 5.2.4.b requirements.*

7. *Install and seal weld the prefabricated plug, if applicable, over the outer cover plate test port and perform dye penetrant weld examinations in accordance with Technical Specification 5.2.4.b requirement.*

8. *Remove the automatic welding machine from the DSC.*

8a. *In accordance with Technical Specification 5.2.4.a, verify that the NS is filled before the draining operation in Step 9 is initiated and continually monitored during the first five minutes of the draining evolution to ensure the NS remains filled.*

9. *Drain the DSC/TC annulus.*

10. *Rig the cask top cover plate and lower the cover plate onto the TC.*

11. *Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.*

12. *Verify that the TC radial dose rates measured at the surface of the decontamination area shielding are compliant with limits specified in Technical Specification 5.2.4.e. The configuration for determining the TC radial surface dose rates shall be in accordance with Technical Specification 5.2.4.e.*

W.8.1.5 TC Downending and Transfer to ISFSI

Note: Licensee shall use remote operations and other mitigating ALARA practices when handling the bare OS197L TC when loaded with fuel as required by the sites ALARA program and the Radiation Protection Program requirements of Technical Specification 5.2.4.a.

Note: Alternate Procedure for Downending of Transfer Cask: Some plants have limited floor hatch openings above the cask/trailer/skid, which limit crane travel (within the hatch opening) that would be needed in order to downend the TC with the trailer/skid in a stationary position. For these situations, alternate procedures are to be developed on a plant-specific basis, with detailed steps for downending.

CAUTION: The surface dose rates during the movement of the bare TC from the decontamination area to the transfer trailer may be as high as 9.8 rem/hr.

All changes on this page are AMD 11

CAUTION: Restrict personnel from the area, as specified by plant's ALARA practices, due to the high cask dose rates. Crane operations shall be performed remotely and an optical targeting system with remote camera monitoring shall be used to minimize personnel exposure upon removal of the decontamination area shielding bell from the cask. Failure of remote operating devices should be considered and proper repair/recovery operations should be planned to keep doses ALARA.

1. *Verify that the neutron shield is filled, in accordance with Technical Specification 5.2.4.a.*

2. *Move the scaffolding away from the cask as necessary.*

3. *Rig and remove the decontamination area shielding bell from the cask. Removal of the shielding bell shall be performed in accordance with the plant's heavy load procedures.*

4. *Re-attach the TC lifting yoke to the crane hook, as necessary. Ready the transport trailer and cask support skid for service.*

5. *The transport trailer should be positioned so that cask support skid is accessible to the crane with the trailer supported on the vertical jacks.*

6. *Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.*

7. *Move the crane forward while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.*

8. *Inspect the positioning of the cask to insure that the cask and trunnion pillow blocks are properly aligned.*

9. *Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.*

10. *Place the inner top shield on the skid (this must be performed inside the fuel handling building). Placement of the shields shall be performed in accordance with the plant's heavy load procedures and shall be evaluated within the plant 72.212 (50.59) for the dry fuel loading procedures.*

11. *Inspect the trunnions to ensure that they are properly seated onto the skid and install the trunnion tower closure plates, if required.*

    CAUTION: Per Technical Specification 4.4.4, during transfer operation of a loaded OS197L TC, every hour, visually monitor the outer top trailer shield vents and the opening around the cask ends for any sign of steaming which may indicate leakage of water from the cask neutron shield. If steaming is determined to be due to leakage of neutron shield water and not due to any rain or snow or other ambient conditions, then Licensee shall take appropriate corrective actions including terminating the transfer operation and returning the loaded cask to the fuel handling building for further assessment.

All changes on this page are AMD 11

The following step may be performed outside if the fuel building weight limits preclude placement of the outer top skid shielding inside the fuel building (See Technical Specification 4.4.3 for restrictions).

CAUTION: Verify that the requirements of Technical Specification 5.3.1b are met prior to next step.

12. *Install the outer top skid shielding. During installation, the bottom most part of the body of the outer top shield shall not be hoisted by the crane more than 4 inches above the top horizontal plate of the inner top shield.*

W.8.1.6 <u>DSC Transfer to the HSM</u>

CAUTION: Per Technical Specification 4.4.4, during transfer operation of a loaded OS197L TC, every hour, visually monitor the outer top trailer shield vents and the opening around the cask ends for any sign of steaming which may indicate leakage of water from the cask neutron shield. If steaming is determined to be due to leakage of neutron shield water and not due to any rain or snow or other ambient conditions, then Licensee shall take appropriate corrective actions including terminating the transfer operation and returning the loaded cask to the fuel handling building for further assessment.

CAUTION: During the actual movement of the transfer cask on the transfer trailer to the ISFSI, the gap between the transfer trailer deck and bottom of the skid shall be monitored (visual inspection) to assure no significant blockage of airflow. Although blockage is improbable as over 60 feet of gap would require sealing, personnel shall maintain a visual scan of the trailer.

1. *Prior to transferring the cask to the ISFSI or prior to positioning the transfer cask at the HSM designated for storage, remove the HSM door using a porta-crane, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.*

CAUTION: Very high dose rates in the empty HSM are expected if adjacent to a loaded HSM. Proper ALARA practices should be followed during these operations.

2. *Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.*

CAUTION: Verify that the requirements of Technical Specification 5.3.1b are met prior to next step.

3. *Using a suitable vehicle, transfer the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.*

4. *Once at the ISFSI, position the transport trailer to within several inches of the HSM.*

All changes on this page are AMD 11

5. *Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.*

6. *If not already installed, install the alignment targets, including the cast top centerline target through the skid shielding.*

7. *Using crane, unbolt and remove the cask top cover plate.*

8. *Back the cask to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer. Extend the transfer trailer vertical jacks.*

9. *Remove the skid tie-down bracket fasteners and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.*

10. *Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.*

11. *Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.*

12. *After the cask is docked with the HSM, verify the alignment of the TC using the optical survey equipment.*

13. *Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug if installed. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.*

14. *Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.*

15. *Recheck all alignment marks in accordance with the Technical Specification 5.3.3 limits and ready all systems for DSC transfer.*

16. *Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.*

17. *Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.*

18. *Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.*

19. *Using the skid positioning system, disengage the cask from the HSM access opening.*

20. *Install the DSC axial retainer through the HSM door opening.*

All changes on this page are AMD 11

21. *Install the HSM door using a portable crane or other suitable lifting device and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Technical Specifications 5.4.1 and 5.4.2.*

22. *Replace the TC top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.*

23. *Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.*

24. *Close and lock the ISFSI access gate and activate the ISFSI security measures.*

25. *Ensure the HSM maximum air exit temperature requirements of Technical Specification 3.1.4 are met.*

All changes on this page are AMD 11

W.8.1.7 Monitoring Operations

1. *Perform routine security surveillance in accordance with the licensee's ISFSI security plan.*

2. *Perform one of the two alternate daily surveillance activities listed below:*

   a. *A daily visual surveillance of the HSM air inlets and outlets to insure that no debris is obstructing the HSM vents in accordance with Technical Specification 5.2.5a requirements.*

   b. *A temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification 5.2.5b requirements.*

W.8.2 Procedures for Unloading the Cask

The operational differences specified above for loading operations when using OS197L TC (relative to the use of OS197 TC described in Chapter 5) will also apply for unloading operations.

W.8.3 Identification of Subjects for Safety Analysis

There is no change relative to Section 5.1.3 regarding criticality control, chemical safety, operational shutdown modes and maintenance techniques.

In addition to the typical instrumentation listed in Table 5.1-1 of Section 5.1.3, the use of OS197L TC shall require optical targets and instruments to implement specific remote crane operations described in Section W.8.1 above.

W.8.4 Fuel Handling Systems

W.8.4.1 Spent Fuel Handling and Transfer

No change. The general description of the spent fuel handling and transfer components presented in Section 5.2.1 remains applicable. However, there are differences in design features and operations as noted below.

W.8.4.1.1 Functional Description

The OS197L TC and the associated transfer equipment are described in sections W.1.2 and W.1.2.1.1. The difference in design features of these components relative to the OS197 TC described in Section 5.2.1.1 are discussed in Appendix W.1.

Note: There are significant differences in operation steps when using the OS197L TC. Hence, the illustration provided in Figure 5.2-1 is not applicable when using the OS197L TC.

W.8.4.1.2 Safety Features

No change. The safety features described in Section 5.2.1.2 remain applicable.

W.8.4.2 Spent Fuel Storage

Descriptions of the operations used for the transfer and retrieval of the DSC from the HSM are presented in Section W.8.1. There is no change to the safety features discussed in Section 5.2.1.2.

W.8.5 Other Operating Systems

No change is needed to the information provided in SAR Section 5.3 except that the accident analysis is provided in Section W.11 for the OS197L system.

W.8.6 Operation Support System

No change to the information provided in Section 5.4 except for the optical targets and facilities required for remote crane operations described in Section W.8.

W.8.7 Control Room and/or Control Areas

No change to the information provided in Section 5.5.

W.8.8 Analytical Sampling

No change to the information provided in Section 5.6.

All changes on this page are AMD 11

All changes on this page are AMD 11

W.8.9 References

8.1 U.S. Nuclear Regulatory Commission, "Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Container," Regulatory Guide 3.61 (February 1989).

8.2 U.S. Nuclear Regulatory Commission, Office of the Nuclear Material Safety and Safeguards, "Safety Evaluation of VECTRA Technologies' Response to Nuclear Regulatory Commission Bulletin 96-04 for the NUHOMS®-24P and NUHOMS®–7P.

8.3 U.S. Nuclear Regulatory Commission Bulletin 96-04, "Chemical, Galvanic or Other Reactions in Spent Fuel Storage and Transportation Casks," July 5, 1996.

8.4 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing."

8.5 U.S. Nuclear Regulatory Commission, Interim Staff Guidance (ISG-22), "Potential Rod Splitting due to Exposures to an Oxidizing Atmosphere during Short-term Cask Loading Operations in LWR of Other Uranium Oxide Based Fuel."

W.9 Acceptance Criteria and Maintenance Program

All changes on this page are AMD 11

The use of the OS197L includes the acceptance criteria, and maintenance program requirements described elsewhere in the UFSAR for the OS197 TC, with the follow clarifications and additional considerations unique to the OS197L.

As described in Section W.3.2, the test load criteria for the upper trunnions of the OS197L are the same as those for the OS197 TC, except that the test load is conservatively equal to 300% of the design load instead of 150% for the OS197 TC in Section 4.2.3.3.

Consistent with Section 4.3.9 the transfer cask is designed to require only minimal maintenance, limited to periodic inspection of critical components and replacement of damaged or nonfunctioning components. A discussion of these requirements is provided in Section 4.5. Section 4.5 of the UFSAR describes routine and annual visual/functional inspections and the repair or replacement actions to be taken if any of the inspections fail. For the OS197L the routine inspections should include the supplemental shielding (visual inspections for damage) and the mechanical and hydraulic/pneumatic components (visual and functional inspection) of the lifting yokes or other hardware used for remote operations when using the OS197L TC.

In addition, the applicable pre-operational testing requirements described in Section 9.2, and the training requirements in Section 9.3 are applicable when the OS197L TC is used, with the following additional requirements:

1. *The pre-operational testing and operation programs for licensees using the OS197L system should specifically include the unique operational features of the OS197L system based on normal planned operating procedures and based on procedures for recovery from a crane malfunction and failure of other remote operations equipment such as the optical targeting system, cameras, etc. The unique operational features should include as a minimum: 1) operations with the yoke to engage and disengage the TC trunnions during operations at the cask handling area and during upending/downending, 2) use of supplemental shielding in the cask handling area and on the transfer trailer/skid, and 3) operations including remote targeting or any other hardware needed for remote handling. These unique operational features for normal and recovery conditions should be included in dry run or practice exercises prior to each loading campaign, even if done at the same site with the same personnel.*

2. *The training programs for licensees using the OS197L system should specifically include the unique operational features of the OS197L system based on normal planned operating procedures and based on procedures for recovery from a crane malfunction and failure of other remote operations equipment such as the optical targeting system, cameras, etc.*

The OS197L payloads are the 61BT and 32PT DSCs. The acceptance tests and maintenance programs for these DSCs as described in Chapters K.9 and M.9 of the UFSAR are not changed by the use of the OS197L TC.

All changes on this page are AMD 11

W.10 Radiation Protection

As discussed in Section W.5, use of the OS197L TC does not significantly affect personnel doses (during closure operations, handling, or storage) or site boundary dose rates *during normal operations with supplemental shielding as described in Section W.1.* The OS197L TC is used only for loading/unloading and transfer operations, and the storage conditions are unchanged. Therefore, the personnel doses, occupational exposures and site bounding dose rates documented for each DSC/HSM storage configuration in Chapter K.10 *and* Chapter M.10 remain unchanged and are applicable to operations using the OS197L TC.

The use of the OS197L TC is not expected to have any significant impact on personnel doses during normal operation since the operations for placement and removal of bare OS197L TC from the fuel pool into the decontamination area shielding sleeve, placement and removal of the decontamination area shielding bell, engagement of the yoke to the cask trunnions, movement of the cask to the trailer, lowering of the cask onto the trailer and placement of the *skid* shielding on the cask will be performed remotely using cameras and laser/target positioning.

The OS197L uses remote handling devices for movement of the TC during loading and transfer operations. In the event of a failure of *either* the remote handling device *or a malfunction of the crane,* an evaluation has been performed to assess the additional occupational exposure during recovery operations. This evaluation was performed using the dose rates from the OS197L TC when loaded with a NUHOMS® 32PT DSC.

Additional occupational exposures due to operations unique to the OS197L TC are evaluated in the following sections:

W.10.1 Recovery/Repair Operations due to Remote Handling Device Malfunction

The OS197L uses remote handling devices for movement of the TC during loading and transfer operations. In the event of a failure of *either* the remote handling device *or a malfunction of the crane,* an evaluation has been performed to assess the additional occupational exposure during recovery operations. This evaluation was performed using the dose rates from the OS197L TC when loaded with a NUHOMS® 32PT DSC.

For this evaluation, the crane is postulated to fail as the OS197L TC is being lowered onto either the decontamination area or the transfer trailer.

Dose rates around various shielding configurations are presented in Chapter W.5. The dose rate distribution utilized in the occupational exposure calculations during a crane malfunction can be estimated using equations provided below.

The bounding dose rate at a radial distance, r, from the side of a bare cask can be approximated by the expression:

$$D(\vec{r}) = \sum_{i=0}^{3} a_i \cdot \exp(-b_i \cdot r) \qquad Eq.\ W.10\text{-}1$$

where the values for a_i and b_i are given in Table W.10-1 for $0 \le r \le 10$ and $10 < r \le 200$. This equation represents dose rates that are in a very good agreement with data in Table W.5-3, Table W.5-6 and Table W.5-9 (OS197L TC Bare Cask).

Table W.10-1
Parameters of Eq. W.10-1

Radial Distance (meters)	Parameters ([a] = mrem/hr and [b] = m^{-1})							
	a_0	b_0	a_1	b_1	a_2	b_2	a_3	b_3
$0 \le r \le 10$	1.9339	0.0	2696.5243	2.7571	5562.0454	0.7258	1574.6363	0.1965
$10 < r \le 200$	0.0085	0.0	11.0493	0.0208	569.6182	0.0963	0	0

Similarly, the maximum dose rate within the outer radius of the cask, R, at a distance, H (see Figure W.5-3), above the cask top in the axial direction can be approximated by the expressions:

$$D(H) = (25.2 - 107) * H + 107, \text{ for } r < R \text{ and } 0 \le H \le 1 \text{ meter} \qquad \text{Eq. W.10-2a}$$

$$D(H) = 26 / H, \text{ for } r < R \text{ and } 1 < H < 10 \text{ meters} \qquad \text{Eq. W.10-2b}$$

To provide an example of how exposure to a worker can be calculated, consider a bare cask hang-up due to a crane malfunction which requires a worker to perform some manual operations on the crane so that the cask can be lowered and laid on its side on a trailer. It is assumed that this occurs in a large room and at a distance from the walls and ceiling sufficient to ignore backscatter, so no backscatter contributions are included in this example. However, the significance of backscatter contributions should always be considered when calculating this type of occupational exposure. Factors such as a small room geometry (defined as having dimensions less than twice the length of the cask) or operations performed close to the wall or ceiling may require the use of backscatter correction factors.

For this scenario, a worker must perform the following steps:

1. *Use a ladder to reach the elevation of a crane bridge,*
2. *Use a walkway to reach the crane bridge,*
3. *Crawl on the crane bridge itself to reach a location just above the cask top,*
4. *Move to the cask top and perform necessary repair operation on the crane or cask,*
5. *Return to the crane bridge,*
6. *Remain in place on the crane bridge while the cask is lowered to the trailer and tilted on its side,*
7. *Crawl back across the crane bridge to the walkway,*
8. *Use the walkway to reach the ladder, and*
9. *Descend the ladder.*

All changes on this page are AMD 11

All changes on this page are AMD 11

Figure W.10-1 depicts scenario as seen in a plan view.

It is assumed that the elevation of the crane bridge, the length of the walkway, and the distance from the walkway to the cask are all 15.24 m (50 ft). It is also assumed that the top of the cask is 1 m below the crane bridge when the hang-up occurs and that the trailer is 0.91 m (3 ft) above the ground. When applying this type of calculation to a specific site, it is important to use the actual distances expected to be encountered during the operation.

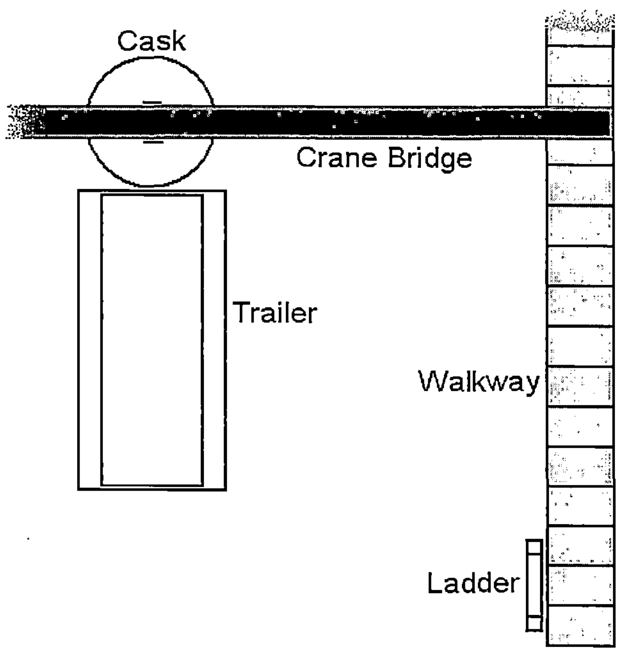

Figure W.10-1

1. *The ladder is at a radial distance of 21.55 m from the side of the cask. A dose rate of 79 mrem/hr is predicted at that distance. Assuming a constant climbing rate of 1550 m/hr (1 mile/hr) [10.1, 10.2], 35 secs are required to climb the ladder, which exposes the worker to **0.77** mrem.*

2. *As the worker moves along the walkway, the radial distance to the cask decreases from 21.55 m to 15.24 m. Conservatively, it is assumed that the worker is exposed to the maximum dose rate of 139 mrem/hr at a radial distance of 15.24 m. Assuming a constant walking speed of 4830 m/hr (3 miles/hr), 11 secs are required to reach the crane bridge, which exposes the worker to **0.44** mrem.*

3. *As the worker crawls over the crane bridge, the radial distance to the cask decreases linearly from 15.24 m to 0 m. Assuming a constant crawling speed of 966 m/hr or one fifth the normal walking speed, 57 secs are required to reach the top of the cask, which exposes the worker to **17.03** mrem.*

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4. *The operations are performed directly on top of the cask, where the predicted dose rate is 107 mrem/hr. Assuming 10 minutes are required to reach the top of the cask from the crane bridge and perform the operations, the worker is exposed to* ***17.83*** *mrem.*

5. *As the cask is lowered, the axial distance increases from 0 m to 8.25 m. Assuming the cask is lowered at a constant rate and requires 0.75 hrs, the worker is exposed to 11.00 mrem.*

6. *As the cask is laid on its side, the worker transitions from 8.25 m above the cask top to 11.19 m from the cask side. Conservatively it is assumed that the worker is exposed to the maximum predicted dose rate of 203 mrem/hr that occurs at a radial distance of 11.19 m from the side of the cask. Assuming the cask it tilted over a period of 0.5 hrs, the worker is exposed to* ***101.33*** *mrem.*

   Note: The following sequence of steps 7 through 10 are the reverse of step 4 through 1 above. Speed and completion times remain unchanged.

7. *Operations are performed at a radial distance of 11.19 m from the cask, which corresponds to a dose rate of 203 mrem/hr. The worker is exposed to* ***33.78*** *mrem.*

8. *As the worker crawls back across the crane bridge, the radial distance to the cask increases from 11.19 m to 18.91 m. Conservatively, it is assumed that the worker is exposed to the maximum predicted dose rate of 203 mrem/hr at a radial distance of 11.19 m. The worker is exposed to* ***3.20*** *mrem.*

9. *The walkway is now at a radial distance of 18.91 m from the cask where the predicted dose rate is 100 mrem/hr. The worker is exposed to* ***0.31*** *mrem.*

10. *As the worker descends the ladder, the radial distance to the cask decreases from 18.91, passing at a minimum at 15.24 m. Conservatively it is assumed that the worker is exposed to the maximum predicted dose rate of 139 mrem/hr at a radial distance of 15.24 m. The worker is exposed to* ***1.37*** *mrem.*

The summation of the total exposure received by a worker from the above 10 steps results in a total exposure of ***186.96*** *mrem. If more than one worker is involved in conducting the above described repair/recovery operations, the exposure must be scaled accordingly.*

W.10.2 Inspection of Decontamination Shield Openings for Blockage

In addition to the operations that are performed using remote handling equipment, there are additional minor operational steps that are necessitated due to the OS197L TC. For example, an operational step is needed during decontamination operations to ensure that the "openings" in the decontamination area shield are not blocked. The bottom openings are located at the bottom radial location of the decontamination area shield and the top openings are located at the top radial location of the decontamination area shielding bell. A discussion of the dose rate results associated with cask decontamination is included in Section W.5.4.10.2 of the SAR.

The bare cask radial surface dose rates at these axial locations are conservatively employed to determine the dose rate fields. The maximum dose rate near the top opening using the bare cask *radial dose rate results from Table W.5-8 and Table W.5-11 is approximately 3000 mrem/hr (at a distance greater than 500 cm from the bottom of the OS197L TC). The maximum radial dose rate at the side of the OS197L TC during decontamination as displayed in Figure W.5-2 (results corresponding to curve 4) is less than 100 mrem/hr. The average dose rate at the top opening location is conservatively estimated to be less than 700 mrem/hr (Section W.5.4.10.2).*

Since this operation is of the order of a minute or less, the total contribution to the occupational exposure is less than 12 mrem. Workers are expected to follow the appropriate ALARA practices while performing this step, particularly at the top of the OS197L TC.

All changes on this page are AMD 11

W.10.3 References

10.1 Shrawan Kumar, "Biomechanics in Ergonomics," TAYLOR & FRANCIS, Copyright 1999, ISBN-10 0-7484-0704-9.

10.2 US 29 CFR, "Occupational Safety and Health Standards," Standard Number: 1910.27.

All changes on this page are AMD 11

W.11 Accident Analyses

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This section describes the postulated accident events that could occur during fuel loading, draining, drying, welding and transfer of the DSC using a NUHOMS® OS197L TC. Sections which do not affect the evaluation*s* presented in Appendices K.11 *and* M.11 for *the NUHOMS®-61BT and 32PT* DSC designs are identified as "No change." Detailed analys*es* of the events are provided in other sections and are referenced herein. *The cask support skid supplemental shielding is referred to as trailer shielding throughout this chapter.*

W.11.1 Postulated Accidents

Only those accidents affecting the OS197L TC are addressed in this section. There is no change to accident evaluations affecting other NUHOMS® components.

W.11.1.1 OS197L TC Missile Impact Analysis

This event is described in Section 8.2.2.4. The OS197L TC uses a 2.68" steel shell in lieu of a 1.5" steel shell with a nominal 3.5" lead annulus and a 0.5" inner liner for OS197 TC. The missile impact analyses for the OS197 TC are therefore bounding for the OS197L TC.

W.11.1.2 Earthquake

This event is described in Section 8.2.3.D. The OS197L TC configuration (cg location, cask length, trunnion location and bottom forging configuration) does not significantly differ from that of the OS197 TC. The OS197L TC remains stable when subjected to the design basis earthquake.

W.11.1.2.1 OS197L TC in a Vertical Configuration during Vacuum Drying and Welding Operations

The bottom forging on which the cask is resting during vertical cask operations is the same size and configuration as the OS197 TC. The OS197L TC cg location is not significantly altered by the change in the cask shell configuration. The addition of the decontamination area shield will provide a larger diameter, more stable shell, outside the cask envelope, thereby potentially enhancing the OS197L TC seismic capacity.

W.11.1.2.2 OS197L TC in a Horizontal Configuration during Transfer Operations

The cask seismic stresses for the OS197L TC are bounded by the OS197 TC stresses due to the similar configurations of the cask ends (top and bottom forgings and covers) and larger thickness structural shell.

The trailer with the OS197L TC, with the additional shielding, remains stable for the design basis seismic accelerations.

W.11.1.3 OS197L TC Accidental Cask Drop

This event is described in Sections 8.2.5.2.B, D and E.

See Section *W.3.7* for a discussion of the OS197L TC drop accident. This drop accident is bounded by the results for the OS197 TC drop accident discussed in Section 8.2.

W.11.1.4 Loss of Neutron Shield

This event is described in Section 8.2.5.3.

For the accident condition (the unlikely cask drop scenario) a complete loss of the OS197L TC neutron shield is postulated similar to the OS197 TC evaluation described in Section 8.2.5.3. *The* analysis conservatively assumes that all the trailer shielding is lost. However, the trailer shield is fabricated using two sets of plate shields (the inside shield is 2.5" thick, the outside shield is 3" thick) which may be damaged in a drop but are unlikely to separate completely from the skid and cask. *A comparison of the dose rate results for various OS197L shielding configurations including complete loss of trailer shielding (shown as "Bare Cask" configuration) is shown in the table below.*

Assuming the non-mechanistic drop scenario occurs and the trailer shields and the cask are dislodged completely from the trailer and skid, recovery actions are required to manipulate the shields or providing supplemental shielding to reduce dose rates to a reasonable value until a long term recovery plan is in place. *As discussed above, the dose rates for this accident case are calculated assuming a complete loss of trailer shielding.*

OS197L TC Accident Condition Dose Rates
(From results shown in Table W.5-4)

Transfer Cask Configuration	Dose Rate Component	Dose Rates at Different Distances from Side Surface – Accident Condition No Neutron Shield			
		On Side Surface	4.57 meters (15')	100 meters	609.9 meters (2000')
		Dose Rate, mrem/hr	Dose Rate, mrem/hr	Dose Rate, mrem/hr	Dose Rate, mrem/hr
OS197 L TC (with supplemental inner and outer trailer shielding)	Neutron	*727*	*79.0*	*0.32*	*2.31e-4*
	Gamma	134	*40.0*	*0.14*	*3.04e-4*
	Total	791	*104*	*0.46*	*5.34e-4*
OS197 L TC (with supplemental inner and without outer trailer shielding)	Neutron	1466	*87.0*	*0.48*	*7.36e-4*
	Gamma	540	*59.0*	*0.29*	*1.07e-3*
	Total	1543	*129*	*0.77*	*1.81e-3*
OS197L TC (Bare Cask)	Neutron	4,194	*175*	*0.20*	*4.97e-5*
	Gamma	15,305	*1,332*	*2.30*	*2.43e-3*
	Total	18,210	*1,438*	*2.48*	*2.46e-3*

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The dose rates on the ends of the OS197L TC will be the same as the OS197 TC since the top and bottom forging and cover plate configurations have not been modified. *Therefore,* the dose rates at the *controlled area* boundary, assuming a 100 meter boundary, would be approximately *2.48* mrem/hr during the timeframe that the cask trailer shield is dislodged from the cask and until the trailer shield is repositioned.

All changes on this page are AMD 11

The 8 hours of recovery period assumed is appropriate because the repositioning of the trailer shields will be performed using lifting hardware pre-positioned prior to transfer operations. This will facilitate quick positioning using a crane to minimize the need for personnel to approach the cask. *The dose rates at the controlled area boundary after repositioning of the inner and outer trailer shields are 0.77 and 0.46 mrem/hour respectively.*

The total dose due to an 8-hour exposure at the controlled area boundary (100 meters) is calculated to be slightly less than 20.0 mrem. Note that this is lower than the 42 mrem exposure reported in UFSAR Section M.11.2.5.3 for the 32PT DSC with the OS197 TC. In summary, the total dose at the 100-meter *controlled area* boundary still remains very low and below the regulatory limit of 5,000 mrem.

The total dose due to an 8-hour exposure to off-site individuals at the site boundary (2000 ft) is calculated to be 0.02 mrem. Note that this is lower than the 0.09 mrem exposure reported in UFSAR Section M.11.2.5.3 for the 32PT DSC and 0.13 mrem exposure reported in UFSAR Section K.11.2.5.3 for the 61BT DSC with the OS197 TC.

The thermal evaluation for this accident condition is included in Chapter W.4.

W.11.1.5 Accidental Drop of Top Trailer Shielding

Placement of the inner and outer shields on the skid inside the fuel/reactor building is to be performed in accordance with the plant's heavy loads procedures. If a single failure proof crane is not used, the licensee is to evaluate the accidental drop of the shields under the provisions of 10 CFR 50.59 and 10 CFR 72.212 and evaluate consequences of this drop accident.

In the case when fuel/reactor building floor loads limit placement of both the inner and outer trailer shields inside the fuel/reactor building, the outer top trailer shield may be placed outside the fuel/reactor building. This condition is evaluated for accidental drop of the outer top trailer shield onto the already mounted inner trailer shield.

The stresses in the inner shield are evaluated in accordance with Subsection NF stress criteria for accident (Level D) conditions. For Level D loads, the Subsection NF stress criteria for accident loads use the Appendix F stress limits. Based on a conservative elastic analysis model used in the stress evaluation and using conservation of energy principles, the maximum drop height which will meet the level D stress limits is on the order of 4 inches. Thus, the movement of the outer shield over the skid is to be controlled such that the maximum drop height does not exceed 4 inches.

W.12 Operating Controls and Limits

The Technical Specification changes, due to the addition of the OS197L TC System, are included in Attachment A to NUHOMS® CoC 1004.

All changes on this page are AMD 11