FINAL SAFETY ANALYSIS REPORT
FOR THE
STANDARDIZED NUHOMS®
HORIZONTAL MODULAR STORAGE SYSTEM
FOR IRRADIATED NUCLEAR FUEL

NON-PROPRIETARY

By
Transnuclear, Inc.
Hawthorne, NY

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EXECUTIVE SUMMARY

This Final Safety Analysis Report (No. NUH-003, Revision 7, NRC Docket No. 72-1004) provides the generic safety analysis for the standardized NUHOMS®[1] system for storage of light water reactor spent nuclear fuel assemblies. This system provides for the safe dry storage of spent fuel in a passive Independent Spent Fuel Storage Installation (ISFSI) which fully complies with the requirements of 10CFR72 and ANSI 57.9. The related NUHOMS®-24P Topical Report (No. NUH-002, Revision 1A, NRC Project No. M-49) was approved by the U.S. Nuclear Regulatory Commission on April 21, 1989. The original NUHOMS®-07P Topical Report (No. NUH-001, Revision 1A, NRC Project No. M-39) was approved by the U.S. Nuclear Regulatory Commission on March 28, 1986.

This Final Safety Analysis Report (FSAR) formed the basis for generic NRC certification of the standardized NUHOMS® system and will be used by 10CFR50/10CFR72 general license holders in accordance with 10CFR72 Subparts K and L. It is also suitable for reference in 10CFR72 site specific license applications. In January 1995, the USNRC issued a generic Certificate of Compliance to VECTRA for the standardized NUHOMS® canister/module horizontal cask storage system. The Nuclear Regulatory Commission staff does not intend to repeat the review in order to authorize the use of a standardized NUHOMS® ISFSI by a general license holder.

The principal features of the standardized NUHOMS® system which differ from the previously approved NUHOMS®-24P system are:

1. A free-standing prefabricated horizontal storage module founded on an ISFSI basemat which is not important to safety.

2. A standardized dry shielded canister for on-site dry storage and eventual off-site shipment of spent PWR or BWR fuel assemblies.

3. Removal of site specific dependencies to allow direct implementation by 10CFR72 general license holders.

4. Design qualification for five-year cooled PWR and BWR spent fuel.

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[1] NUHOMS® is a registered trademark of Transnuclear, Inc.

The NUHOMS® system provides long-term interim storage for spent fuel assemblies which have been out of the reactor for a sufficient period of time and which comply with the criteria set forth in this FSAR. The fuel assemblies are confined in a helium atmosphere by a canister containment pressure vessel. The canister is protected and shielded by a massive reinforced concrete module. Decay heat is removed from the canister and the concrete module by a passive natural draft convection ventilation system.

The canisterized spent fuel assemblies are transferred from the plant's spent fuel pool to the concrete storage modules located at the ISFSI in a transfer cask. The cask is aligned with the storage module and the canister is inserted into the module by means of a hydraulic ram. The NUHOMS® system is a totally passive installation that is designed to provide shielding and safe confinement of spent fuel for a range of postulated accident conditions and natural phenomena. As a condition of the USNRC Certificate of Compliance, temperature monitoring of the concrete module is required.

Revision 4A of this FSAR consists of a revision to the previously submitted report and incorporates the conditions of use specified by the Certificate and US NRC's Safety Evaluation Report that were not included in earlier revisions, along with revisions to reflect design modifications and utility comments.

Revision 5 of this FSAR incorporates all design modifications and supporting analysis implemented per Condition 9 of USNRC Certificate of Compliance (CoC) since issuance of Revision 4A. It also incorporates changes due to approval of Amendments 1 and 2 to the CoC.

Revision 6 of this FSAR incorporates all design modifications implemented per Condition 9 of CoC 1004 since issuance of FSAR Revision 5. It also incorporates changes implemented under CoC Amendment No. 3.

Revision 7 of this FSAR incorporates all design modifications implemented per 72.48 since the issuance of FSAR Revision 6. It also incorporates changes implemented due to approval of Amendment No. 4 to CoC 1004.

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LIST OF ABBREVIATIONS

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALARA	As Low as is Reasonably Achievable
ANF	Advanced Nuclear Fuels
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
10CFR	Code of Federal Regulations, Title 10
CE	Combustion Engineering
DBT	Design Basis Tornado
DSC	Dry Shielded Canister
GE	General Electric
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
NDE	Non-Destructive Examination
NDRC	National Defense Research Committee
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Modular Storage
NUREG	Nuclear Regulatory Guide
OBE	Operating Basis Earthquake
OSHA	Occupational Health and Safety Administration
PI	Project Instruction
PWR	Pressurized Water Reactor
QP	Quality Procedure
R.G.	NRC Regulatory Guide
FSAR	Final Safety Analysis Report
SFA	Spent Fuel Assembly
SSE	Safe Shutdown Earthquake
TC	Transfer Cask
TR	Topical Report
U.S.	United States
W	Westinghouse

LIST OF ABBREVIATIONS

(Continued)

atm	atmosphere
bar	bar
cm	centimeter
°C	degrees Centigrade
°F	degrees Fahrenheit
fps	feet per second
ft/s	feet per second
ft	foot
ft-lb	foot pounds
He	helium
in	inch
kg	kilogram
kW	kilowatt
k-in	kip inch
ksi	kips per square inch
MWD/MTU	megawatt days per metric ton uranium
MWe	megawatts electric
MWt	megawatts thermal
Hg	Mercury
m	meter
Ci/cm^2	Curies per square centimeter
MeV	Megaelectron volt
mph	miles per hour
mm	millimeter
mrem/hr	millirem per hour
mR/hr	milliroentgen per hour
n	neutron
k_{eff}	neutron multiplication factor, effective
kN	kilonewton
N	Newton
lb	pound
lbf	pounds-force
psf	pounds per square foot
psi	pounds per square inch
psia	pounds per square inch, absolute
psig	pounds per square inch, gauge
sec	second
sq. mi.	square mile
kips	thousand pounds
ton	ton
w/o	without
wt. %	weight %

1. INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

This Final Safety Analysis Report (the terms, FSAR or SAR, are used interchangeably in this document) describes the design and forms the generic licensing basis for 10CFR72 Subpart L (1.1) certification of the standardized NUHOMS® horizontal cask system for dry storage of PWR or BWR spent nuclear fuel assemblies. The NUHOMS® system provides for the horizontal storage of spent fuel in a dry shielded canister (DSC) which is placed in a concrete horizontal storage module (HSM). The NUHOMS® system is designed to be installed at any reactor site or any new site where an independent spent fuel storage installation (ISFSI) is required.

The original NUHOMS® Topical Report (NUH-001, Revision 1A, NRC Project No. M-39) was approved by the United States Nuclear Regulatory Commission (NRC) on March 28, 1986 for storage of seven spent PWR fuel assemblies per DSC and HSM (NUHOMS®-07P) (1.12, 1.13). The NUHOMS®-07P system is designed to be compatible with the IF-300 shipping cask. The DSC internal basket incorporates borated guide sleeves to ensure criticality safety during wet loading operations without credit for burnup or soluble boron.

The NUHOMS® Topical Report was revised (NUH-002, Revision 0, NRC Docket No. M-49) to provide the generic design criteria and safety analysis for the larger 24 spent PWR fuel assembly design (NUHOMS®-24P) and its associated on-site transfer cask. NRC approval of the NUHOMS®-24P Topical Report was granted on April 26 1989 (1.10, 1.11). Unlike the NUHOMS®-07P design, no borated neutron absorbing material is used in the internal basket design of the NUHOMS®-24P DSC for criticality safety. Credit for soluble boron is used as the approval basis. Credit for burnup is also evaluated as an alternative design acceptance basis for the NUHOMS®-24P DSC design pending future generic acceptance by the NRC. The approved NUHOMS®-24P Topical Report forms the principal basis for the standardized NUHOMS® system presented in this FSAR. The NRC has issued Certificate of Compliance (CoC) 1004, dated January 23, 1995, for the standardized NUHOMS® system.

This FSAR also includes the NUHOMS®-52B DSC, which is designed to store 52 BWR fuel assemblies with the fuel assembly flow channels intact. The NUHOMS®-52B utilizes the same HSM as does the standardized NUHOMS®-24P DSC. New criticality, thermal and structural analyses for the 52B basket are included as are the specifications of spent fuel assemblies to be stored. The 52B basket includes fixed neutron absorbing plates for criticality safety, similar to that of the NUHOMS®-07P DSC. Unborated plates may be used pending a burnup credit analysis to be submitted when burnup credit is generically accepted by the NRC.

The NRC approved Amendment No. 1 to CoC 1004 on April 2000. This amendment reflects the transfer of the CoC from VECTRA Technologies, Inc. to Transnuclear West Inc.

Amendment No. 2 to CoC 1004, approved on September 5, 2000, adds fuel qualification tables and updates Fuel Specification 1.2.1 to reflect additional fuel parameters for both the PWR and BWR fuels. The fuel qualification tables provide a simplified approach for users of the

NUHOMS® storage system in selection of acceptable assemblies during loading. In addition, Amendment No. 2 authorizes the storage of Burnable Poison Rod Assemblies (BPRAs) in the NUHOMS®-24P long cavity DSC. A detailed description of the authorized contents and supporting analyses for the storage of PWR fuel with BPRAs is provided in Appendix J.

Amendment No. 3 to CoC 1004, approved on September 12, 2001, authorizes the addition of the NUHOMS®-61BT DSC to the standardized NUHOMS® system. The NUHOMS®-61BT DSC is designed to store 61 intact BWR fuel assemblies and meets the storage and transportation requirements of 10CFR72 and 10CFR71, respectively. A detailed description of the authorized contents and supporting safety analyses for this system are provided in Appendix K.

TN has added NUHOMS®-24PT2 DSC to the standardized NUHOMS® system. The NUHOMS®-24PT2 DSC is a modified version of the NUHOMS®-24P DSC, designed to store 24 intact PWR fuel assemblies with or without BPRAs. This DSC meets the storage and transportation requirements of 10CFR72 (CoC 1004) and 10CFR71 (CoC 9255), respectively. A detailed description of the authorized contents and supporting safety analyses for this system are provided in Appendix L.

Revision 6 adds enhanced versions of the standardized HSM and transfer cask, designated as HSM Model 102 and OS197H, respectively, to the standardized NUHOMS® system.

Appendix B has been revised to include a validation of the fuel effective conductivity values used in the standardized NUHOMS® thermal analysis against the NUHOMS®-7P test data.

Amendment No. 4 to CoC 1004, approved on February 12, 2002, authorizes the addition of low burn-up spent fuel in the NUHOMS®-24P DSC.

1.1 Introduction

Due to the unavailability of nuclear fuel reprocessing or a permanent geologic repository in the United States (U.S.), long-term storage of spent fuel assemblies (SFAs) has become necessary. To date, storage systems have, to a large extent, relied on the plant's spent fuel pools. However, as existing pools have begun to approach their capacity (with high-density storage racks), out-of-pool dry storage system designs have emerged. NUHOMS® is a proven system for dry storage which has been in use at reactor sites since March of 1989.

Figure 1.1-1, Figure 1.1-2 and Figure 1.1-3 show the primary components and arrangement of an ISFSI utilizing the NUHOMS® system. The SFAs are loaded into the DSC (which is placed inside the transfer cask) in the fuel pool at the reactor site. The transfer cask containing the loaded DSC is removed from the pool and placed in the cask decon area where sealing, draining, and drying operations are performed. The DSC cavity is then backfilled with helium. Multi-layer, double seal welds at each end of the DSC and multi-layer circumferential and longitudinal welds are utilized to assure that no leakage of helium can occur. The cask is then placed on a transport trailer in the plant's fuel/reactor building and towed to the ISFSI located on-site. At the ISFSI location, the loaded transfer cask is aligned with the HSM and the DSC is pushed out of the transfer cask into the HSM using a hydraulic ram. Once inside the HSM, the DSC is in safe, passive dry storage.

The various components of the NUHOMS® system are further described in Sections 1.2 and 1.3. The design and the conservative generic analyses of the system components are described in detail in the remainder of this FSAR. The principal design features of a NUHOMS® ISFSI are:

1. Canisterized Spent Fuel in a Welded Containment Vessel Shielded by a Prefabricated Concrete Module

   This provides for a high integrity multiple barrier system to ensure the safe storage of spent fuel which can be easily implemented by a licensee on a timely economical basis.

2. Horizontal Transfer of the DSC into and out of the HSM

   This obviates the need for a critical heavy lift of the SFAs at the storage location (i.e. away form the plant's safety-related systems), optimizes the amount of material required for biological shielding, and results in a passive, low profile, impact-resistant storage structure. This also provides a means for canister retrieval and eventual off-site shipment in a compatible licensed shipping cask without future reliance on plant facilities.

3. Transport of the DSC from the fuel/reactor building to the HSM in a Shielded Atmospheric On-site TC

   This provides radiation shielding and structural protection for the DSC during the transfer operation while providing passive heat removal for the canisterized spent fuel.

4. Shielded End Plug Assemblies on the DSC

   This enables contact handling and monitoring of the DSCs at the top and bottom end locations when the DSC is inside the transfer cask or the HSM.

5. Phased Construction of HSMs

   This facilitates ISFSI licensing and phased construction of the HSM arrays, thus economizing and distributing the cost for fuel storage over the time span when storage is actually required.

6. Passive Natural Circulation Air Cooling

   This keeps the maximum fuel rod cladding temperature below acceptable limits to preclude damage during long term storage.

7. Acceptance of Equivalent Spent Fuel

   The NUHOMS® system accepts PWR or BWR spent fuel assemblies with equivalent decay heat and radiological source term values enveloped by those corresponding to fuel with the cooling time, initial uranium content, initial enrichment, and fuel burnup as outlined in Chapter 3.

8. Helium Storage Atmosphere

   This provides effective heat transfer and prevents oxidation of the fuel cladding and fuel pellets. The double seal welded DSC containment boundary assures that the helium atmosphere is maintained.

This FSAR is written for the U.S. NRC for review under Title 10 of the Code of Federal Regulations, Part 72 (10CFR72). In addition, this document provided the technical basis for issuance of a generic CoC 1004 to Transnuclear, Inc. (TN) for the standardized NUHOMS® canister/module storage cask system to allow use by a 10CFR72 general license holder in accordance with 10CFR72 Subpart L (1.1).

To facilitate direct referencing, the format, the numbering system, the section headings, and the content have followed NRC Regulatory Guide 3.48 (1.2). Numbers in parentheses indicate references which are listed at the end of each chapter. SI units are used in the first three chapters of this SAR. Where the general design features of the system are discussed, commonly used units are included in parentheses. For Chapters 4 through 11, the units commonly used in the U.S. for the various design and analysis work are used. SI units are provided in parentheses where meaningful.

Table 1.2-4
Typical Plant Equipment and Materials Used for NUHOMS® DSC Loading, Closure, and Transfer Operations
(for information only)

1. Fuel Pool Lighting
2. Under Water Viewing Box
3. Fuel Handling Equipment
4. Cask Handling Crane (≥ 100 ton capacity)
5. Slings and Lifting Devices
6. Consumables and Tools
   - Demineralized Water for Decon
   - Waterproof Tape
   - Bottled Helium
   - Low Voltage Electricity
   - Torque Wrench
   - Compressed Air
   - Temperature Probe
   - Shielding Blankets
   - Welding Materials
7. Plant Radwaste Handling System
8. Radiation Monitoring Equipment
9. Helium Leak Detector
10. Heavy Haul Towing Vehicle
11. Mobile Telescoping Crane
12. Optical Survey Equipment
13. Portable Welding Equipment

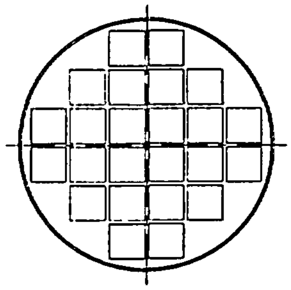
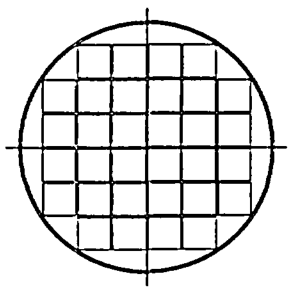
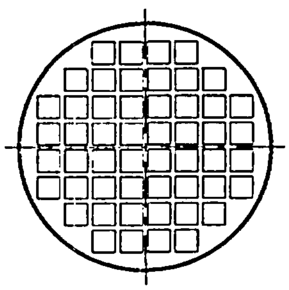
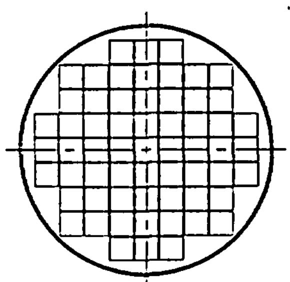
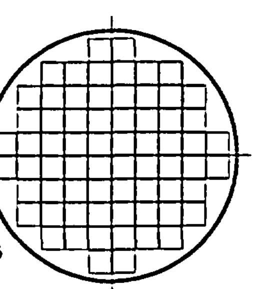
CONFIGURATION	DESCRIPTION
	NUHOMS®-24P BASKET FOR PWR FUEL WITH NO BORATED NEUTRON ABSORBING MATERIAL AND CREDIT FOR SOLUBLE BORON
	NUHOMS®-32PT (FUTURE) BASKET FOR PWR FUEL WITH BORATED NEUTRON ABSORBING MATERIAL OR CREDIT FOR BURNUP
	NUHOMS®-52B BASKET FOR CHANNELLED BWR FUEL WITH BORATED NEUTRON ABSORBING MATERIAL AND NO CREDIT FOR BURNUP
	NUHOMS®-61BT BASKET FOR BWR FUEL (WITH OR WITHOUT CHANNELS) WITH BORATED NEUTRON ABSORBING MATERIAL AND NO CREDIT FOR BURNUP
FD115 	NUHOMS®-68B (FUTURE) BASKET FOR UNCHANNELLED BWR FUEL WITH BORATED NEUTRON ABSORBING MATERIAL OR CREDIT FOR BURNUP

Figure 1.2-1
Standardized NUHOMS® Systems Canister Baskets for PWR and BWR Spent Fuel
(for information only

controlled access. The necessary civil work required to prepare the ISFSI site is the same as that for an ISFSI utilizing vertical storage casks.

Two alternate designs of the standardized HSM are available for licensees' use: the original HSM, now designated as HSM Model 80, and HSM Model 102. HSM Model 102 design is similar to HSM Model 80 design except for the following two features:

- The steel encased composite door of HSM Model 80 design is replaced by a two foot thick reinforced concrete door with a steel liner on its inside surface. The steel liner mitigates DSC damage from spalled concrete due to tornado generated missile impact.

- The inlet and outlet vents, which are formed in concrete for HSM Model 80, are lined with 1½" steel plates.

The above features included with HSM Model 102 are improvements to the original HSM Model 80 design that increase the shielding capabilities of the HSM. The heat transfer capability (decay heat rejection from the DSC to the HSM and heat removal from the HSM by natural convection) of both HSM Model 80 and HSM Model 102 designs are equivalent. Appendix E drawings show both models. Each model can store a DSC with maximum weight up to 102 kips which includes 24P, 52B, 24PT2 and 61BT DSCs.

1.3.2 Transfer Systems Descriptions

1.3.2.1 On-Site TC

The transfer cask used in the NUHOMS® system provides shielding and protection from potential hazards during the DSC closure operations and transfer to the HSM. Two alternate configurations of the transfer cask are available for the licensees' use. The basic configuration, where the cask is provided with a solid neutron shield, is described herein as the "Standardized Cask." An alternate configuration, where a liquid neutron shield is provided instead, is described in this SAR as the "OS197 and OS197H Casks." The configuration of the OS197 is a slightly modified version of the NRC approved cask (with a liquid neutron shield) as described in the NUHOMS®–24P Topical Report (1.10). The standardized transfer cask documented in this SAR has a gross weight of less than 90.7 Te (100 tons) and is limited to on-site use under 10CFR72. The OS197 and OS197H transfer casks, which are also limited to on-site use under 10CFR72, have a maximum gross weight of 94.6 Te (104.25 tons) and 113.4 Te (125 Tons), respectively. In addition, the licensee may also elect to utilize a future transfer cask having a gross weight of about 113.4 Te (125 tons) which can be used on-site under 10CFR72, but is also suitable for future off-site shipment of intact NUHOMS® canisters under 10CFR71. Where applicable, any other NRC licensed NUHOMS® transfer or transportation cask is acceptable for use with the standardized NUHOMS® system subject to an application specific safety evaluation.

The standardized transfer cask for the NUHOMS® system, has a 4.75m (186.75 inches) long inner cavity, a 1.73m (68 inches) inside diameter and a maximum payload capacity of 40,900 kg (90,000 pounds) wet and 36,300 kg (80,000 pounds) dry. A cask collar is used to extend the transfer cask cavity length by .25m (10 inches) for use with the longer DSCs for BWR fuel. The OS197 and OS197H transfer casks with a longer cavity length of 196.75 inches (no cask collar) may be used for DSCs with BWR fuel and when combined with a cask spacer may also be used to load DSCs with PWR fuel. The transfer cask is designed to meet the requirements of 10CFR72 for on-site transfer of the DSC from the plant's fuel pool to the HSM. As shown in Figure 1.3-6, the transfer cask is constructed from two concentric cylindrical steel shells with a bolted top cover plate and a welded bottom end assembly. The annulus formed by these two shells is filled with cast lead to provide gamma shielding. The transfer cask also includes an outer steel jacket which is filled with a hydrogen rich solid material or water for neutron shielding. The top and bottom end assemblies also incorporate a solid neutron shield material.

The transfer cask is designed to provide sufficient shielding to ensure that dose rates are ALARA. Two lifting trunnions are provided for handling the transfer cask in the plant's fuel/reactor building using a lifting yoke and an overhead crane. Lower support trunnions are provided on the cask for pivoting the transfer cask from/to the vertical and horizontal positions on the support skid/transport trailer. A cover plate is provided to seal the bottom hydraulic ram access penetration of the cask during fuel loading.

1.3.2.2 Transfer Equipment

Transport Trailer: The NUHOMS® transport trailer consists of a heavy industrial trailer with a payload capacity of 113.4 Te (125 tons). The trailer transports the cask support skid and the loaded transfer cask between the plant's fuel/reactor building and the ISFSI. The trailer is designed to ride as low to the ground as possible to minimize the HSM height and the transfer cask height during transport and DSC transfer operations. Figure 1.3-7 shows the heavy haul industrial trailer used with the standardized NUHOMS® system. The trailer is equipped with four hydraulic leveling jacks to provide vertical travel for alignment of the cask with the HSM. The trailer is towed by a conventional heavy haul truck tractor or other suitable prime mover. The nominal trailer bed height during canister transfer to the HSM is such that the transfer cask is typically not elevated more than 1.68m (5'-6") above grade as measured from the lowest point on the cask. This is well below the 2.0m (80 inch) drop height used as the accident drop design basis of the cask and canister.

Cask Support Skid: The NUHOMS® system cask support skid is similar in design and operation to other transportation cask skids used for shipment of fuel. The key differences are:

1. There is no ancillary equipment mounted on the skid.

2. The skid is mounted on a surface with sliding support bearings and hydraulic positioners to provide alignment of the cask with the HSM. Brackets with locking bolts are provided to prevent movement during trailer towing.

3. The hydraulic ram is mounted on the skid or, as an option, the ram can be set-up using a frame structure bolted to the cask bottom and a rear support tripod.

Information withheld under 10 CFR 2.390(d)

1.4 Identification of Agents and Contractors

The prime contractor for design and procurement of the NUHOMS® system components is Transnuclear, Inc. (TN). TN will subcontract the fabrication and on-site construction to qualified firms on a project specific basis.

The generic design activities for the NUHOMS® Safety Analysis Report (NUH-003) were performed by TN with Duke Power Company, Inc. as a subcontractor for the PWR criticality analysis. TN is responsible for the design and analysis of the DSC, the HSM, the on-site transfer cask, and the associated transfer equipment.

Table 3.1-1
Principal Acceptance Parameters for PWR Fuel to be Stored in NUHOMS® -24P and -24PT2 DSC

Title or Parameter	Specifications
Fuel	Only intact, unconsolidated PWR fuel assemblies (with or without BPRAs) with the following requirements.
Physical Parameters (without BPRAs)	
Maximum Nominal Assembly Length (unirradiated) • With Burnup ≤ 45,000 MWd/MTU	165.75 in (standard cavity) 171.71 in (long cavity)
Nominal Assembly Width (unirradiated)	8.536 in
Maximum Assembly Weight	1682 lbs
No. of Assemblies per DSC	≤ 24 intact assemblies
Fuel Cladding	Zircalloy-clad fuel with no known or suspected gross cladding breaches
Physical Parameters (with BPRAs)	
Maximum Assembly + BPRA Length (unirradiated) • With Burnup >32,000 and ≤ 45,000 MWd/MTU • With Burnup ≤ 32,000 MWd/MTU	171.71 in (long cavity) 171.96 in (long cavity)
Nominal Assembly Width (unirradiated)	8.536 in
Maximum Assembly + BPRA Weight	1682 lbs
No. of Assemblies per DSC	≤ 24 intact assemblies
No. of BPRAs per DSC	≤ 24 BPRAs
Fuel Cladding	Zircalloy-clad fuel with no known or suspected gross cladding breaches
Nuclear Parameters	
Fuel Initial Enrichment	≤ 4.0 wt. % U-235
Fuel Burnup and Cooling Time	Per Table 3.1-8a (without BPRAs) or Per Table 3.1-8c (with BPRAs)
BPRA Cooling Time (Minimum)	5 years for B&W Designs 10 years for Westinghouse Designs
Alternate Nuclear Parameters	
Initial Enrichment	≤ 4.0 wt. % U-235
Burnup	≤ 40,000 MWd/MTU and per Figure 3.3-3
Decay Heat (Fuel + BPRA)	≤ 1.0 kW per assembly
Neutron Fuel Source	≤ 2.23 x 10^8 n/sec per assy with spectrum bounded by that in Chapter 7 of FSAR
Gamma (Fuel + BPRA) Source	≤ 7.45 x 10^{15} g/sec per assy with spectrum bounded by that in Chapter 7 of FSAR

Table 3.1-1a
PWR Fuel Assembly Designs Suitable for Storage

Type(1)	Width (in)	Assembly Unirradiated Length (w/o* BPRAs) (in)	Assembly Unirradiated Length (with BPRAs) (in)	Assembly Weight (w/o* BPRAs) (lbs)	Assembly Weight (with BPRAs) (lbs)	Heavy Metal Weight (kg-U)	Cladding Material
B&W 15x15(8)	8.536	165.75	170.875	1550.0	1682.0	475.0	Zircaloy-4
CE 14x14 Fort Calhoun(2)	8.100	147.00	*n/a*	1220.0	*n/a*	365.6	Zircaloy-4
CE 15x15 Palisades(3)	8.250	149.00	*n/a*	1360.0	*n/a*	412.4	Zircaloy-4
CE 14x14 Standard/Generic	8.100	157.00	*n/a*	1270.0	*n/a*	382.2	Zircaloy-4
Westinghouse 14x14(5)	7.763	160.13	*n/a*	1302.0	*n/a*	405.0	Zircaloy-4
Westinghouse 15x15(6)	8.434	160.10	*n/a*	1472.0	*n/a*	460.0	Zircaloy-4
Westinghouse 17x17(7)	8.434	160.10	167.220	1482.0	1663.2	461.0	Zircaloy-4
Limit:	8.536	165.75	171.710/171.96(9)	1682.0	1682.0	475.0	

(1) Each fuel assembly must be qualified for storage per 72-1004 CoC Technical Specifications.
(2) Includes Exxon/ANF FT. CALHOUN 14 X 14 ANF
(3) Includes Exxon/ANF 15x15 CE
(4) Not used
(5) Includes Exxon/ANF 14x14 Westinghouse
(6) Includes Exxon/ANF 15x15 Westinghouse
(7) Includes Babcock and Wilcox WE 17 X 17 B&W Mark BW
(8) Excludes Westinghouse 15x15 reload fuel for B&W 15x15 reactors
(9) Maximum allowed burnup is 32,000 MWd/MTU for the 171.96 long assemblies (plus BPRAs)

* w/o means without

Table 3.1-2
Principal Acceptance Parameters for BWR Fuel to be Stored in NUHOMS® -52B DSC

Title or Parameter	Specifications
Fuel	Only intact, unconsolidated BWR fuel assemblies with the following requirements
Physical Parameters	
Maximum Assembly Length (unirradiated)	176.16 in
Nominal Assembly Width* (unirradiated)	5.454 in
Maximum Assembly Weight	725 lbs
No. of Assemblies per DSC	≤ 52 intact channeled assemblies
Fuel Cladding	Zircaloy-clad fuel with no known or suspected gross cladding breaches
Nuclear Parameters	
Fuel Initial Lattice Enrichment	≤ 4.0 wt. % U-235
Fuel Burnup and Cooling Time	Per Table 3.1-8b
Alternate Nuclear Parameters	
Initial Enrichment	≤ 4.0 wt. % U-235
Burnup	≤ 35,000 MWd/MTU and per Figure 3.3-3
Decay Heat	≤ 0.37 kW per assembly
Neutron Source	≤ 1.01 x 10^8 n/sec per assy with spectrum bounded by that in Chapter 7 of FSAR
Gamma Source	≤ 2.63 x 10^{15} g/sec per assy with spectrum bounded by that in Chapter 7 of FSAR

* Outside dimension of the fuel channel

Table 3.1-2a

BWR Fuel Assembly Designs Suitable for Storage in NUHOMS®-52B DSC

Type(1)	Channeled Width (in)	Unirradiated Length (in)	Assembly Weight (lbs)	Heavy Metal Weight (kg-U)	Cladding Material
GE 6x6 Dresden-1(2)	4.850	136.00	400	111.4	Zircaloy-2
GE 7x7(3)	5.438	175.87	690	194.9	Zircaloy-2
GE 8x8(4)	5.440	176.05	690	186.7	Zircaloy-2
Limit:	5.454	176.16	725	198.0	

(1) Each fuel assembly must be qualified for storage per Technical Specifications of CoC 1004.

(2) Includes Exxon/ANF DRESDEN-1 6x6 ANF.

(3) Includes Exxon/ANF GE BWR 7x7 ANF.

(4) Includes Exxon/ANF GE BWR 8x8 ANF.

across the tornado is 20.7 kN/m^2 (3.0 psi), and the rate of pressure drop is 13.8 kN/m^2 (2.0 psi) per second. The maximum transit time based on the 2.2 m/sec (5 miles per hour) minimum translational speed specified for Region I is not used since an infinite transit time is conservatively assumed.

3.2.1.2 Determination of Forces on Structures

The effects of a DBT are evaluated for the NUHOMS® transfer cask and HSM. Tornado loads are generated for three separate loading phenomena: First, pressure or suction forces created by drag as air impinges and flows past the transfer cask or HSM; second, suction due to tornado generated pressure drop of 3 psi; and third, impact, forces created by tornado-generated missiles impinging on the HSM and TC. The atmospheric pressure change induced forces are considered. In the following paragraphs, the determination of these forces is described.

The determination of the DBT velocity pressure is based on the following equation as specified in ANSI A58.1-1982 (3.9).

$$q = 0.00256\, K_z (IV)^2 \qquad (3.2\text{-}1)$$

Table 5 of ANSI A58.1 (3.9) defines the Importance Factor (I) to be 1.07 and the velocity pressure exposure coefficient (K_z) to be 0.8 applied to the full HSM height of 4.6 m (15 feet). Since the generic design basis HSM dimensions are relatively small compared to the 45.7 m (150 ft) rotational radius of the DBT, the velocity value of combined rotational and translational wind velocity of 160 m/sec (360 miles per hour) is conservatively used in Equation 3.2-1 as follows:

$$q = 0.00256 \times 0.8 \times [1.07 \times 360]^2 = 304 \text{ psf} \qquad (3.2\text{-}2)$$

The calculated DBT velocity pressure is converted to a design wind pressure by multiplying this value by the appropriate pressure and gust response coefficients specified in Figure 2 and Table 8.4 of ANSI A58.1-1982. Note that with a gust response coefficient of 1.32 as used in Table 3.2-2, the wind pressure used in this analysis bounds that obtained using the basic wind pressure formula $q = 0.00256 \times V^2$. The magnitude and direction of the design pressures for various HSM surfaces and the corresponding pressure coefficients are tabulated in Table 3.2-2. The effects of overturning and sliding of the HSM under these design pressures are also evaluated and are reported with the stress analysis results in Section 8.2.

The transfer cask is also evaluated for a 19 kN/m^2 (397 psf) DBT velocity pressure since this load magnitude envelops that for a closed cylindrical structure such as the cask. The transfer cask stress analysis for tornado wind loads is contained in Section 8.2.

The determination of impact forces created by DBT generated missiles for the HSM is based on the criteria provided by NUREG-0800, Section 3.5.1.4, III.4 (3.8). Accordingly, four types of missiles are postulated:

- For the massive high kinetic energy deformable missile specified in NUREG-0800, a 1800 kg (3,967 pound) automobile with a 1.86 m^2 (20 square foot) frontal area impacting at normal incidence with a velocity of 195 fps is assumed.
- For the rigid penetration-resistant missile specified, a 125 kg (276 pound), 0.2m (eight inch) diameter blunt-nosed armor piercing artillery shell, impacting at normal incidence with a velocity of 185 fps is assumed.
- For the protective barrier impingement missile specified, a 25.4 mm (one inch) diameter solid steel sphere is assumed.
- Wood plank missile, 101.6 mm (four inch) x 304.8 mm (twelve inch) x 3.66 m (twelve foot), weighing 200 lbs. traveling end or with a velocity 440 fps.

For the overall effects of a DBT missile impact, overturning, and sliding on the HSM, the force due to the governing deformable missile is applied to the structure at the most adverse location. Conservation of momentum is assumed to demonstrate that sliding and/or tipping of a single stand alone module will not result in an unacceptable condition for the module. The coefficient of restitution is assumed to be zero and the missile energy is transferred to the module to be dissipated as sliding friction, or an increase in potential energy due to raising the center of gravity. The force is evenly distributed over the impact area. The magnitude of the impact force for design of the local reinforcing is calculated in accordance with Bechtel Topical Report "Design of Structures for Missile Impact" (3.50).

For the local damage analysis of the HSM for DBT missiles, the governing rigid penetration-resistant missile is used for the evaluation of concrete penetration, spalling, scabbing and perforation thickness. The modified National Defense Research Committee (NDRC) empirical formula is used for this evaluation as recommended in NUREG-0800, Section 3.5.3 (3.25). The results of these evaluations are reported in Section 8.2.

The evaluation of tornado-generated missile loads on the transfer cask is addressed in Appendix C.5.

3.2.1.3 Ability of Structures to Perform

The HSM protects the DSC from adverse environmental effects and is the principal NUHOMS® structure exposed to tornado wind and missile loads. Furthermore, all components of the HSM (regardless of their safety classification) are designed to withstand tornadoes and tornado-based missiles. The transfer cask protects the DSC during transit to the ISFSI from adverse environmental effects such as tornado winds. The analyses of the HSM and transfer cask for tornado effects is contained in Sections 8.2.1 and 8.2.2.

Table 3.2-1
Summary of NUHOMS® Component Design Loadings

Component	Design Load Type	SAR Section Reference	Design Parameters	Applicable Codes
Horizontal Storage Module[1]:	—	—	—	ACI 349-85, ACI 349R-85 (design); ACI 318-83 (construction only)
	Design Basis Wind Load	3.2.1	Max. wind pressure : 397 psf Max. speed: 360 mph	NRC Reg. Guide 1.76 and ANSI A58.1 1982
	Design basis tornado wind load + Missile load	3.2.1	Maximum wind speed of 360 mph, and a pressure drop of 3 psi + Missile types: Automobile, 4000 lbs, 195 fps; 8" diameter shell, 276 lbs, 185fps; 1 in. diameter, solid steel sphere; wood plank, 4 in x 12 in x 12 ft, 200 lbs, 440 fps.	NRC Reg. Guide 1.76 and ANSI A58.1, 1982. NUREG-0800, Section 3.5.1.4
	Flood	3.2.2	Maximum water height: 50 feet Maximum velocity: 15 ft./sec.	10CFR72.122(b)
	Seismic	3.2.3	Hor. ground acceleration: 0.25g (both directions) Vert. ground acceleration: 0.17g with Reg. Guide 1.60 spectra at 7% damping.	NRC Reg. Guides 1.60 & 1.61
	Snow and Ice	3.2.4	Maximum load: 110 psf (included in live load)	ANSI A58.1-1982
	Dead Load	8.1.1.5	Dead weight including loaded DSC (concrete density of 150 pcf)	ANSI 57.9-1984
	Normal and Off-Normal Operating Temperatures	8.1.1.5	DSC with spent fuel rejecting 24.0 kW of decay heat for 5 yr. cooling time. Ambient air temperature range of -40°F to 125°F for off-normal case	ANSI 57.9-1984

(1) See Appendix K for information associated with the NUHOMS®-61BT DSC.

Table 3.2-1
Summary of NUHOMS® Component Design Loadings
(continued)

Component	Design Load Type	SAR Section Reference	Design Parameters	Applicable Codes
	Accident Condition Temperatures	8.2.7.2	Same as off-normal conditions with HSM vents blocked for 40 hours	ANSI 57.9-1984
	Normal Handling Loads	8.1.1.1	For concrete component evaluation 80,000 lb.(DSC HSM insertion) 60,000 lb (DSC HSM extraction)	ANSI 57.9-1984
	Off-normal Handling Loads	8.1.1.4	For concrete component evaluation 80,000 lb (DSC HSM insertion) 80,000 lb (DSC HSM extraction)	ANSI 57.9-1984
	Live Load	8.1.1.5	Design load: 200 psf (includes snow and ice loads)	ANSI 57.9-1984
	Fire and Explosions	3.3.6	Enveloped by other design basis events	10CFR72.122(c)
Dry Shielded Canister[1]:	---	---	---	ASME Code, Section III, Subsection NB, Class 1 Component
	Flood	3.2.2	Maximum water height: 50 ft.	10CFR72.122(b)
	Seismic	3.2.2	Horizontal ground acc.: 0.25g Vertical ground acc.: 0.17g	NRC Reg. Guides 1.60 & 1.61
	Dead Load	8.1.1.2	Weight of loaded 24P & 52B DSC: 80,000 lb. enveloping. Weight of loaded 24PT2 DSC: 85,000 lb. enveloping.	ANSI 57.9-1984
	Normal and Off-Normal Pressure	8.1.1.2	Enveloping internal pressure of ≤10.0 psig	10CFR72.122(h)
	Test Pressure	8.1.1.2	Enveloping internal pressure of 12 psig applied w/o DSC outer top cover plate	10CFR72.122(h)

(1) See Appendix K for information associated with the NUHOMS®-61BT DSC.

3.3.2.2 Ventilation - Offgas

The NUHOMS® system relies on natural convection through the air space in the HSM to cool the DSC. This passive convective ventilation system is driven by the pressure difference due to the stack effect (ΔP_s) provided by the height difference between the bottom of the DSC and the HSM air outlet, which is larger than the flow pressure drop (ΔP_f) at the design air inlet and outlet temperatures. The details of the ventilation system design are provided in Chapters 4 and 8.

There are no radioactive releases of effluents during normal and off-normal storage operations. Also, there are no credible accidents which cause significant releases of radioactive effluents from the DSC. Therefore, there are no off-gas or monitoring system requirements for the HSM. The only time an off-gas system is required is during DSC drying operations. During this operation, the spent fuel pool or plant's radwaste system is used to process the air and helium which is evacuated from the DSC.

3.3.3 Protection by Equipment and Instrumentation Selection

3.3.3.1 Equipment

The HSM, DSC, and on-site transfer cask are the equipment which is important to safety. Other equipment important to safety associated with the NUHOMS® system is the equipment required for handling operations within the plant's fuel/reactor building and are regulated by the plant's 10CFR50 operating license.

3.3.3.2 Instrumentation

The NUHOMS® system is a totally passive system. No safety-related instrumentation is necessary. The maximum temperatures and pressures are conservatively bounded by analyses (see Section 8.1.3). Therefore, there is no need for monitoring the internal cavity of the DSC for pressure or temperature during normal operations. The DSC is conservatively designed to perform its containment function during all worst case normal, off-normal, and postulated accident conditions. HSM thermal monitoring is required to meet the requirements of Chapter 10.

3.3.4 Nuclear Criticality Safety

The NUHOMS®-24P DSC is designed to meet 10CFR72.124 criticality safety limits during worst case wet loading/unloading operations without the use of fixed neutron absorbing materials (poisons) by two alternative means:

A. Utilizing credit for burnup (uranium depletion and non-volatile fission product buildup) and

B. Utilizing credit for the negative reactivity of soluble boron in the flooded DSC.

Since credit for burnup has not yet received generic approval by the NRC for dry storage applications, credit for soluble boron forms the design basis for the NUHOMS®-24P DSC until credit for burnup is generically accepted.

Several changes have been made to the generic NUHOMS®-24P design configuration analyzed in section 3.3.4.1. These changes include:

- The spacer disc material has been changed from SA-240, Type 304 stainless steel to SA-36 carbon steel. The thickness and location of the spacer discs have not changed.
- The support rod material has been changed from SA-479, Type 304 stainless steel or carbon steel to from SA-479, Type XM-19 stainless steel. The diameter of the support rods has been increased to 3.25 inches.
- The DSC guidesleeve configuration has been changed from twelve interior 12 gage sleeves and twelve 16 gage exterior sleeves to all 12 gage sleeves.
- The four stainless steel clips which connect each guidesleeve to the bottom spacer disc have been removed. To prevent removal of the guidesleeves from the basket if a fuel assembly is becomes stuck during insertion or removal, two stainless steel stops have been added to each guidesleeve between the second and third spacer discs from the top of the basket.
- The DSC guide sleeve length has been increased by 0.5 inch and 0.5 inch high flow channels have been provided at the bottom of guidesleeves on all four sides to allow drainage of any water inside the guidesleeve.

These changes in the 24P DSC basket configuration have been evaluated and the criticality analyses presented in section 3.3.4.1 remains bounding.

The NUHOMS®-52B DSC is designed to meet 10CFR72.124 criticality safety limits utilizing fixed neutron absorbing materials in the internal basket assembly until credit for burnup is generically accepted. The criticality safety analyses for the NUHOMS®-24P and NUHOMS®-52B DSCs are presented below.

3.3.4.1 NUHOMS®-24P DSC Criticality Safety (Fuel with an Equivalent Unirradiated Enrichment of less than 1.45 wt. % U-235)

The NUHOMS®-24P DSC credit for burnup criticality analysis is presented in the paragraphs which follow. The NUHOMS®-24P DSC credit for soluble boron analysis is documented by previous NRC question responses docketed under NUHOMS®-24P

Topical Report[3.51]. These question responses are included as Appendix F of this FSAR for ease of reference.

Appendix F documents that a soluble boron loading of 1810 ppm results in $k_{eff} \leq 0.98$ for zero burnup case. It also concludes that a burnup of 5 GWD/MTU is needed with 1810 ppm soluble boron if $k_{eff} \leq 0.95$ is required.

Section 3.1.2 of the Safety Evaluation Report [3.72] documents the current licensing basis for the specification of 2000 ppm soluble boron for the NUHOMS®-24P. As noted therein "The criticality safety analysis of the misloading of unirradiated fuel assemblies presented in the FSAR [Section 3.3.4.1.5 A] and independent confirmatory calculations performed by the staff shows that the array reactivity can be maintained subcritical (keff <0.95) in this accident situation by filling the DSC with borated water before wet loading or unloading. The minimum level of boration required as determined by the staff analysis, based on 4.0 wt. % U-235 enrichment of unirradiated B&W 15x15 fuel assemblies was determined to be 2,000 ppm. The analysis presented in the FSAR determined the minimum level of boration to be 1,810 ppm. In lieu of resolution of the difference between the staff and FSAR analysis of the required minimum level of boration, the more conservative value of the staff analysis is taken".

FSAR section 3.6 provides the analysis to qualify the storage of PWR fuel assemblies with an equivalent unirradiated enrichment of greater than 1.45 wt. % U-235. Note that the increase in the soluble boron loading for these fuel assemblies is due to the differences in the criticality analysis methodologies, computer codes and assumptions used for the original licensing basis versus the analysis of Section 3.6.

3.3.4.1.1 Control Methods for Prevention of Criticality

A. General Methodology (NUHOMS®-24P)

The NUHOMS®-24P DSC basket is designed to ensure nuclear criticality safety during worst case wet loading operations. Rigorous measures are taken to exclude the possibility of flooding the DSC cavity during the transfer operations and storage period. Prior to these operations, the DSC is vacuum dried, backfilled with helium, double seal welded, and helium leak tested to assure weld integrity. Under these dry conditions there is no possibility of exceeding criticality safety limits. Since the transfer cask and HSM are designed to provide adequate drop and/or missile protection for the DSC, and the DSC basket is designed to maintain the fuel configuration after a drop accident, there is no credible accident scenario which would result in the possibility of water intrusion into the DSC; nor is there a credible accident scenario which would result in the canister being breached and flooded.

Control methods for the prevention of criticality for the NUHOMS®-24P DSC consist of the material properties of the fuel, administrative procedures (i.e., a plant-specific system

using records or tests to document initial enrichment and burnup of the selected fuel assemblies), the geometrical arrangement of the basket and the inherent neutron absorption in the stainless steel guide sleeve assemblies.

Credit for burnup is taken by calculating an initial enrichment equivalent to the fissile inventory of the spent fuel. The CSAS2 criticality analysis sequence included in the SCALE-3 package of computer codes (3.44) is used to demonstrate subcriticality during moderation by pure water having a wide density range. Credit is taken for negative reactivity due to stable fission products.

The DSC basket is shown by analysis in Chapter 8 to maintain its configuration and location of the fuel assemblies after a drop accident. The DSC shell is shown to maintain its integrity during the accident so that no credible accident exists whereby the DSC may be accidentally flooded with fresh water. Water intrusion is not feasible since the DSC has been qualified to be helium leak tight for all postulated events which is a much more limiting condition. ISFSI flooding does result in canister immersion and a water reflector for the spent fuel matrix. However, as has been shown for the NUHOMS®-52B DSC in Section 3.3.4.2, this case does not limit the design. Since moderator intrusion during storage is prevented, subcriticality of the DSC is assured during storage at the ISFSI.

B. Design Parameters for Criticality Model (NUHOMS®-24P)

The geometry and fuel characteristics of the NUHOMS®-24P DSC criticality model are shown in Table 3.3-3. Figure 3.3-1 shows the actual geometry of the DSC and the geometry of the CSAS2 model. Figure 3.3-2 describes the modeling of the fuel assembly guide sleeves with the heterogeneous fuel assembly region inside.

The reactivity equivalence curve presented in Figure 3.3-3 is used to determine the acceptability of storing specific fuel assemblies in the NUHOMS®-24P DSC. The predetermined residual reactivity limit is selected to correspond roughly to a fuel assembly at 80 percent of what is typically considered full burnup. The concept of reactivity equivalency is used to develop a curve of constant reactivity through the enrichment/burnup space assuming the DSC was fully loaded with spent B&W 15x15 fuel assemblies. The resulting curve of reactivity equivalence for the DSC is presented in Figure 3.3-3. The reactivity equivalence curve extends from a zero burnup, initial enrichment equivalent point of 1.45 wt. % (weight percent) U-235 to a high enrichment endpoint corresponding to 4.0 wt. % U-235 initial enrichment irradiated for approximately 37,000 MWD/MTU.

The selection of a 15x15 fuel assembly for PWR criticality calculations has been shown by many analyses to be the most reactive under a variety of conditions when compared to other PWR fuel assemblies (i.e., 14x14, 16x16, and 17x17) (3.28, 3.29, 3.30, 3.31). Thus the B&W 15x15 fuel selected as the design basis for the NUHOMS®-24P canister forms a sufficient basis to permit storage of PWR fuel types which meet the requirements of Section 10.3.1.1.

9. A uniform axial burnup profile is assumed in CSAS2 irradiated fuel equivalence calculation cases. Consideration is given to variations in axial burnup as required by ANSI/ANS-8.17-1984; sensitivity calculations are performed and an appropriate bias is applied.

10. Irradiated fuel is assumed to be cooled 7.5 years following discharge from the reactor.

11. For irradiated fuel equivalence calculations, credit is taken for 34 major fission product absorbers identified as stable sources of negative reactivity in Reference 3.41. Quantitative estimates of negative reactivity credit taken for the major fission product absorbers at several representative initial enrichment points along Figure 3.3-3 reactivity equivalence curve are provided in Table 3.3-5.

Table 3.3-3 provides the nominal dimensions of the NUHOMS®-24P DSC and transfer cask geometry illustrated in Figure 3.3-1. Figure 3.3-1 also provides an illustration of the fuel array and reflectors modeled in CSAS2 for the nominal case.

B. Nominal Case (NUHOMS®-24P)

Referring to Figure 3.3-1, Figure 3.3-2, and Table 3.3-3, the following zones are explicitly modeled in the NUHOMS®-24P DSC nominal case CSAS2 analysis:

1. Nominal DSC basket geometry parameters

2. 0.369 inch OD and 0.370 inch OD fuel pellets (208 rods)

3. 0.430 inch OD Zircaloy fuel clad (208 rods)

4. Void gap between fuel and clad

5. 0.530 inch OD control rod guide tubes (16 tubes)

6. 0.493 inch OD instrument tube

The nominal case CSAS2 calculation in which 50,100 neutron histories are followed, results in a k_{eff} of 0.87170 with a 95 percent probability at a 95 percent confidence level uncertainty of ± 0.00488.

C. Criticality Analysis Variability (NUHOMS®-24P)

The bias and uncertainty methodology applied in the calculation of the NUHOMS®-24P DSC final k_{eff} result is based on CSAS2/123GROUPGMTH calculated results for the set of 40 critical experiments summarized in Table 3.3-6. All 40 critical experiments included in the method benchmark are similar to zero burnup/nominal case flooded DSC conditions (i.e., all are water moderated, low enrichment heterogeneous UO2 systems). A representative number of the benchmark experiments include stainless steel separating materials and are very similar to the zero burnup/nominal case flooded DSC conditions. The inclusion of benchmark systems which differ from flooded DSC conditions in some respects, such as separating materials, is justified by inspection of the Table 3.3-6 k_{eff} results which do not indicate any significant trends. The calculated k_{eff} results for the diverse group of experiments analyzed demonstrates the calculational accuracy of the method under a variety of conditions including conditions representative of the zero burnup/nominal case flooded DSC fuel storage array.

The UO_2 experiments summarized in Table 3.3-6 provide the basis for the method and library bias and variability used in the final k_{eff} calculation. Additional benchmark calculations are performed to demonstrate that the irradiated fuel criticality/equivalence method used is conservative when compared to the method bias basis UO_2 benchmark results. CSAS2/123GROUPGMTH benchmark results for systems containing PuO_2-UO_2 mixed oxide fuel pins are provided in the Table 3.3-7. Benchmark data representative of irradiated fuel assemblies is obtained from the results of CASMO-2 infinite lattice criticality calculations; the results of benchmark comparisons between CASMO-2 and CSAS2/CASMO2/SAS2 calculated k_{inf} values are provided in Table 3.3-8. Inspection of the benchmark results provided in Table 3.3-7 and Table 3.3-8 demonstrates that the criticality/equivalence method used in the subject calculations conservatively overpredicts k_{eff} for systems containing plutonium or irradiated fuel of the type proposed to be stored.

In addition to UO_2 experiments, the CSAS2/123GROUPGMTH method is validated against PuO_2 - UO_2 mixed oxide experiments (3.42) and k_{inf} results generated by CASMO-2E for irradiated B&W 15x15 fuel assembly arrays (3.43). The results of these additional validation runs indicate that the CSAS2/123GROUPGMTH method conservatively overpredicts k_{eff} for systems containing Pu and/or fission products.

D. Additional Biases and Uncertainties (NUHOMS®-24P)

The 95/95 uncertainty in the NUHOMS®-24P DSC nominal case analysis is 0.00488 Δk. A statistical bias of +0.00488 and a 95/95 uncertainty of 0.01161 Δk is associated with the CSAS2 method used. In addition to these uncertainties, there are other considerations which may effect the final k_{eff} value assigned to the array. These considerations are treated as either worst-case in the nominal run or sensitivity runs are performed to determine the Δk associated with a variable parameter (e.g., guide sleeve thickness).

All values of burnup used to develop the Figure 3.3-3 equivalence curve are selected to maintain k_{eff} equal to or below 0.94782.

Components of the final k_{eff} result for all cases analyzed are provided in Table 3.3-11. Additional information on the NUHOMS®-24P DSC criticality analysis with burnup credit is contained in Appendix F.

3.3.4.1.4 Off-Normal Conditions (NUHOMS®-24P)

Postulated off-normal conditions do not result in a NUHOMS®-24P DSC storage array reactivity which exceeds the k_{eff} value calculated and presented in Section 3.3.4.1.3.

The off-normal conditions considered include misloading of one or more high enrichment nonirradiated fuel assemblies into the DSC, and optimum moderation.

A. Misloading a High Enrichment Assembly

Misloading one or more fuel assemblies which do not qualify as acceptable for storage in the NUHOMS®-24P DSC according to the burnup equivalence curve shown in Figure 3.3-3 do not result in a k_{eff} value greater than the 0.95 criterion. The double contingency principle of ANSI/ANS 8.17-1984 can be applied to take credit for dissolved boron which is normally present in the PWR spent fuel pool and DSC during wet loading operations. The approximate 0.34 Δk negative reactivity provided by 2,000 ppm of soluble boron would more than compensate for the additional reactivity added by the misloading of one or more unqualified, high enrichment fuel assemblies.

B. Optimum Moderation

Optimum moderation conditions are considered and a conservative bias is applied in the normal case analysis presented in Section 3.3.4.1.3. Therefore, the presence of a pure water moderator of optimum density does not result in a NUHOMS®-24P DSC storage array reactivity which exceeds the k_{eff} value calculated and presented in Section 3.3.4.1.3. Further, it is possible that moderator densities other than unity could occur if moderator boiling temperatures were to be reached; however, unirradiated high enriched fuel provides no decay heat for moderator heating. Thus, this is a non-mechanistic unrealistic condition. The condition can be conceived for optimal moderation for irradiated fuel with high decay heats; however, the reactivity of the fuel is significantly reduced compared with that of unirradiated fuel.

3.3.4.1.5 Safety Criteria Compliance (NUHOMS®-24P)

The calculated worst-case k_{eff} value for a fully loaded NUHOMS®-24P DSC flooded with pure unborated water of a uniform optimum density is 0.94782. This calculated maximum k_{eff} value includes consideration of geometrical, material, and burnup uncertainties and biases at a 95/95 tolerance level as required by ANSI/ANS 57.2-1983 to demonstrate criticality safety.

Additionally, off-normal conditions potentially resulting in reactivity increases over the normal conditions considered are addressed and found to be acceptable.

The analyses presented in this SAR section and Appendix F demonstrate that the ANSI/ANS 57.2-1983 criteria limiting k_{eff} to 0.95 is satisfied under all postulated conditions for the NUHOMS®-24P DSC.

3.3.4.2 NUHOMS®-52B DSC Criticality Safety

3.3.4.2.1 Control Methods for the Prevention of Criticality

A. General Methodology (NUHOMS®-52B)

The NUHOMS®-52B DSC is designed to provide criticality control through a combination of mechanical and neutronic isolation of fuel assemblies as described in the following paragraphs. A support structure composed of six axially oriented support rods and nine spacer disks provides positive location for the fuel assemblies under both normal and accident conditions. The basket assembly utilizes fixed neutron absorbers which effectively decouple fuel assemblies.

The neutron absorber is a borated stainless steel which can be supplied with boron additions up to 2% to provide thermal neutron absorption. This license application is for 0.75% minimum boron content absorber material. Although borated stainless steel would be an effective load bearing component in the basket assembly, the NUHOMS®-52B DSC basket assembly has been designed such that the neutron absorbers are in no way loaded by other DSC structural components or the spent fuel assemblies. The absorber sheets are fastened to and supported by the basket assembly spacer disks. As a result, the sheets are captured such that no welding or bending is performed on them during fabrication.

Borated stainless steel is chosen for the neutron absorbers due to its desirable neutron attenuation, homogeneity, corrosion resistance, strength, and toughness. Commercial experience with borated stainless material includes a wide range of applications such as the TN-REG and TN-BRP transport/storage casks, Indian Point fuel storage racks, and scram balls in British Magnox reactors.

It should also be noted that a recent regulatory mandate [3.68 and 3.69] has required all U.S. utilities owning and operating nuclear plants to evaluate and protect their facility against the threat of sabotage by car or truck bomb. This evaluation has resulted in increased protection of the plant's vital areas through adoption of explosion-proof barriers and gates. Site specific ISFSI locations within the plant's protected area would be subject to the requirements of this mandate, thus requiring the applicants to ensure the same level of barrier protection for ISFSIs to safeguard against possible sabotage.

Licensees are required to verify that loadings resulting from potential fires and explosions are acceptable in accordance with 10CFR72.212(b)(2).

3.3.7 Materials Handling and Storage

3.3.7.1 Spent Fuel Handling and Storage

All spent fuel handling outside the plant's fuel pool is performed with the fuel assemblies contained in the DSC. Subcriticality during all phases of handling and storage is discussed in Section 3.3.4. The criterion for a safe configuration is an effective mean plus two-sigma neutron multiplication factor (k_{eff}) of 0.95. Section 3.3 calculations show that the expected k_{eff} value is below this limit.

Lift height restrictions are imposed on the TC and DSC with regard to their location and load temperatures. These restrictions are provided in Technical Specifications 1.2.10 and 1.2.13.

3.3.7.1.1 Cladding Temperature Limits

Maximum allowable cladding temperature limits are determined for both BWR and PWR design basis fuel according to the methodology presented in Reference 3.21. The maximum allowable average cladding temperature for long term storage is based on the end of life hoop stress in the cladding and the cladding temperature at the beginning of dry storage. The method is estimated to calculate a storage temperature limit that will result in a probability of cladding breach of less than 0.5% in the peak rod during storage. Using this methodology produces cladding temperature limit of 381°C for design basis PWR fuel and 394°C for the design basis BWR fuel cooled for five years or more. Appendix K addresses the cladding temperature limits for the BWR fuel in the NUHOMS®-61BT DSC and Appendix L addresses the cladding temperature limits for the PWR fuel in the NUHOMS®-24PT2 DSC. Since the damage mechanism in this methodology is thermal creep, the temperature limits are based on an average long term ambient temperature during storage of 70°F.

381°C (718°F) and 394°C (741°F) are the cladding temperature limits calculated for design basis 5-year cooled PWR and BWR fuel, respectively. Three steps were taken to extend the same methodology to the range of cooling times in the Fuel Qualification Table shown in 72-1004 CoC technical specifications. First, the same thermal computer

models used to perform the design basis cladding temperature calculation were run parametrically to determine cladding temperature vs. heat input for the PWR and BWR baskets. Second, the methodology of Reference 3.21 was used to develop a relationship between the maximum cladding temperature limit vs. cooling times beyond 5 years. This relationship is shown as a function of fuel burnup in Figure 3.3-17 for PWR fuel and in Figure 3.3-18 for BWR fuel. Third, these two functions were combined to obtain maximum heat input vs. cooling time. In this way, each cell of the Fuel Qualification Table has its own unique cladding temperature limit based on the same methodology as was used for the design basis fuel assemblies.

Higher cladding temperatures may be sustained for brief periods without affecting cladding integrity, however. During short term conditions such as DSC drying, transfer of the DSC to and from the HSM, and off-normal and accident temperature excursions, the maximum fuel cladding temperature is limited to 570°C (1,058°F) or less. This value is based on the results of experiments which have shown that Zircaloy clad rods subjected to short term temperature excursions below 570°C did not show indications of failure (3.20).

3.3.7.1.2 Fuel Rod Horizontal Storage Effects

There is considerable industry experience in the shipment of fuel assemblies in the horizontal position without any indication of fuel rod creep or sag. During overseas shipments, spent fuel assemblies remain horizontal for up to two to three months with estimated cladding temperatures up to 385°C. It should also be noted that the environment for shipping fuel assemblies, given the handling and transportation shock loadings and vibrations is much harsher than that of passive environment of dry storage.

Analytical studies of fuel rod creep behavior have also been conducted in conjunction with the NRC approval of the NUHOMS®-24P TR as documented in Reference 3.51. The studies utilized the creep equation of M. Peehs, et. al. to determine whether creep of fuel were found to be less than 1% for the total storage period. The deflection of the fuel rods between spacer grids was calculated directly since creep effects were found to be negligible. Using standard beam theory for a uniformly loaded tubular beam, conservatively neglecting the bending stiffness of the fuel itself, the maximum deflection over the storage period was found to be 0.015 inches. Deflections of such magnitude do not impede retrieval of the fuel assemblies from the DSC, therefore these effects are not evaluated further.

3.3.7.1.3 Surface Contamination Limits

DSC exterior contamination is minimized by preventing spent fuel pool water from contacting the DSC exterior. DSC loading procedures require that the annulus between the transfer cask and DSC be filled with demineralized water and sealed prior to immersion in the spent fuel pool. Annulus sealing is accomplished by an inflatable seal between the transfer cask and DSC. The combination of the above operations provides assurance that the DSC exterior surface has less residual contamination than required for

Table 3.3-2
Radioactive Material Confinement Barriers for NUHOMS® System

Confinement Barriers and Systems

1. Fuel Cladding
2. DSC Containment Pressure Boundary
3. Inner DSC Closure Weld
4. Outer DSC Closure Weld
5. DSC Cover Plates

Table 3.3-3
Design Parameters for Criticality Analysis of the NUHOMS®-24P DSC

Parameter	Design Value
Fuel Assemblies	
Number/Type	24/PWR
Rod Array	15x15
Number of Fuel Rods	208
Number of CR Guide Tubes	16
Number of Instrument Tubes	1
Rod Pitch (inch)	0.568
Burnup Credit	Yes
Fissile Content (% initial U equivalent)	
U-235 (Credit for Soluble Boron)	4.00
U-235 (Credit for Burnup)	1.45
U-238	Balance
Fuel Pellets	
Density (g/cm^3)	10.41 max.
Diameter (inch)	0.369 & 0.370
Fuel Rod Cladding	
Material	Zircaloy-4
Thickness (inch)	0.0265
Outside Diameter (inch)	0.430
Control Rod Guide Tubes	
Material	Zircaloy-4
Thickness (inch)	0.016
Outside Diameter (inch)	0.530
Instrument Tube	
Material	Zircaloy-4
Thickness (inch)	0.026
Outside Diameter (inch)	0.493

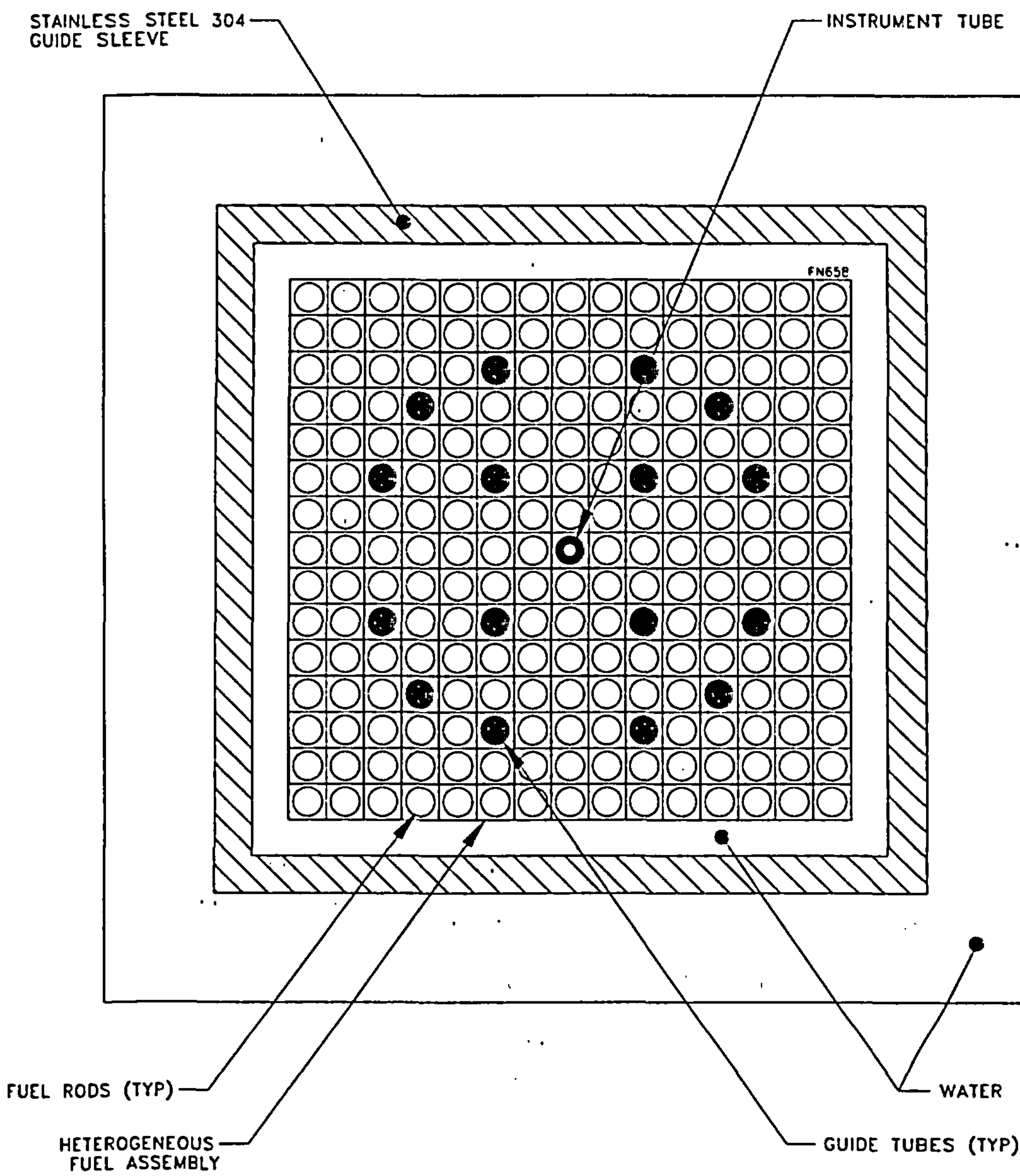


Figure 3.3-2
KENO Model for NUHOMS®-24P Fuel

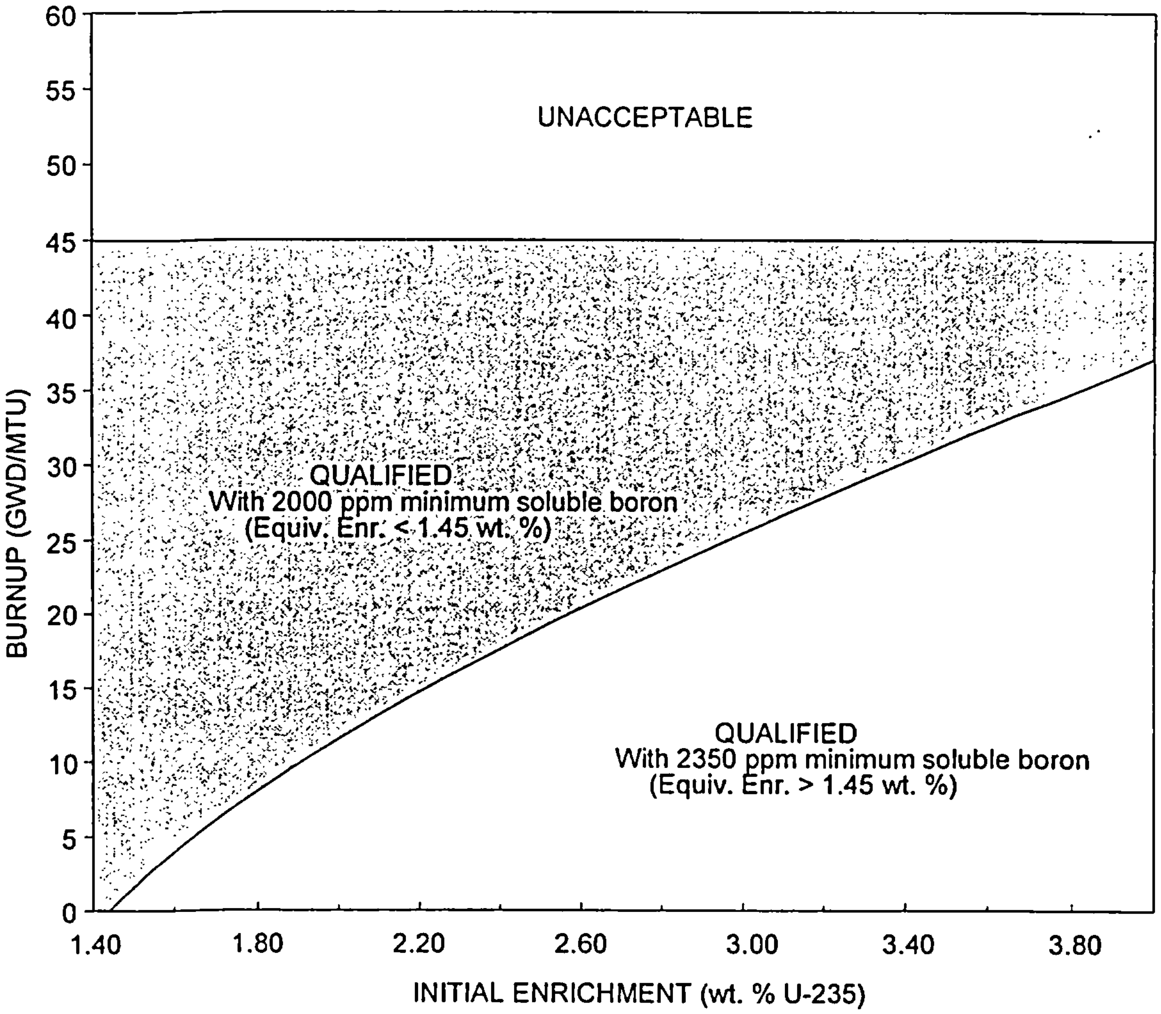


Figure 3.3-3
NUHOMS®-24P DSC Burnup Equivalence Curve

3.4 Classification of Structures, Components, and Systems

Table 3.4-1 provides a list of major NUHOMS® ISFSI components and their classification. Components are classified in accordance with the criteria of 10CFR72 (3.6). Structures, systems, and components classified as "important to safety" are defined in 10CFR72.3 as those features of the ISFSI whose function is:

A. To maintain the conditions required to store spent fuel safely.

B. To prevent damage to the spent fuel container during handling and storage.

C. 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.

These criteria are applied to the NUHOMS® system components in determining their classification in the paragraphs which follow.

3.4.1 Dry Shielded Canister

The DSC is considered "important to safety" since it provides fuel assembly support required to maintain the assumed fuel geometry for criticality control. Accidental criticality inside a DSC could lead to off-site doses comparable with the limits in 10CFR100 which must be prevented. The DSC also provides the primary containment for radioactive materials. Therefore, the DSC is designed to remain intact under all accident conditions identified in Chapter 8 with no loss of function. The DSC is designed, constructed, and tested in accordance with "important to safety" requirements as defined by 10CFR72, Subpart G, paragraph 72.140(b) and described in Chapter 11. The welding materials required to make the closure welds on the DSC inner and outer top cover plates are purchased to the same ASME Code criteria as the DSC (Subsection NB, Class 1).

3.4.2 Horizontal Storage Module

The HSM is considered "important to safety" since it provides physical protection and shielding for the spent fuel container (DSC) during storage. The reinforced concrete HSM is designed in accordance with ACI 349-85 and the level of testing, inspection, and documentation provided during construction and maintenance is in accordance with the quality assurance requirements as defined in 10CFR72, Subpart G, paragraph 72.140(b) and as described in Chapter 11.

3.4.3 ISFSI Basemat and Approach Slabs

The ISFSI basemat and approach slabs are not considered "important to safety" and are designed, constructed, maintained, and tested as a commercial grade items.

Licensees are required to perform an assessment to confirm that the license seismic criteria are met and that the HSM foundation design meets the applicable design requirements.

3.4.4 Transfer Equipment

3.4.4.1 Transfer Cask and Yoke

The on-site transfer cask is "important to safety" since it protects the spent fuel container (DSC) during handling and is part of the primary load path used while handling the DSC in the fuel/reactor building. An accidental drop of a loaded transfer cask has the potential for creating conditions in the plant which must be evaluated. Therefore the transfer cask is designed, constructed, and tested in accordance with "important to safety" requirements as defined by 10CFR72, Subpart G, paragraph 72.140(b) and described in Chapter 11.

The lifting yoke used for handling of the transfer cask within the fuel/reactor building is designed and procured as a "safety related" component as it is used by the licensee (utility) under the 10CFR50 (3.65) program. The lifting yoke is controlled by NUREG-0612 (3.66) and is designed to ANSI N14.6-1986 criteria for non-redundant yokes. Therefore, the lifting yoke is designed, constructed, and tested in accordance with "safety related" requirements as defined by 10CFR50, Appendix B and described in Chapter 11.

Due to site unique requirements, rigid or sling lifting members may be used to augment the lifting yoke. These members shall be designed, fabricated and tested in accordance with the same requirements as the cask lifting yoke.

3.4.4.2 Other Transfer Equipment

The NUHOMS® transfer equipment (i.e., ram, skid, trailer) are necessary for the successful loading of the DSC into the HSM. However, the analyses described in Chapter 8 demonstrate that the performance of these items is not required 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. Therefore, these components are considered not "important to safety" and need not comply with the requirements of 10CFR72. These components are designed, constructed, and tested in accordance with good industry practices.

3.5 Decommissioning Considerations

The DSC is designed to interface with a transportation system for the eventual off-site transport of intact canisters by the DOE to either a monitored retrievable storage facility (MRS) or a permanent geologic repository, as discussed in Section 9.6. Decommissioning of the NUHOMS® ISFSI will be performed in a manner consistent with the decommissioning of the plant itself since all NUHOMS® system components are constructed of materials similar to those found in existing plants.

If the fuel is to be removed from the DSC at the plant prior to shipment, the DSC will likely be contaminated internally by crud from the spent fuel and may be slightly activated by spontaneous neutron emissions from the spent fuel. The DSC internals may be cleaned to remove surface contamination and the DSC disposed of as low-level waste. Alternatively, if the contamination and activation levels of the DSC are small enough (to be determined on a case-by-case basis), it may be possible to decontaminate the DSC and dispose of it as commercial scrap pending NRC rulings on below regulatory concern (BRC) waste disposal issues.

While the intent for the NUHOMS® system includes the eventual disposal of each DSC following fuel removal, current closure weld designs do not preclude future development of a non-destructive closure removal technique that allows for reuse of the DSC shell/basket assembly. Economic and technical conditions existing at the time of fuel removal would be assessed prior to making a decision to reuse the DSC.

The exact decommissioning plan for the ISFSI will be dependent on the DOE's fuel transportation system capability and requirements for a specific plant. Because of the minimal contamination of the outer surface of the DSC, no contamination is expected on the internal passages of the HSM. It is anticipated that the prefabricated HSMs can be dismantled and disposed of using commercial demolition and disposal techniques. Alternatively, the HSMs may be refurbished and reused at another site or at the MRS for storage of intact NUHOMS® DSCs transported from the plant.

3.6 Criticality Evaluation for Fuel With an Equivalent Unirradiated Enrichment of Greater than 1.45 wt. % U-235

The NUHOMS®-24P design criticality analysis is documented in Section 3.3.4.1 and Appendix J of this FSAR. A new evaluation has been performed to determine the boron concentration in the DSC cavity water during loading/unloading operations necessary to keep the system reactivity below 0.95 for fuels with an equivalent unirradiated enrichment of greater than 1.45 wt. % U-235 (FSAR Figure 3.3-3).

KENO-Va (CSAS25 of SCALE 4.4)[3.71] is used to demonstrate that the worst case fuel assembly type B&W 15x15 fuel designs with a maximum of 4.0 wt. % enriched U-235 fuel are bounded by the upper subcritical limit for storage in the standardized NUHOMS®-24P system. The analysis is performed with and without Burnable Poison Rod Assemblies (BPRAs). No credit was taken for BPRA cladding and absorbers, rather the BPRA is modeled as $^{11}B_4C$ in the entire guide tube. Thus, the highly borated moderator between the guide tube and the BPRA rodlet is modeled as $^{11}B_4C$. The inclusion of more Boron-11 and carbon enhances neutron scattering causing the neutron population in the fuel assembly to be slightly increased which increases reactivity. The analysis was performed with soluble boron in the DSC cavity and with pure water or void in the fuel-cladding gap and upper plenum region.

3.6.1 Discussion and Results

The results demonstrate that boron loading of 2350 ppm will maintain the system reactivity below 0.95. This is demonstrated by modeling the design basis B&W 15x15 fuel assembly (with and without BPRAs) with pure water or void in the fuel-cladding gap and upper plenum region in a canister flooded with borated water with a moderator density range of 0.0001 g/cc to approximately 1.0 g/cc.

The evaluation used unirradiated B&W 15x15 Mark B loaded with 4.0 wt. % enriched U-235 fuel pins in 2350 ppm borated water. Three parameters are included: moderator density, contents of guide tubes and instrument tubes, and the fuel-cladding gap contents. Consistent with the criticality analysis presented in Appendix J of the FSAR, it is assumed that a comparison of fuel assemblies with and without BPRAs is bounded by modeling the volumes which could be occupied by BPRAs as filled with $^{11}B_4C$. This is done to bound the reactivity effects of BPRAs stored with fuel. Modeling the BPRAs as $^{11}B_4C$ is appropriate since the net effect is to substitute material with a very small thermal absorption cross section for material which in reality would have some remaining B_{10}. The scattering or moderating effect of pure $^{11}B_4C$ is very similar to Zircaloy. In addition, the instrument tubes are conservatively assumed to be filled with $^{11}B_4C$. It is assumed that 100% dense pure water fills the fuel-cladding gap and upper plenum region. A series of calculations are also run with void in the cladding gap and plenum region to determine the bounding condition.

3.6.2 Package Fuel Loading

The package fuel loading remains unchanged from previous analysis presented in Section 3.3.4.1 and Appendix J of the FSAR. The B&W 15x15 fuel assembly design parameters and layout used in this evaluation are given in Appendix J Table J.6-1 and Figure J.6-1.

3.6.3 Model Specification

Four cases are evaluated: (1) with BPRAs (2) without BPRAs (3) with water in the fuel cladding gap for cases 1 and 2 and (4) without water in the fuel cladding gap for cases 1 and 2. For each case, simulations at moderator densities ranging from approximately 1.0 g/cc to 0.0001 g/cc were performed to ensure that the resulting k_{eff} does not exceed the maximum allowable upper subcritical limit (USL) value.

The model simulates the actual NUHOMS®-24P DSC geometry including the 24 fuel assemblies within the guide sleeves, the four axial support rods, and the steel DSC shell. Each of the fuel assembly components is modeled discretely using the dimensions as described in section 3.6.2. The DSC is flooded with borated water and an additional 12 inches of borated water outside the DSC is added for increased reflection. In addition, the worst case dimensional tolerances for the DSC basket were used. The models assumed minimum spacer disk ligament widths, maximum over sleeve wall thickness, and all fuel assemblies pushed toward the center of the DSC in the guidesleeves.

The "worst case" k_{eff} values from the KENO Va runs are adjusted for uncertainty, such that:

$$k_{eff} = k_{calculated} + 2\sigma$$

This approach is conservative since the $k_{calculated}$ values assume:

- Unirradiated fuel – no credit taken for fissile depletion or fission product poisoning.
- No credit is taken for residual neutron absorber in the BPRAs.
- The fuel enrichment is modeled as uniform everywhere throughout the assembly with a 4.0 wt. % lattice average enrichment.
- All fuel rods are modeled with 100% dense pure water in the fuel-cladding gap and plenum region.

- 3-D modeling is implemented in all KENO models. The DSC is modeled having a finite length with water albedo boundary conditions specified on all sides.

- Only the active fuel length and the upper plenum region are explicitly modeled. The presence of fuel assembly components above and below these regions is modeled as borated water.

- The most material condition was assumed for the over sleeves. The neutron absorption gained in the thicker steel sheets is offset by the decreased absorption in the displaced borated water. The least material condition was assumed for the guide sleeve thickness. This allowed for the fuel assemblies to be pushed as far inward as possible.

The KENO models consist of 1586 axial layers stacked into an array. The layers consist of partial spacer disk or partial moderator regions inside and outside of the active fuel region and upper plenum region as presented in Figure 3.6-1. The very top and bottom layers of the model are the DSC steel cylinders. The DSC model contains 8 spacer disk regions, each 2 inches thick, surrounded by a total of 9 non-disk regions. The center-to-center spacing of the spacer disk intervals varies over a range from 5.5 inches to 22.6 inches. A 12 inch water differential albedo boundary condition is used for all cases.

UNIT 33 is a slice through the cask at the DSC spacer disk level. UNIT 34 is a slice through the moderator region between spacer disks as shown in Figure 3.6-2. UNIT 37 is a slice through the moderator region between the spacer disks including the over sleeve. UNIT 134 is a slice through the moderator region between the spacer disks at the plenum height. UNIT 137 is a slice through the moderator region between the spacer disks including the over sleeve at the plenum height. UNIT numbers 1-8 are used to represent the active fuel assemblies in both the spacer disk region and in the moderator region. Units 100-800 are used to represent the plenum region in both the spacer disk region and in the moderator region. The fuel assemblies are inserted into the model using KENO's HOLE capability.

No attempt has been made to model fission products, burnable poisons, or axial and radial variations in initial fuel enrichment. Instead, fuel assemblies have been modeled as if they were composed of only a single enrichment unirradiated fuel. This assumption results in a very large margin of conservatism in the calculated k_{eff}.

All calculations were performed using SCALE4.4 [3.71], CSAS25 and the 44-group ENDF/B-V cross section library.

The dimensional data with worst case tolerances for the NUHOMS®-24P DSC used in the CSAS25 (KENO-Va) models is given in Table 3.6-1. A quarter-core layout of fuel assemblies, support tie rods and stainless shell is shown in Figure 3.6-3.

3.6.4 Criticality Analysis

The maximum k_{eff} for the NUHOMS®-24P was determined to be 0.9339±0.0009 (0.9357 w/2 σ) from KENO Va case with pure water in the fuel-cladding gap and upper plenum region, no BPRAs, internal moderation at 0.4 g/cc, and a boron loading of 2350 ppm. Therefore, the maximum final k_{eff} (0.9357) in the moderator density in the range of 0.0001 g/cc to approximately 1.0 g/cc, meets the USL criteria of less than 0.9410 as defined in FSAR section 3.6.5. The criticality results are presented in Figure 3.6-4. Table 3.6-2 through Table 3.6-5 present the results for each set of KENO cases.

3.6.5 Critical Benchmark Experiments

The criticality safety analysis presented in Section 3.6 used the CSAS25 module of the SCALE system of codes.

The analysis presented in Section 3.6 uses the fresh fuel assumption for criticality analysis. The analysis employed the 44-group ENDF/B-V cross-section library because it has a small bias, as determined by the 125 benchmark calculations described in reference [3.70]. The USL Method 1 (USL-1) was determined using the results of these 125 benchmark calculations.

The benchmark problems used to perform this verification are representative of benchmark arrays of commercial light water reactor (LWR) fuels with the following characteristics:

(1) water moderation

(2) boron neutron absorbers

(3) unirradiated light water reactor type fuel (no fission products or "burnup credit") near room temperature (vs. reactor operating temperature)

(4) close reflection

(5) Uranium Oxide

The 125 uranium oxide experiments were chosen to model a wide range of uranium enrichments, fuel pin pitches, assembly separation, concentration of soluble boron and control elements in order to test the codes ability to accurately calculate k_{eff}. These experiments are discussed in detail in reference [3.70]. The Run ID referred to in the following sub-sections are identical to those used in [3.70].

3.6.5.1 Benchmark Experiments and Applicability

A summary of all of the pertinent parameters for each experiment is included in Table 3.6-6 along with the results of each run. The best correlation is observed for fuel assembly separation distance with a correlation of 0.65. All other parameters show much lower correlation ratios indicating no real correlation. All parameters were evaluated for trends and to determine the most conservative USL.

The USL is calculated in accordance to NUREG/CR-6361. USL applies a statistical calculation of the bias and its uncertainty plus an administrative margin (0.05) to the linear fit of results of the experimental benchmark data. The basis for the administrative margin is from NUREG-1536. Results from the USL evaluation are presented in Table 3.3.6-7.

The criticality evaluation used the same cross section set, fuel materials and similar material/geometry options that were used in the 125 benchmark calculations as shown in Table 3.6-6. The modeling techniques and the applicable parameters listed in Table 3.3.6-8 for the actual criticality evaluations fall within the range of those addressed by the benchmarks in Table 3.6-6.

3.6.5.2 Results of the Benchmark Calculations

The results from the comparisons of physical parameters of each of the fuel assembly types to the applicable USL value are presented in Table 3.3.6-8. The minimum value of the USL was determined to be 0.9410 based on comparisons to the limiting assembly parameters as shown in Table 3.3.6-8.

3.6.6 Example Keno V.a Input Listing

B&W 15x15 Mark B Without BPRAs

```
=csas25
4.0% Enriched, 2350 ppm, .4 g/cc, water in the gap, 8" upper plenum
```

Title line describing 4.0 wt. % U-235 enrich with a soluble boron content of 2350 ppm, 0.4 g/cc moderator density inside the DSC, pure water in the fuel-cladding gap and plenum region, and an actual plenum height of 8.7"

```
44GROUPNDF5 LATTICECELL
```

The 44-group ENDF/B-V cross section library is called out using square lattice geometry. Material Specifications Follow:

```
UO2           1  0.9453            293.0  92235 4.0  92238 96.0   END
SS304         2  1.00              293.0                          END
'Borated Water'
H             3  0  0.0267         293.0                          END
O             3  0  0.0134         293.0                          END
B-11          3  0  4.194E-05      293.0                          END
B-10          3  0  1.042E-05      293.0                          END
CARBONSTEEL   4  1.00              293.0                          END
H2O           5  DEN=1.0000  1.0   293.0                          END
PB            7  1.00              293.0                          END
AL            8     0  7.0275-03   293.0                          END
CA            8     0  1.4835-03   293.0                          END
C             8     0  8.2505-03   293.0                          END
H             8     0  5.0996-02   293.0                          END
FE            8     0  1.0628-04   293.0                          END
O             8     0  3.7793-02   293.0                          END
SI            8     0  1.2680-03   293.0                          END
H2O           9  DEN=1.0000  1.0   293.0                          END
ZIRC4        10   1.00             293.0                          END
END COMP
SQUAREPITCH  1.44272 0.9362 1  3 1.0922  10 0.9576   5          END
4.0% Enriched, 2350 ppm, .4 g/cc, water in the gap, 8" upper plenum
READ PARA
gen=500 npg=1000 nsk=5 nub=yes run=yes tme=300 plt=no
END PARA
READ GEOM
   UNIT 1     COM='FA #1,#2,#4 AND GS, SD, INNER FA(S), outer sleeve, NE QUAD'
       CUBOID          3          1
       11.1316    -11.5506    11.1316    -11.5506     0.2540    0.0
   HOLE 32        -0.69215    -0.69215          0
       CUBOID          2          1
      11.7031   -12.1221    11.7031   -12.1221     0.2540       0.0
       CUBOID          3          1
    12.12215 -12.12215  12.12215 -12.12215     0.2540       0.0
```

Fuels Assemblies 1, 2, and 4 are all located in the NE quadrant of the DSC model and are the inner most assemblies. Unit 1 describes the geometry used for the guide sleeve locations that also have the outer sleeve. The third cuboid in the model defines the outer boundary of the unit and is composed of borated water. The second cuboid defines the guide sleeve + over sleeve position within this volume. The first cuboid is the borated water region within the guide sleeve. The difference between cuboid one and cuboid two is the thickness of the guide sleeve and the over sleeve. Hole 32 is the "active fuel assembly" and is placed in the lower left corner of the guide sleeve.

```
   UNIT 2     COM='FA #3,#5,#6 AND GS, SD, OUTER FA(S), NE QUAD.'
       CUBOID          3          1
      11.1316    -11.5506     11.1316    -11.5506     0.2540    0.0
   HOLE 32       -0.69215    -0.69215          0
```

```
    CUBOID        2          1
   11.3856    -11.8046    11.3856    -11.8046     0.2540        0.0
    CUBOID        3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540        0.0
```

Unit 2 describes the geometry used for just the guide sleeve locations. The third cuboid in the model defines the spacer disk cutout size. The second cuboid defines the guide sleeve position within the cutout. The first cuboid is the borated water region within the guide sleeve. The difference between cuboid one and cuboid two is the thickness of the guide sleeve. Hole 32 is the "active fuel assembly" and is placed in the lower left corner of the guide sleeve.

The following units are similar to units 1 and 2 with the exception of the fuel assembly position. In all cases the fuel assemblies are placed in the most radially inward position within the guide sleeve.

```
UNIT 3    COM='FA #7,#8,#10 AND GS, SD, INNER FA(S),outer sleeve NW QUAD.'
    CUBOID        3          1
    11.5506    -11.1316    11.1316    -11.5506     0.2540    0.0
HOLE 32        0.69215    -0.69215         0
    CUBOID        2          1
   12.1221    -11.7031    11.7031    -12.1221     0.2540    0.0
    CUBOID        3          1
 12.12215 -12.12215  12.12215 -12.12215     0.2540        0.0
UNIT 4    COM='FA #9,#11,#12 AND GS, SD, OUTER FA(S), NW QUAD'
    CUBOID        3          1
    11.5506    -11.1316    11.1316    -11.5506     0.2540    0.0
HOLE 32        0.69215    -0.69215         0
    CUBOID        2          1
   11.8046    -11.3856    11.3856    -11.8046     0.2540    0.0
    CUBOID        3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540        0.0
UNIT 5    COM='FA #13,#14,#16 AND GS, SD, INNER FA(S),outer sleeve SW QUAD.'
    CUBOID        3          1
    11.5506    -11.1316    11.5506    -11.1316     0.2540    0.0
HOLE 32        0.69215     0.69215         0
    CUBOID        2          1
   12.1221    -11.7031    12.1221    -11.7031     0.2540    0.0
    CUBOID        3          1
 12.12215 -12.12215  12.12215 -12.12215     0.2540        0.0
UNIT 6    COM='FA #15,#17,#18 AND GS, SD, OUTER FA(S), SW QUAD'
    CUBOID        3          1
    11.5506    -11.1316    11.5506    -11.1316     0.2540    0.0
HOLE 32        0.69215     0.69215         0
    CUBOID        2          1
   11.8046    -11.3856    11.8046    -11.3856     0.2540    0.0
    CUBOID        3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540        0.0
UNIT 7    COM='FA #19,#20,#22 AND GS, SD, INNER FA(S),outer sleeve SE QUAD'
    CUBOID        3          1
    11.1316    -11.5506    11.5506    -11.1316     0.2540    0.0
HOLE 32       -0.69215     0.69215         0
    CUBOID        2          1
   11.7031    -12.1221    12.1221    -11.7031     0.2540        0.0
    CUBOID        3          1
 12.12215 -12.12215  12.12215 -12.12215     0.2540        0.0
UNIT 8    COM='FA #21,#23,#24 AND GS, SD, OUTER FA(S), SE QUAD'
    CUBOID        3          1
    11.1316    -11.5506    11.5506    -11.1316     0.2540    0.0
HOLE 32       -0.69215     0.69215         0
    CUBOID        2          1
   11.3856    -11.8046    11.8046    -11.3856     0.2540        0.0
    CUBOID        3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540        0.0
UNIT 100  COM='FA #1,#2,#4 AND GS, SD, INNER FA(S), outer sleeve, NE QUAD'
    CUBOID        3          1
    11.1316    -11.5506    11.1316    -11.5506     0.2540    0.0
HOLE 132      -0.69215    -0.69215         0
    CUBOID        2          1
   11.7031    -12.1221    11.7031    -12.1221     0.2540        0.0
```

```
    CUBOID         3         1
 12.12215 -12.12215  12.12215 -12.12215     0.2540       0.0
```

Unit 100 through Unit 800 are similar to units 1 through 8 with the exception that Hole 32 is replaced with Hole 132 which models the plenum region of the fuel rod.

```
UNIT 200  COM='FA #3,#5,#6 AND GS, SD, OUTER FA(S), NE QUAD.'
    CUBOID         3         1
   11.1316    -11.5506    11.1316    -11.5506     0.2540    0.0
HOLE 132      -0.69215    -0.69215         0
    CUBOID         2         1
   11.3856   -11.8046    11.3856   -11.8046     0.2540        0.0
    CUBOID         3         1
 11.80465 -11.80465  11.80465 -11.80465     0.2540       0.0
UNIT 300  COM='FA #7,#8,#10 AND GS, SD, INNER FA(S),outer sleeve NW QUAD.'
    CUBOID         3         1
    11.5506     -11.1316    11.1316    -11.5506     0.2540    0.0
HOLE 132         0.69215    -0.69215         0
    CUBOID         2         1
   12.1221      -11.7031    11.7031    -12.1221     0.2540    0.0
    CUBOID         3         1
 12.12215 -12.12215  12.12215 -12.12215     0.2540       0.0
UNIT 400  COM='FA #9,#11,#12 AND GS, SD, OUTER FA(S), NW QUAD'
     CUBOID         3         1
    11.5506     -11.1316    11.1316    -11.5506     0.2540    0.0
HOLE 132         0.69215    -0.69215         0
    CUBOID         2         1
   11.8046      -11.3856    11.3856    -11.8046     0.2540    0.0
    CUBOID         3         1
 11.80465 -11.80465  11.80465 -11.80465     0.2540       0.0
UNIT 500  COM='FA #13,#14,#16 AND GS, SD, INNER FA(S),outer sleeve SW QUAD.'
    CUBOID         3         1
    11.5506     -11.1316    11.5506    -11.1316     0.2540    0.0
HOLE 132         0.69215     0.69215         0
    CUBOID         2         1
   12.1221      -11.7031    12.1221    -11.7031     0.2540    0.0
    CUBOID         3         1
 12.12215 -12.12215  12.12215 -12.12215     0.2540       0.0
UNIT 600  COM='FA #15,#17,#18 AND GS, SD, OUTER FA(S), SW QUAD'
    CUBOID         3         1
    11.5506     -11.1316    11.5506    -11.1316     0.2540    0.0
HOLE 132        0.69215     0.69215         0
    CUBOID         2         1
   11.8046      -11.3856    11.8046    -11.3856     0.2540    0.0
    CUBOID         3         1
 11.80465 -11.80465  11.80465 -11.80465     0.2540       0.0
UNIT 700  COM='FA #19,#20,#22 AND GS, SD, INNER FA(S),outer sleeve SE QUAD'
    CUBOID         3         1
    11.1316    -11.5506    11.5506    -11.1316     0.2540    0.0
HOLE 132      -0.69215    0.69215         0
    CUBOID         2         1
   11.7031   -12.1221    12.1221   -11.7031     0.2540        0.0
    CUBOID         3         1
 12.12215 -12.12215  12.12215 -12.12215     0.2540       0.0
UNIT 800  COM='FA #21,#23,#24 AND GS, SD, OUTER FA(S), SE QUAD'
    CUBOID         3         1
    11.1316    -11.5506    11.5506    -11.1316     0.2540    0.0
HOLE 132      -0.69215    0.69215         0
    CUBOID         2         1
   11.3856   -11.8046    11.8046   -11.3856     0.2540        0.0
    CUBOID         3         1
 11.80465 -11.80465  11.80465 -11.80465     0.2540       0.0
UNIT 9     COM='FA #1,#2,#4 AND GS, NCN FUEL outer sleeve'
    CUBOID         3         1
    11.1316    -11.5506    11.1316    -11.5506     0.2540    0.0
    CUBOID         2         1
   11.7031   -12.1221    11.7031   -12.1221     0.2540        0.0
    CUBOID         3         1
 12.12215 -12.12215  12.12215 -12.12215     0.2540       0.0
```

Unit 9 through Unit 16 are similar to units 1 through 8 with the exception that they model non-fuel rod positions within the DSC.

```
UNIT 10     COM='FA #3,#5,#6 AND GS, NON FUEL'
    CUBOID          3          1
    11.1316    -11.5506      11.1316     -11.5506      0.2540     0.0
    CUBOID          2          1
   11.3856    -11.8046    11.3856    -11.8046     0.2540         0.0
    CUBOID          3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540         0.0
UNIT 11     COM='FA #7,#8,#10 AND GS, NON FUEL outer sleeve'
    CUBOID          3          1
    11.5506     -11.1316     11.1316     -11.5506      0.2540     0.0
    CUBOID          2          1
   12.1221      -11.7031     11.7031    -12.1221      0.2540    0.0
    CUBOID          3          1
 12.12215 -12.12215  12.12215 -12.12215     0.2540         0.0
UNIT 12      COM='FA #9,#11,#12 AND GS, NON FUEL'
    CUBOID          3          1
    11.5506     -11.1316     11.1316     -11.5506      0.2540     0.0
    CUBOID          2          1
   11.8046      -11.3856     11.3856    -11.8046      0.2540    0.0
    CUBOID          3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540         0.0
UNIT 13     COM='FA #13,#14,#16 AND GS, NON FUEL outer sleeve'
    CUBOID          3          1
    11.5506     -11.1316     11.5506     -11.1316      0.2540     0.0
    CUBOID          2          1
   12.1221      -11.7031     12.1221     -11.7031      0.2540     0.0
    CUBOID          3          1
 12.12215 -12.12215  12.12215 -12.12215     0.2540         0.0
UNIT 14     COM='FA #15,#17,#18 AND GS, NON FUEL'
    CUBOID          3          1
    11.5506     -11.1316     11.5506     -11.1316      0.2540     0.0
    CUBOID          2          1
   11.8046      -11.3856     11.8046     -11.3856      0.2540     0.0
    CUBOID          3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540         0.0
UNIT 15     COM='FA #19,#20,#22 AND GS, NON FUEL outer sleeve'
    CUBOID          3          1
    11.1316    -11.5506      11.5506     -11.1316      0.2540     0.0
    CUBOID          2          1
   11.7031    -12.1221    12.1221    -11.7031     0.2540         0.0
    CUBOID          3          1
 12.12215 -12.12215  12.12215 -12.12215     0.2540         0.0
UNIT 16     COM='FA #21,#23,#24 AND GS, NON FUEL'
    CUBOID          3          1
    11.1316    -11.5506      11.5506     -11.1316      0.2540     0.0
    CUBOID          2          1
   11.3856    -11.8046    11.8046    -11.3856     0.2540         0.0
    CUBOID          3          1
 11.80465 -11.80465  11.80465 -11.80465     0.2540         0.0
```

Unit 29 models an active fuel rod slice 0.1" high. Unit 129 models a slice through the plenum region 0.1" high. Unit 30 and Unit 31 model the guide tube and instrument tube respectively.

```
UNIT 29     COM='FUEL ROD CELL, 0.2540 CM HIGH'
  CYLINDER          1          1
   0.4681      0.2540        0.0
  CYLINDER          5          1
   0.4788      0.2540        0.0
  CYLINDER          10          1
   0.54610      0.2540        0.0
    CUBOID          3          1
   0.72136    -0.72136    0.72136    -0.72136     0.2540         0.0
UNIT 129     COM='FUEL Rod Plenum CELL, 0.2540 CM HIGH'
  CYLINDER          5          1
   0.4788      0.2540        0.0
```

```
  CYLINDER        10         1
   0.54610     0.2540       0.0
    CUBOID         3         1
   0.72136   -0.72136    0.72136   -0.72136     0.2540       0.0
UNIT 30    COM='GUIDE TUBE CELL, 0.2540 CM HIGH'
  CYLINDER         3         1
   0.6325     0.2540       0.0
  CYLINDER        10         1
   0.67310     0.2540       0.0
    CUBOID         3         1
   0.72136   -0.72136    0.72136   -0.72136     0.2540       0.0
UNIT 31    COM='INSTRUMENT TUBE CELL, 0.2540 CM HIGH'
  CYLINDER         3         1
   0.5601     0.2540       0.0
  CYLINDER        10         1
   0.62611     0.2540       0.0
    CUBOID         3         1
   0.72136   -0.72136    0.72136   -0.72136     0.2540       0.0
```

Unit 32 contains Array 1, a 0.1" high slice of the active fuel region of the design basis 15x15 B&W fuel assembly.

```
UNIT 32  COM='15x15 Zr-4 CLAD FUEL ASSY, 0.2540 CM HIGH'
   ARRAY 1   -10.8204   -10.8204        0.0
```

Unit 132 contains Array 1, a 0.1" high slice of the plenum region of the design basis 15x15 B&W fuel assembly.

```
UNIT 132 COM='15x15 Zr-4 CLAD FUEL plenum region, 0.2540 CM HIGH'
   ARRAY 3   -10.8204   -10.8204        0.0
```

Unit 33 is representative of a slice of the cask at the spacer disk level in the active fuel region. This slice is only 0.1" high. All the holes modeled are representative of the cutouts in units 1-8. The cylinders represent the DSC thickness.

```
UNIT 33  COM='0.2540 CM HIGH SLICE THRU CASK AT SPACER DISK'
  CYLINDER         4         1
     83.185     0.2540       0.0
  HOLE 2      13.3731    13.3731        0.0
  HOLE 2      39.4843    13.3731        0.0
  HOLE 2      64.9605   13.3731       0.0
  HOLE 2      13.3731    39.4843        0.0
  HOLE 2      39.4843    39.4843        0.0
  HOLE 2      13.3731    64.9605        0.0
  HOLE 4     -13.3731    13.3731        0.0
  HOLE 4     -39.4843    13.3731        0.0
  HOLE 4     -64.9605    13.3731        0.0
  HOLE 4    -13.3731    39.4843       0.0
  HOLE 4    -39.4843    39.4843       0.0
  HOLE 4    -13.3731    64.9605       0.0
  HOLE 6    -13.3731   -13.3731       0.0
  HOLE 6    -39.4843   -13.3731       0.0
  HOLE 6    -64.9605   -13.3731       0.0
  HOLE 6    -13.3731   -39.4843       0.0
  HOLE 6    -39.4843   -39.4843       0.0
  HOLE 6    -13.3731   -64.9605       0.0
  HOLE 8     13.3731   -13.3731       0.0
  HOLE 8     39.4843   -13.3731       0.0
  HOLE 8     64.9605   -13.3731       0.0
  HOLE 8     13.3731   -39.4843       0.0
  HOLE 8     39.4843   -39.4843       0.0
  HOLE 8     13.3731   -64.9605       0.0
  CYLINDER         2         1
   85.3313     0.2540       0.0
    CUBOID         3         1
   100.0   -100.0    100.0   -100.0     0.2540       0.0
```

Unit 133 is representative of a slice of the cask at the spacer disk level in the plenum region. This slice is only 0.1" high. All the holes modeled are representative of the cutouts in units 100-800. The cylinders represent the DSC thickness.

```
UNIT 133  COM='0.2540 CM HIGH SLICE THRU CASK AT SPACER DISK'
  CYLINDER         4          1
     83.185     0.2540       0.0
  HOLE 200     13.3731   13.3731        0.0
  HOLE 200     39.4843   13.3731        0.0
  HOLE 200     64.9605  13.3731       0.0
  HOLE 200     13.3731   39.4843        0.0
  HOLE 200     39.4843   39.4843        0.0
  HOLE 200     13.3731   64.9605        0.0
  HOLE 400    -13.3731   13.3731        0.0
  HOLE 400    -39.4843   13.3731        0.0
  HOLE 400    -64.9605   13.3731        0.0
  HOLE 400   -13.3731   39.4843       0.0
  HOLE 400   -39.4843   39.4843       0.0
  HOLE 400   -13.3731   64.9605       0.0
  HOLE 600   -13.3731  -13.3731       0.0
  HOLE 600   -39.4843  -13.3731       0.0
  HOLE 600   -64.9605  -13.3731       0.0
  HOLE 600   -13.3731  -39.4843       0.0
  HOLE 600   -39.4843  -39.4843       0.0
  HOLE 600   -13.3731  -64.9605       0.0
  HOLE 800    13.3731  -13.3731       0.0
  HOLE 800    39.4843  -13.3731       0.0
  HOLE 800    64.9605  -13.3731       0.0
  HOLE 800    13.3731  -39.4843       0.0
  HOLE 800    39.4843  -39.4843       0.0
  HOLE 800    13.3731  -64.9605       0.0
  CYLINDER         2          1
   85.3313    0.2540        0.0
    CUBOID         3          1
   100.0   -100.0    100.0   -100.0      0.2540         0.0
```

Unit 34 is representative of a slice of the DSC through the space between the spacer disks in the active fuel region. This slice is only 0.1" high. All the holes modeled are representative of the cutouts in units 1-8. The cylinders represent the DSC thickness. Unit 77 is inserted as a hole four times to represent the support rods in the basket.

```
UNIT 34   COM='0.2540 CM HIGH SLICE THRU CASK AT MODERATOR'
  CYLINDER         3          1
     83.185     0.2540       0.0
  HOLE 2      13.3731   13.3731        0.0
  HOLE 2      39.4843   13.3731        0.0
  HOLE 2      64.9605  13.3731       0.0
  HOLE 2      13.3731   39.4843        0.0
  HOLE 2      39.4843   39.4843        0.0
  HOLE 2      13.3731   64.9605        0.0
  HOLE 4     -13.3731   13.3731        0.0
  HOLE 4     -39.4843   13.3731        0.0
  HOLE 4     -64.9605   13.3731        0.0
  HOLE 4    -13.3731   39.4843       0.0
  HOLE 4    -39.4843   39.4843       0.0
  HOLE 4    -13.3731   64.9605       0.0
  HOLE 6    -13.3731  -13.3731       0.0
  HOLE 6    -39.4843  -13.3731       0.0
  HOLE 6    -64.9605  -13.3731       0.0
  HOLE 6    -13.3731  -39.4843       0.0
  HOLE 6    -39.4843  -39.4843       0.0
  HOLE 6    -13.3731  -64.9605       0.0
  HOLE 8     13.3731  -13.3731       0.0
  HOLE 8     39.4843  -13.3731       0.0
  HOLE 8     64.9605  -13.3731       0.0
  HOLE 8     13.3731  -39.4843       0.0
  HOLE 8     39.4843  -39.4843       0.0
  HOLE 8     13.3731  -64.9605       0.0
  HOLE 77       60.833    34.7218        0.0
```

```
 HOLE 77        -60.833   -34.7218        0.0
 HOLE 77         60.833   -34.7218        0.0
 HOLE 77        -60.833    34.7218        0.0
 CYLINDER           2          1
  85.3313      0.2540        0.0
   CUBOID           3          1
  100.0    -100.0    100.0   -100.0     0.2540         0.0
```

Unit 134 is representative of a slice of the DSC through the space between the spacer disks in the plenum region. This slice is only 0.1" high. All the holes modeled are representative of the cutouts in units 100-800. The cylinders represent the DSC thickness. Unit 77 is inserted as a hole four times to represent the support rods in the basket.

```
 UNIT 134 COM='0.2540 CM HIGH SLICE THRU CASK AT MODERATOR'
 CYLINDER          3          1
    83.185     0.2540        0.0
 HOLE 200      13.3731    13.3731        0.0
 HOLE 200      39.4843    13.3731        0.0
 HOLE 200      64.9605   13.3731       0.0
 HOLE 200      13.3731    39.4843        0.0
 HOLE 200      39.4843    39.4843        0.0
 HOLE 200      13.3731    64.9605        0.0
 HOLE 400     -13.3731    13.3731        0.0
 HOLE 400     -39.4843    13.3731        0.0
 HOLE 400     -64.9605    13.3731        0.0
 HOLE 400    -13.3731   39.4843        0.0
 HOLE 400    -39.4843   39.4843        0.0
 HOLE 400    -13.3731   64.9605        0.0
 HOLE 600    -13.3731  -13.3731        0.0
 HOLE 600    -39.4843  -13.3731        0.0
 HOLE 600    -64.9605  -13.3731        0.0
 HOLE 600    -13.3731  -39.4843        0.0
 HOLE 600    -39.4843  -39.4843        0.0
 HOLE 600    -13.3731  -64.9605        0.0
 HOLE 800     13.3731  -13.3731        0.0
 HOLE 800     39.4843  -13.3731        0.0
 HOLE 800     64.9605  -13.3731        0.0
 HOLE 800     13.3731  -39.4843        0.0
 HOLE 800     39.4843  -39.4843        0.0
 HOLE 800     13.3731  -64.9605        0.0
 HOLE 77         60.833    34.7218        0.0
 HOLE 77        -60.833   -34.7218        0.0
 HOLE 77         60.833   -34.7218        0.0
 HOLE 77        -60.833    34.7218        0.0
 CYLINDER           2          1
  85.3313      0.2540        0.0
   CUBOID           3          1
  100.0    -100.0    100.0   -100.0     0.2540         0.0
UNIT 36     COM='BOTTOM SLICE OF DSC (-8.67-0.0")'
  CYLINDER           2          1
  85.3313     22.0218        0.0
   CUBOID           3          1
  100.0    -100.0    100.0   -100.0     22.0218         0.0
```

Unit 37 is representative of a slice of the DSC through the space between the spacer disks in the location of the over sleeve in the active fuel region. This slice is only 0.1" high. All the holes modeled are representative of the cutouts in units 100-800. The cylinders represent the DSC thickness. Unit 77 is inserted as a hole four times to represent the support rods in the basket.

```
UNIT 37  COM='0.2540 CM HIGH SLICE THRU CASK AT MODERATOR WITH OVER SLEEVE'
 CYLINDER          3          1
    83.185     0.2540        0.0
 HOLE 1      13.3731    13.3731        0.0
 HOLE 1      39.4843    13.3731        0.0
 HOLE 2      64.9605   13.3731       0.0
 HOLE 1      13.3731    39.4843        0.0
 HOLE 2      39.4843    39.4843        0.0
 HOLE 2      13.3731    64.9605        0.0
```

```
HOLE 3    -13.3731   13.3731    0.0
HOLE 3    -39.4843   13.3731    0.0
HOLE 4    -64.9605   13.3731    0.0
HOLE 3   -13.3731   39.4843    0.0
HOLE 4   -39.4843   39.4843    0.0
HOLE 4   -13.3731   64.9605    0.0
HOLE 5   -13.3731  -13.3731    0.0
HOLE 5   -39.4843  -13.3731    0.0
HOLE 6   -64.9605  -13.3731    0.0
HOLE 5   -13.3731  -39.4843    0.0
HOLE 6   -39.4843  -39.4843    0.0
HOLE 6   -13.3731  -64.9605    0.0
HOLE 7    13.3731  -13.3731    0.0
HOLE 7    39.4843  -13.3731    0.0
HOLE 8    64.9605  -13.3731    0.0
HOLE 7    13.3731  -39.4843    0.0
HOLE 8    39.4843  -39.4843    0.0
HOLE 8    13.3731  -64.9605    0.0
HOLE 77       60.833    34.7218     0.0
HOLE 77      -60.833   -34.7218     0.0
HOLE 77       60.833   -34.7218     0.0
HOLE 77      -60.833    34.7218     0.0
CYLINDER        2          1
 85.3313     0.2540       0.0
  CUBOID        3          1
 100.0  -100.0    100.0   -100.0     0.2540        0.0
```

Unit 137 is representative of a slice of the DSC through the space between the spacer disks in the location of the over sleeve in the fuel rod plenum region. This slice is only 0.1" high. All the holes modeled are representative of the cutouts in units 100-800. The cylinders represent the DSC thickness. Unit 77 is inserted as a hole four times to represent the support rods in the basket.

```
UNIT 137 COM='0.2540 CM HIGH SLICE THRU CASK AT MODERATOR WITH OVER SLEEVE'
 CYLINDER          3          1
   83.185      0.2540        0.0
 HOLE 100     13.3731    13.3731      0.0
 HOLE 100     39.4843    13.3731      0.0
 HOLE 200     64.9605   13.3731     0.0
 HOLE 100     13.3731    39.4843      0.0
 HOLE 200     39.4843    39.4843      0.0
 HOLE 200     13.3731    64.9605      0.0
 HOLE 300    -13.3731    13.3731      0.0
 HOLE 300    -39.4843    13.3731      0.0
 HOLE 400    -64.9605    13.3731      0.0
 HOLE 300   -13.3731   39.4843     0.0
 HOLE 400   -39.4843   39.4843     0.0
 HOLE 400   -13.3731   64.9605     0.0
 HOLE 500   -13.3731  -13.3731     0.0
 HOLE 500   -39.4843  -13.3731     0.0
 HOLE 600   -64.9605  -13.3731     0.0
 HOLE 500   -13.3731  -39.4843     0.0
 HOLE 600   -39.4843  -39.4843     0.0
 HOLE 600   -13.3731  -64.9605     0.0
 HOLE 700    13.3731  -13.3731     0.0
 HOLE 700    39.4843  -13.3731     0.0
 HOLE 800    64.9605  -13.3731     0.0
 HOLE 700    13.3731  -39.4843     0.0
 HOLE 800    39.4843  -39.4843     0.0
 HOLE 800    13.3731  -64.9605     0.0
 HOLE 77       60.833    34.7218      0.0
 HOLE 77      -60.833   -34.7218      0.0
 HOLE 77       60.833   -34.7218      0.0
 HOLE 77      -60.833    34.7218      0.0
 CYLINDER        2          1
  85.3313     0.2540       0.0
   CUBOID        3          1
  100.0  -100.0    100.0   -100.0     0.2540        0.0
```

The following units model non-fuel rod regions and include moderator and disk spacer regions.

```
UNIT 52    COM='0.2540 CM SLICE THRU CASK, SLEEVE , (0.1")'
  CYLINDER        3         1
     83.185     0.2540        0.0
  HOLE 9      13.3731   13.3731        0.0
  HOLE 9      39.4843   13.3731        0.0
  HOLE 10     64.9605   13.3731        0.0
  HOLE 9      13.3731   39.4843        0.0
  HOLE 10     39.4843   39.4843        0.0
  HOLE 10     13.3731   64.9605        0.0
  HOLE 11    -13.3731   13.3731        0.0
  HOLE 11    -39.4843   13.3731        0.0
  HOLE 12    -64.9605   13.3731        0.0
  HOLE 11    -13.3731   39.4843        0.0
  HOLE 12    -39.4843   39.4843        0.0
  HOLE 12    -13.3731   64.9605        0.0
  HOLE 13    -13.3731  -13.3731        0.0
  HOLE 13    -39.4843  -13.3731        0.0
  HOLE 14    -64.9605  -13.3731        0.0
  HOLE 13    -13.3731  -39.4843        0.0
  HOLE 14    -39.4843  -39.4843        0.0
  HOLE 14    -13.3731  -64.9605        0.0
  HOLE 15     13.3731  -13.3731        0.0
  HOLE 15     39.4843  -13.3731        0.0
  HOLE 16     64.9605  -13.3731        0.0
  HOLE 15     13.3731  -39.4843        0.0
  HOLE 16     39.4843  -39.4843        0.0
  HOLE 16     13.3731  -64.9605        0.0
  HOLE 77        60.833    34.7218        0.0
  HOLE 77       -60.833   -34.7218        0.0
  HOLE 77        60.833   -34.7218        0.0
  HOLE 77       -60.833    34.7218        0.0
  CYLINDER        2         1
   85.3313     0.2540       0.0
    CUBOID        3         1
   100.0   -100.0    100.0   -100.0      0.2540        0.0
UNIT 53    COM='0.2540CM SLICE THRU CASK, SD, AND SLEEVE , (0.1")'
  CYLINDER        4         1
     83.185     0.2540        0.0
  HOLE 10      13.3731   13.3731        0.0
  HOLE 10      39.4843   13.3731        0.0
  HOLE 10     64.9605   13.3731        0.0
  HOLE 10      13.3731   39.4843        0.0
  HOLE 10     39.4843   39.4843        0.0
  HOLE 10     13.3731   64.9605        0.0
  HOLE 11    -13.3731   13.3731        0.0
  HOLE 11    -39.4843   13.3731        0.0
  HOLE 12    -64.9605   13.3731        0.0
  HOLE 11    -13.3731   39.4843        0.0
  HOLE 12    -39.4843   39.4843        0.0
  HOLE 12    -13.3731   64.9605        0.0
  HOLE 13    -13.3731  -13.3731        0.0
  HOLE 13    -39.4843  -13.3731        0.0
  HOLE 14    -64.9605  -13.3731        0.0
  HOLE 13    -13.3731  -39.4843        0.0
  HOLE 14    -39.4843  -39.4843        0.0
  HOLE 14    -13.3731  -64.9605        0.0
  HOLE 15     13.3731  -13.3731        0.0
  HOLE 15     39.4843  -13.3731        0.0
  HOLE 16     64.9605  -13.3731        0.0
  HOLE 15     13.3731  -39.4843        0.0
  HOLE 16     39.4843  -39.4843        0.0
  HOLE 16     13.3731  -64.9605        0.0
  CYLINDER        2         1
   85.3313     0.2540       0.0
    CUBOID        3         1
   100.0   -100.0    100.0   -100.0      0.2540        0.0
UNIT 70    COM='10.922 CM SLICE THRU CASK, SLEEVE (Flared), (4.3")'
  CYLINDER        3         1
```

```
       83.185      10.922        0.0
    HOLE 9       13.3731    13.3731       0.0
    HOLE 9       39.4843    13.3731       0.0
    HOLE 10      64.9605    13.3731       0.0
    HOLE 9       13.3731    39.4843       0.0
    HOLE 10      39.4843    39.4843       0.0
    HOLE 10      13.3731    64.9605       0.0
    HOLE 11     -13.3731    13.3731       0.0
    HOLE 11     -39.4843    13.3731       0.0
    HOLE 12     -64.9605    13.3731       0.0
    HOLE 11     -13.3731    39.4843       0.0
    HOLE 12     -39.4843    39.4843       0.0
    HOLE 12     -13.3731    64.9605       0.0
    HOLE 13     -13.3731   -13.3731       0.0
    HOLE 13     -39.4843   -13.3731       0.0
    HOLE 14     -64.9605   -13.3731       0.0
    HOLE 13     -13.3731   -39.4843       0.0
    HOLE 14     -39.4843   -39.4843       0.0
    HOLE 14     -13.3731   -64.9605       0.0
    HOLE 15      13.3731   -13.3731       0.0
    HOLE 15      39.4843   -13.3731       0.0
    HOLE 16      64.9605   -13.3731       0.0
    HOLE 15      13.3731   -39.4843       0.0
    HOLE 16      39.4843   -39.4843       0.0
    HOLE 16      13.3731   -64.9605       0.0
    HOLE 78        60.833    34.7218       0.0
    HOLE 78       -60.833   -34.7218       0.0
    HOLE 78        60.833   -34.7218       0.0
    HOLE 78       -60.833    34.7218       0.0
    CYLINDER          2          1
     85.3313     10.922        0.0
      CUBOID          3          1
     100.0   -100.0    100.0   -100.0     10.922      0.0
  UNIT 73    COM='11.5 CM HIGH SLICE THRU MOD, NON FUEL OR SLEEVE'
    CYLINDER          3          1
       83.185      11.506        0.0
    HOLE 79       60.833    34.7218       0.0
    HOLE 79      -60.833   -34.7218       0.0
    HOLE 79       60.833   -34.7218       0.0
    HOLE 79      -60.833    34.7218       0.0
    CYLINDER          2          1
     85.3313     11.506        0.0
      CUBOID          3          1
     100.0   -100.0    100.0   -100.0     11.506       0.0
  UNIT 76    COM='TOP of DSC (25.5 cm)'
    CYLINDER          4          1
       83.185      25.5016       0.0
    CYLINDER          2          1
     85.3313     25.5016       0.0
      CUBOID          3          1
     100.0   -100.0    100.0   -100.0     25.5016      0.0
  UNIT 77    COM='SUPPORT ROD, 0.635 CM HIGH'
    CYLINDER          4          1
        3.81     0.2540        0.0
  UNIT 78    COM='SUPPORT ROD, 10.922 CM HIGH'
    CYLINDER          4          1
        3.81     10.922       0.0
  UNIT 79    COM='SUPPORT ROD, 10.16 CM HIGH'
    CYLINDER          4          1
        3.81     10.16      0.0
```

Unit 90 is the global unit that contains the global array of 1586 units stacked to form the finite model of the DSC.

```
  GLOBAL UNIT 90
     ARRAY 2   -131.445   -131.445        0
END GEOM
READ ARRAY
```

Array 1 is a 15x15 array representative of the active fuel region of the design basis fuel assembly. Array 2 is a stack of 1586 slices constituting the KENO model. Array 3 is a 15x15 array representative of the plenum region of the fuel rods.

```
   COM='15x15 Zr-4 CLAD FUEL ASSEMBLY SLICE, FUEL REGIONS'
   ARA=1       NUX=15     NUY=15     NUZ=1
   FILL
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
       29  29  29  29  29  30  29  29  29  30  29  29  29  29  29
       29  29  29  30  29  29  29  29  29  29  29  30  29  29  29
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
       29  29  30  29  29  30  29  29  29  30  29  29  30  29  29
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
       29  29  29  29  29  29  29  31  29  29  29  29  29  29  29
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
       29  29  30  29  29  30  29  29  29  30  29  29  30  29  29
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
       29  29  29  30  29  29  29  29  29  29  29  30  29  29  29
       29  29  29  29  29  30  29  29  29  30  29  29  29  29  29
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
       29  29  29  29  29  29  29  29  29  29  29  29  29  29  29
   END FILL
   COM='STACK 0.1" DISKS + NON UNIFORM REG. TO APPROXIMATE DSC BASKET'
   ARA=2       NUX=1      NUY=1      NUZ=1586
   FILL  36 55R34 20R33 206R37 20R33 191R34 20R33 191R34 20R33
         191R34 20R33 191R34 20R33 191R34 20R33 67R37 87R137 52R52
         20R53 70 73 76
   END FILL
   COM='15x15 Zr-4 CLAD FUEL ASSEMBLY SLICE, plenum REGIONS'
   ARA=3       NUX=15     NUY=15     NUZ=1
   FILL
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
       129 129 129 129 129 30  129 129 129 30  129 129 129 129 129
       129 129 129 30  129 129 129 129 129 129 129 30  129 129 129
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
       129 129 30  129 129 30  129 129 129 30  129 129 30  129 129
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
       129 129 129 129 129 129 129 31  129 129 129 129 129 129 129
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
       129 129 30  129 129 30  129 129 129 30  129 129 30  129 129
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
       129 129 129 30  129 129 129 129 129 129 129 30  129 129 129
       129 129 129 129 129 30  129 129 129 30  129 129 129 129 129
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
       129 129 129 129 129 129 129 129 129 129 129 129 129 129 129
   END FILL
END ARRAY
READ START
   NST=1
END START
```

The model's boundary condition is specified as water albedo on surfaces.

```
READ BOUNDS
   all=water
END BOUNDS
READ PLOT
   TTL='CASK MATERIAL PLOT  - PLAN VIEW - bottom'
   PIC=MAT
   XUL=-131.4   YUL=131.4    ZUL=66.04
   XLR=131.4    YLR=-131.4   ZLR=66.04
   UAX=1.0    VDN=-1.0
   NAX=1000 LPI=10.0 END
   TTL='CASK MATERIAL PLOT  - FA#1'
   PIC=MAT
   XUL=0.0    YUL=28.0    ZUL=66.04
   XLR=28.0    YLR=0.0   ZLR=66.04
```

```
  UAX=1.0    VDN=-1.0
  NAX=1000 LPI=10 END
  TTL='CASK MATERIAL PLOT  - FA#3'
  PIC=MAT
  XUL=52.0    YUL=28.0    ZUL=66.04
  XLR=80.0    YLR=0.0     ZLR=66.04
  UAX=1.0    VDN=-1.0
  NAX=1000 LPI=10 END
  SCR=YES
  TTL='CASK MATERIAL PLOT  - PLAN VIEW - top'
  PIC=MAT
  XUL=-131.4   YUL=131.4    ZUL=372.5
  XLR=131.4    YLR=-131.4   ZLR=372.5
  UAX=1.0    VDN=-1.0
  NAX=1000 LPI=10.0 END
  TTL='CASK MATERIAL PLOT  - FA#5'
  PIC=MAT
  XUL=0.0    YUL=28.0    ZUL=372.5
  XLR=28.0    YLR=0.0   ZLR=372.5
  UAX=1.0    VDN=-1.0
  NAX=1000 LPI=10 END
  TTL='CASK MATERIAL PLOT  - FA#6'
  PIC=MAT
  XUL=52.0    YUL=28.0    ZUL=372.5
  XLR=80.0    YLR=0.0     ZLR=372.5
  UAX=1.0    VDN=-1.0
  NAX=1000 LPI=10 END
  SCR=YES
END PLOT
END DATA
END
```

Table 3.6-1
NUHOMS®-24P DSC Dimensional Data (Worst Case Tolerances)

	inches	cm
Guide sleeve inside dimension	8.93	22.6822
Guide sleeve thickness	0.1	0.2540
Guide sleeve outside dimension	9.13	23.1902
Over sleeve thickness	0.125	0.3175
Over sleeve outside dimension	9.38	23.8252
Spacer disk hole inside dimension	9.295 square	23.6093
Minimum Ligaments (Location shown on Figure 3.6-3)	1.21/0.96/0.71	3.0734/2.4384/1.8034

Table 3.6-2
Criticality Results (with Water in Fuel-Cladding Gap and Without BPRA)

Moderator Density, g/cc	$k_{calculated}$	1σ	k_{eff}	USL
0.9982	0.8317	0.0010	0.8337	0.9410
0.9000	0.8499	0.0010	0.8519	0.9410
0.8000	0.8668	0.0008	0.8684	0.9410
0.7000	0.8883	0.0010	0.8903	0.9410
0.6000	0.9084	0.0010	0.9104	0.9410
0.5000	0.9280	0.0010	0.9300	0.9410
0.4000	0.9339	0.0009	0.9357	0.9410
0.3000	0.9304	0.0010	0.9324	0.9410
0.2000	0.8880	0.0010	0.8900	0.9410
0.1000	0.7703	0.0009	0.7721	0.9410
0.0001	0.4526	0.0007	0.4540	0.9410

Table 3.6-3
Criticality Results (Water in Fuel-Cladding gap With BPRA)

Moderator Density, g/cc	$k_{calculated}$	1σ	k_{eff}	USL
0.9982	0.8384	0.0008	0.8400	0.9410
0.9000	0.8512	0.0008	0.8528	0.9410
0.8000	0.8698	0.0010	0.8718	0.9410
0.7000	0.8879	0.0009	0.8897	0.9410
0.6000	0.9068	0.0009	0.9086	0.9410
0.5000	0.9231	0.0009	0.9249	0.9410
0.4000	0.9320	0.0009	0.9338	0.9410
0.3000	0.9271	0.0011	0.9293	0.9410
0.2000	0.8907	0.0009	0.8925	0.9410
0.1000	0.7779	0.0009	0.7797	0.9410
0.0001	0.4792	0.0006	0.4804	0.9410

Table 3.6-4
Criticality Results (Void in Fuel-Cladding Gap and Without BPRA)

Moderator Density, g/cc	$k_{calculated}$	1σ	k_{eff}	USL
0.9982	0.8337	0.0009	0.8355	0.9410
0.9000	0.8509	0.0010	0.8529	0.9410
0.8000	0.8691	0.0009	0.8709	0.9410
0.7000	0.8871	0.0009	0.8889	0.9410
0.6000	0.9057	0.0009	0.9075	0.9410
0.5000	0.9188	0.0010	0.9208	0.9410
0.4000	0.9276	0.0009	0.9294	0.9410
0.3000	0.9172	0.0009	0.9190	0.9410
0.2000	0.8705	0.0012	0.8729	0.9410
0.1000	0.7357	0.0008	0.7373	0.9410
0.0001	0.3754	0.0005	0.3764	0.9410

Table 3.6-5
Criticality Results (Void in Fuel-Cladding Gap and With BPRA)

Moderator Density, g/cc	$k_{calculated}$	1σ	k_{eff}	USL
0.9982	0.8336	0.0011	0.8358	0.9410
0.9000	0.8485	0.0011	0.8507	0.9410
0.8000	0.8648	0.0010	0.8668	0.9410
0.7000	0.8821	0.0009	0.8839	0.9410
0.6000	0.9042	0.0010	0.9062	0.9410
0.5000	0.9180	0.0010	0.9200	0.9410
0.4000	0.9252	0.0010	0.9272	0.9410
0.3000	0.9135	0.0009	0.9153	0.9410
0.2000	0.8685	0.0010	0.8705	0.9410
0.1000	0.7415	0.0009	0.7433	0.9410
0.0001	0.4047	0.0006	0.4059	0.9410

Table 3.6-6
Benchmarking Results

Run ID	U Enrich. Wt%	Pu Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1σ
B1645SO1	2.46		1.41	1.015		32.8194	0.9967	0.0009
B1645SO2	2.46		1.41	1.015		32.7584	1.0002	0.0011
BW1231B1	4.02		1.511	1.139		31.1427	0.9966	0.0012
BW1231B2	4.02		1.511	1.139		29.8854	0.9972	0.0009
BW1273M	2.46		1.511	1.376		32.2106	0.9965	0.0009
BW1484A1	2.46		1.636	1.841	1.636	34.5304	0.9962	0.0010
BW1484A2	2.46		1.636	1.841	4.908	35.1629	0.9931	0.0010
BW1484B1	2.46		1.636	1.841		33.9421	0.9979	0.0010
BW1484B2	2.46		1.636	1.841	1.636	34.5820	0.9955	0.0012
BW1484B3	2.46		1.636	1.841	4.908	35.2609	0.9969	0.0011
BW1484C1	2.46		1.636	1.841	1.636	34.6463	0.9931	0.0011
BW1484C2	2.46		1.636	1.841	4.908	35.2422	0.9939	0.0012
BW1484S1	2.46		1.636	1.841	1.636	34.5105	1.0001	0.0010
BW1484S2	2.46		1.636	1.841	1.636	34.5569	0.9992	0.0010
BW1484SL	2.46		1.636	1.841	6.544	35.4151	0.9935	0.0011
BW1645S1	2.46		1.209	0.383	1.778	30.1040	0.9990	0.0010
BW1645S2	2.46		1.209	0.383	1.778	29.9961	1.0037	0.0011
BW1810A	2.46		1.636	1.841		33.9465	0.9984	0.0008
BW1810B	2.46		1.636	1.841		33.9631	0.9984	0.0009
BW1810C	2.46		1.636	1.841		33.1569	0.9992	0.0010
BW1810D	2.46		1.636	1.841		33.0821	0.9985	0.0013
BW1810E	2.46		1.636	1.841		33.1600	0.9988	0.0009
BW1810F	2.46		1.636	1.841		33.9556	1.0031	0.0011
BW1810G	2.46		1.636	1.841		32.9409	0.9973	0.0011
BW1810H	2.46		1.636	1.841		32.9420	0.9972	0.0011
BW1810I	2.46		1.636	1.841		33.9655	1.0037	0.0009
BW1810J	2.46		1.636	1.841		33.1403	0.9983	0.0011
DSN399-1	4.74		1.6	3.807	1.8	33.9520	1.0036	0.0015
DSN399-2	4.74		1.6	3.807	5.8	34.4207	0.9989	0.0016
DSN399-3	4.74		1.6	3.807		35.3140	1.0024	0.0015
DSN399-4	4.74		1.6	3.807		35.3784	0.9977	0.0013
EPRU65	2.35		1.562	1.196		33.9106	0.9960	0.0011
EPRU65B	2.35		1.562	1.196		33.4013	0.9993	0.0012
EPRU75	2.35		1.905	2.408		35.8671	0.9958	0.0010
EPRU75B	2.35		1.905	2.408		35.3043	0.9996	0.0010
EPRU87	2.35		2.21	3.687		36.6129	1.0007	0.0011
EPRU87B	2.35		2.21	3.687		36.3499	1.0007	0.0011
NSE71SQ	4.74		1.26	1.823		33.7610	0.9979	0.0012
NSE71W1	4.74		1.26	1.823		34.0129	0.9988	0.0013
NSE71W2	4.74		1.26	1.823		36.3037	0.9957	0.0010
P2438BA	2.35		2.032	2.918	5.05	36.2277	0.9979	0.0013
P2438SLG	2.35		2.032	2.918	8.39	36.2889	0.9986	0.0012
P2438SS	2.35		2.032	2.918	6.88	36.2705	0.9974	0.0011
P2438ZR	2.35		2.032	2.918	8.79	36.2840	0.9987	0.0010
P2615BA	4.31		2.54	3.883	6.72	35.7286	1.0019	0.0014
P2615SS	4.31		2.54	3.883	8.58	35.7495	0.9952	0.0015
P2615ZR	4.31		2.54	3.883	10.92	35.7700	0.9977	0.0014
P2827L1	2.35		2.032	2.918	13.27	36.2526	1.0057	0.0011
P2827L2	2.35		2.032	2.918	11.25	36.2908	0.9999	0.0012

Table 3.6-6
Benchmarking Results, continuing

Run ID	U Enrich. Wt%	Pu Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1σ
P2827L3	4.31		2.54	3.883	20.78	35.6766	1.0092	0.0012
P2827L4	4.31		2.54	3.883	19.04	35.7131	1.0073	0.0012
P2827SLG	2.35		2.032	2.918	8.31	36.3037	0.9957	0.0010
P3314BA	4.31		1.892	1.6	2.83	33.1881	0.9988	0.0012
P3314BC	4.31		1.892	1.6	2.83	33.2284	0.9992	0.0012
P3314BF1	4.31		1.892	1.6	2.83	33.2505	1.0037	0.0013
P3314BF2	4.31		1.892	1.6	2.83	33.2184	1.0009	0.0013
P3314BS1	2.35		1.684	1.6	3.86	34.8594	0.9956	0.0013
P3314BS2	2.35		1.684	1.6	3.46	34.8356	0.9949	0.0010
P3314BS3	4.31		1.892	1.6	7.23	33.4247	0.9970	0.0013
P3314BS4	4.31		1.892	1.6	6.63	33.4162	0.9998	0.0012
P3314SLG	4.31		1.892	1.6	2.83	34.0198	0.9974	0.0012
P3314SS1	4.31		1.892	1.6	2.83	33.9601	0.9999	0.0012
P3314SS2	4.31		1.892	1.6	2.83	33.7755	1.0022	0.0012
P3314SS3	4.31		1.892	1.6	2.83	33.8904	0.9992	0.0013
P3314SS4	4.31		1.892	1.6	2.83	33.7625	0.9958	0.0011
P3314SS5	2.35		1.684	1.6	7.8	34.9531	0.9949	0.0013
P3314SS6	4.31		1.892	1.6	10.52	33.5333	1.0020	0.0011
P3314W1	4.31		1.892	1.6		34.3994	1.0024	0.0013
P3314W2	2.35		1.684	1.6		35.2167	0.9969	0.0011
P3314ZR	4.31		1.892	1.6	2.83	33.9954	0.9971	0.0013
P3602BB	4.31		1.892	1.6	8.3	33.3221	1.0029	0.0013
P3602BS1	2.35		1.684	1.6	4.8	34.7750	1.0027	0.0012
P3602BS2	4.31		1.892	1.6	9.83	33.3679	1.0039	0.0012
P3602N11	2.35		1.684	1.6	8.98	34.7438	1.0023	0.0012
P3602N12	2.35		1.684	1.6	9.58	34.8391	1.0030	0.0012
P3602N13	2.35		1.684	1.6	9.66	34.9337	1.0013	0.0012
P3602N14	2.35		1.684	1.6	8.54	35.0282	0.9974	0.0013
P3602N21	2.35		2.032	2.918	11.2	36.2821	0.9987	0.0011
P3602N22	2.35		2.032	2.918	10.36	36.1896	1.0025	0.0011
P3602N31	4.31		1.892	1.6	14.87	33.2094	1.0057	0.0013
P3602N32	4.31		1.892	1.6	15.74	33.3067	1.0093	0.0012
P3602N33	4.31		1.892	1.6	15.87	33.4174	1.0107	0.0012
P3602N34	4.31		1.892	1.6	15.84	33.4683	1.0045	0.0013
P3602N35	4.31		1.892	1.6	15.45	33.5185	1.0013	0.0012
P3602N36	4.31		1.892	1.6	13.82	33.5855	1.0004	0.0014
P3602N41	4.31		2.54	3.883	12.89	35.5276	1.0109	0.0013
P3602N42	4.31		2.54	3.883	14.12	35.6695	1.0071	0.0014
P3602N43	4.31		2.54	3.883	12.44	35.7542	1.0053	0.0015
P3602SS1	2.35		1.684	1.6	8.28	34.8701	1.0025	0.0013
P3602SS2	4.31		1.892	1.6	13.75	33.4202	1.0035	0.0012
P3926L1	2.35		1.684	1.6	10.06	34.8519	1.0000	0.0011
P3926L2	2.35		1.684	1.6	10.11	34.9324	1.0017	0.0011
P3926L3	2.35		1.684	1.6	8.5	35.0641	0.9949	0.0012
P3926L4	4.31		1.892	1.6	17.74	33.3243	1.0074	0.0014
P3926L5	4.31		1.892	1.6	18.18	33.4074	1.0057	0.0013
P3926L6	4.31		1.892	1.6	17.43	33.5246	1.0046	0.0013
P3926SL1	2.35		1.684	1.6	6.59	33.4737	0.9995	0.0012
P3926SL2	4.31		1.892	1.6	12.79	33.5776	1.0007	0.0012

Table 3.6-6
Benchmarking Results, continuing

Run ID	U Enrich. Wt%	Pu Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1σ
P4267B1	4.31		1.8901	1.59		31.8075	0.9990	0.0010
P4267B2	4.31		0.89	1.59		31.5323	1.0033	0.0010
P4267B3	4.31		1.715	1.09		30.9905	1.0050	0.0011
P4267B4	4.31		1.715	1.09		30.5061	0.9996	0.0011
P4267B5	4.31		1.715	1.09		30.1011	1.0004	0.0011
P4267SL1	4.31		1.89	1.59		33.4737	0.9995	0.0012
P4267SL2	4.31		1.715	1.09		31.9460	0.9988	0.0016
P62FT231	4.31		1.891	1.6	5.19	32.9196	1.0012	0.0013
P71F14F3	4.31		1.891	1.6	5.19	32.8237	1.0009	0.0014
P71F14V3	4.31		1.891	1.6	5.19	32.8597	0.9972	0.0014
P71F14V5	4.31		1.891	1.6	5.19	32.8609	0.9993	0.0013
P71F214R	4.31		1.891	1.6	5.19	32.8778	0.9969	0.0012
PAT80L1	4.74		1.6	3.807	4.9	35.0253	1.0012	0.0012
PAT80L2	4.74		1.6	3.807	4.9	35.1136	0.9993	0.0015
PAT80SS1	4.74		1.6	3.807	4.9	35.0045	0.9988	0.0013
PAT80SS2	4.74		1.6	3.807	4.9	35.1072	0.9960	0.0013
W3269A	5.7		1.422	1.93		33.1480	0.9988	0.0012
W3269B1	3.7		1.105	1.432		32.4055	0.9961	0.0011
W3269B2	3.7		1.105	1.432		32.3921	0.9963	0.0011
W3269B3	3.7		1.105	1.432		32.2363	0.9944	0.0011
W3269C	2.72		1.524	1.494		33.7727	0.9989	0.0012
W3269SL1	2.72		1.524	1.494		33.3850	0.9981	0.0014
W3269SL2	5.7		1.422	1.93		33.0910	1.0005	0.0013
W3269W1	2.72		1.524	1.494		33.5114	0.9966	0.0014
W3269W2	5.7		1.422	1.93		33.1680	1.0014	0.0014
W3385SL1	5.74		1.422	1.932		33.2387	1.0009	0.0012
W3385SL2	5.74		2.012	5.067		35.8818	0.9997	0.0013
EPRI70UN	0.71	2	1.778	1.2		31.6775	0.9983	0.0012
EPRI70B	0.71	2	1.778	1.2		30.9021	1.0009	0.0012
EPRI87UN	0.71	2	2.2098	2.53		33.3230	1.0096	0.0011
EPRI87B	0.71	2	2.2098	2.53		31.6775	0.9983	0.0012
EPRI99UN	0.71	2	2.5146	3.64		35.1817	1.0063	0.0011
EPRI99B	0.71	2	2.5146	3.64		34.4098	1.0095	0.0011
SAXTON52	0.71	6.6	1.3208	1.68		30.2980	1.0020	0.0014
SAXTON56	0.71	6.6	1.4224	2.16		31.4724	1.0010	0.0014
SAXTON56B	0.71	6.6	1.4224	2.16		31.0038	0.9994	0.0013
SAXTN735	0.71	6.6	1.8669	4.7		34.1848	1.0007	0.0016
SATN792	0.71	6.6	2.01168	5.67		34.6401	1.0026	0.0013
SAXTN104	0.71	6.6	2.6416	10.75		35.8333	1.0054	0.0014
Correlation	0.31	-0.26	0.43	0.25	0.65	-0.01	N/A	N/A

Table 3.6-7
USL-1 Results

Parameter	Range of applicability	USL-1
ent (wt. % U-235)	2.4 2.8 3.3 3.8 – 5.7	0.9424 0.9430 0.9435 0.9438
Fuel Rod Pitch (cm)	0.89 1.1 1.4 1.6 1.9 – 2.6	0.9396 0.9408 0.9421 0.9433 0.9439
Water/Fuel Volume Ratio	0.38 1.9 3.3 – 11	0.9414 0.9425 0.9426
Assembly Separation (cm)	1.6 4.4 7.1 9.8 – 21	0.9410 0.9425 0.9440 0.9441
nergy Group Causing Fission (AEG)	30 – 37	0.9433

Table 3.6-8
USL Determination for Criticality Analysis

Parameter	Value from Limiting Analysis	Bounding USL-1
U Enrichment (wt. % U-235)	4.0	0.9438
Fuel Rod Pitch (cm)	1.4427	0.9421
Water/Fuel Ratio	1.663	0.9414
Assembly Separation (cm)	1.80 – 3.07	0.9410
Average Energy Group Causing Fission (AEG)	33	0.9444

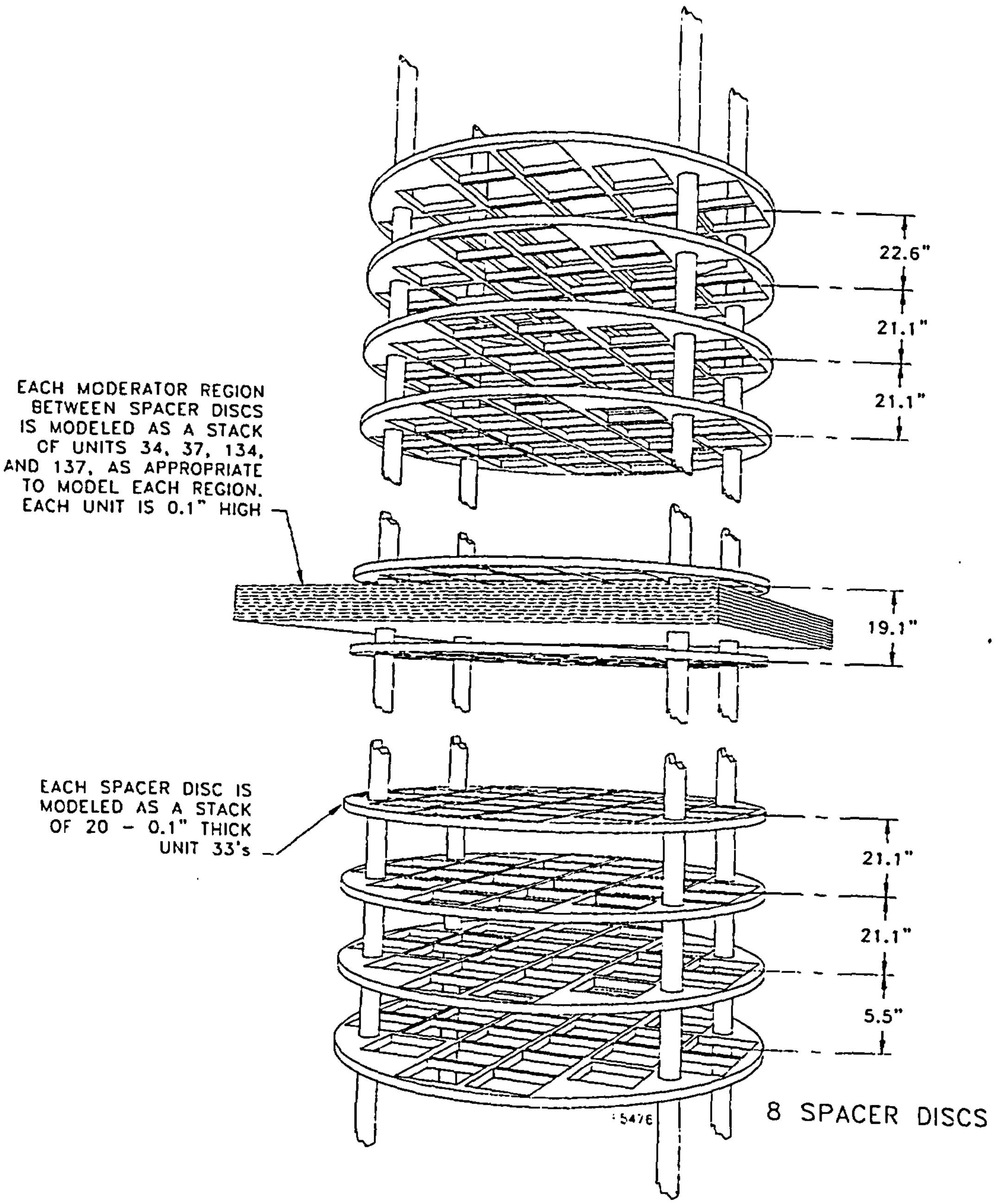


Figure 3.6-1
KENO Model of the DSC Basket

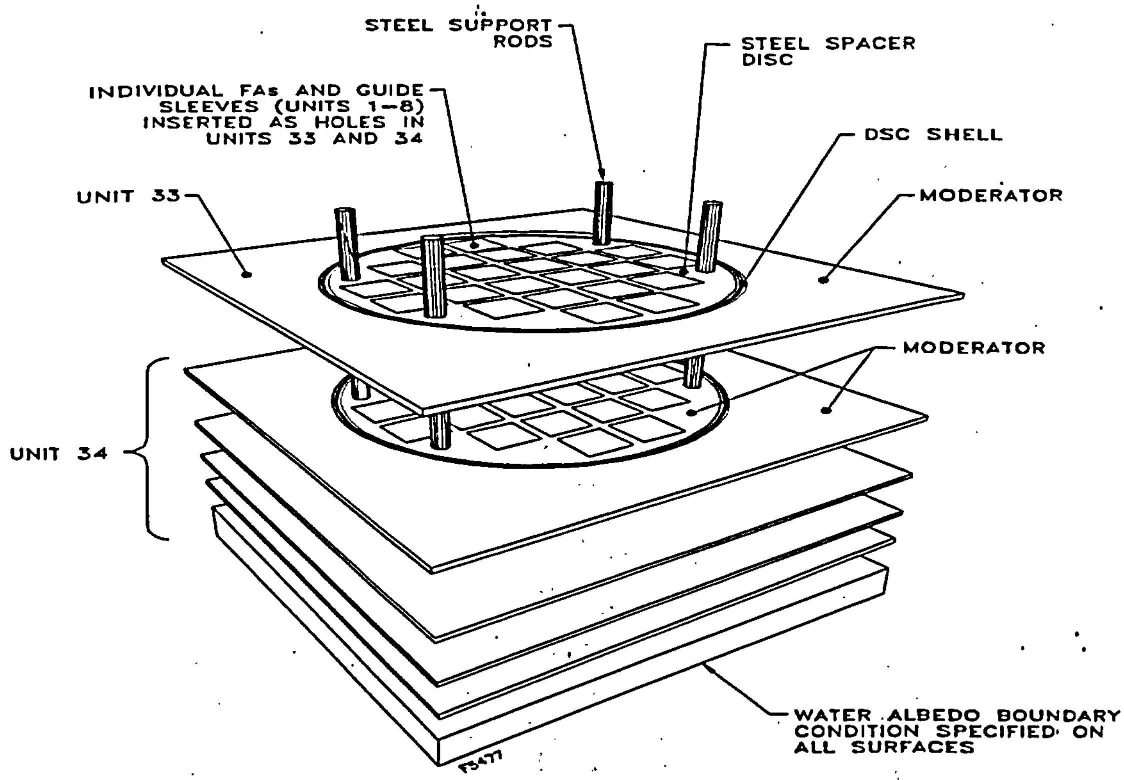


Figure 3.6-2
Exploded View of the KENO Model

Information withheld under 10 CFR 2.390(d)

Figure 3.6-3
DSC Geometry

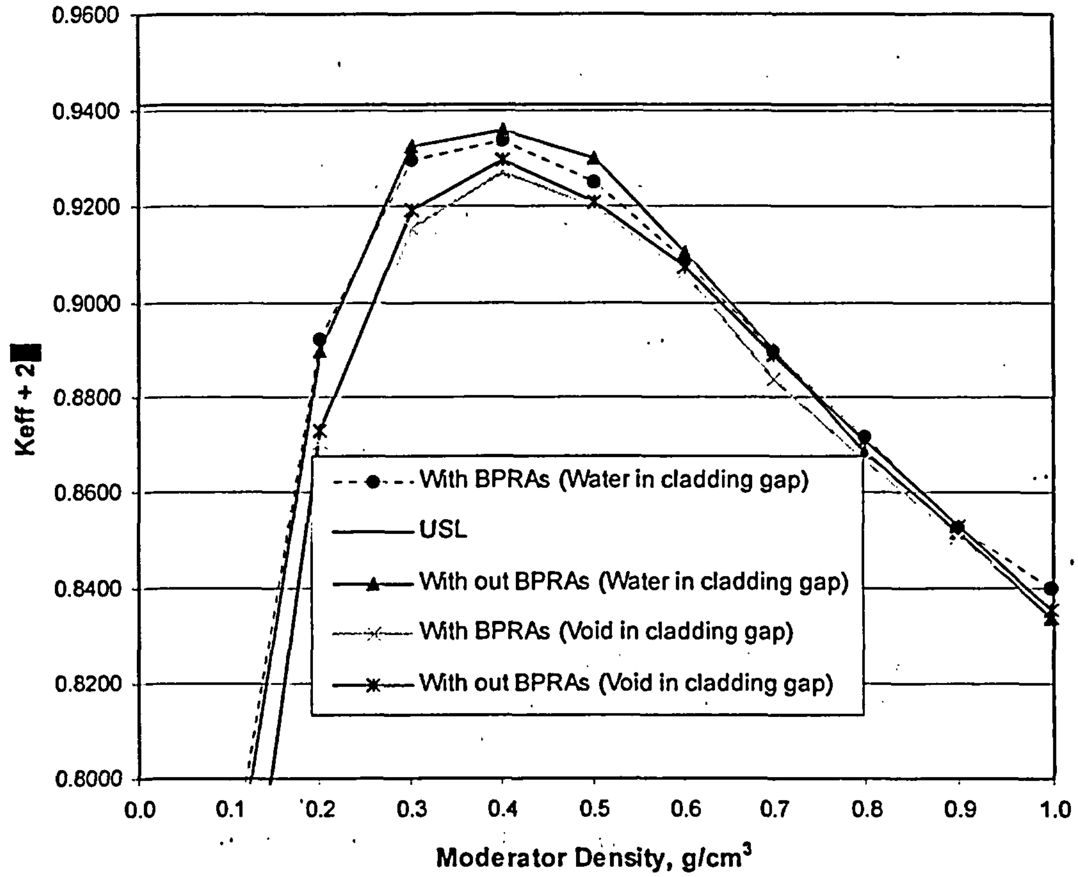


Figure 3.6-4
<u>**Criticality Results**</u>

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Pressure and leak tests of the DSC confinement boundary welds are performed according to the requirements of the ASME Code to the extent practical with exceptions as discussed in Section 4.8

The principal materials of construction for the NUHOMS® DSC are stainless and carbon steel. All structural component parts of the DSC are fabricated from these materials. Carbon steel is used for the DSC internals excluding the 24P guide sleeves, oversleeves, and support rods and the 52B spacer sleeves, support rods, and poison plates which are stainless steel. The DSC cylindrical shell and the cover plates which form the DSC containment pressure boundary are stainless steel. Lead is used as a shielding material in the long-cavity PWR canister shield plugs.

4.2.3.2 The Horizontal Storage Module

The design of the prefabricated NUHOMS® HSM has been developed in accordance with the applicable codes and a quality assurance program suitable for design of structures important to safety, as documented in Chapter 3. The design of the prefabricated NUHOMS® HSM has been performed using techniques similar to those reviewed and approved by the NRC for the NUHOMS®-24P design (4.13). The width and height of the HSM cavity and effective shielding thicknesses are the same.

The HSM is a massive reinforced concrete structure that provides protection for the DSC against tornado missiles and other potentially adverse natural phenomena. The HSM also serves as the principal biological shield for the spent fuel during storage. The NUHOMS® HSM design is illustrated in Figures 1.2-2 and 1.3-4. Drawings for the HSM are contained in Appendix E. Details for the standardized HSM are depicted in the illustrations and drawings referred to herein.

The HSM contains four shielded air inlet openings in the lower side walls of the structure to admit ambient ventilation air into the HSM as shown in Figure 4.2-6 and Figure 4.2-7. The air inlet and outlet openings in HSM Model 102 are lined with 1½ inch thick steel plates for improved shielding. The cooling ventilation air flows around the DSC (see Figure 1.3-5) to the top of the HSM. Air warmed by the DSC is exhausted through four shielded vent openings near the HSM roof slab. Adjacent modules are spaced to provide adequate ventilation flow and shielding. This passive system provides an effective means for spent fuel decay heat removal. A heat shield is provided between the DSC and HSM concrete to mitigate concrete temperatures.

The DSC rests on a frame structure with support rails in the cavity of the HSM, which is anchored to the HSM floor slab, side wall and front wall opening. The DSC support structure is fabricated from structural steel as shown in Figure 4.2-8. The DSC support structure member sizes and connection details are shown on the Appendix E drawings. The support structure is leveled and bolted to the HSM floor slab and side wall during module assembly. The support rails extend into the HSM front wall access opening, which is slightly larger in diameter than the DSC. The HSM access opening has a

stepped flange sized to facilitate docking of the transfer cask, as shown in Figure 4.2-9. This configuration minimizes streaming of radiation through the HSM opening during DSC transfer.

The DSC support rails are set parallel and leveled, then welded to embedded plates in the HSM opening. Thermal expansion of the support rails is accommodated by the DSC support system design. The top surfaces of the rails, on which the DSC slides, are coated with a dry film lubricant (see previous discussion for DSC) which is suitable for a radiation environment. The support rail sliding surfaces consist of hardened stainless steel cover plates for corrosion protection and added lubricity. Inside the HSM, the heat rejected from the DSC has a drying effect. Thus, the HSM atmosphere is benign in terms of corrosion, decay heat warms the air, thus preventing the accumulation or condensation of moisture inside the HSM.

The DSC is prevented from sliding along the support rails during a postulated seismic event by rail stops attached to the back ends of the DSC support rails, and a retainer located in the front access door of the HSM. The DSC axial retainer design is shown in Figure 4.2-9.

Clearance between the axial retainer and the DSC is designed for the maximum DSC thermal growth which occurs during the postulated HSM blocked vent case, as discussed in Section 8.2-7. During normal storage there is a small gap which may allow movement of the DSC relative to the HSM. This motion would produce a small increase in the DSC axial force due to seismic loads if these forces were sufficient to overcome friction between the rails and DSC. For conservatism, these effects have been included in the design analysis.

The HSM wall and roof thicknesses are primarily dictated by shielding requirements. The massive walls adequately protect the DSC against tornado missiles and other adverse natural phenomena. The tornado generated missile effects are considered to bound any other reasonable impact-type accident. The HSM wall thickness for individual modules and HSM arrays are specified on the Appendix E drawings and discussed in Section 4.1.2.

The entrance to the HSM Model 80 is covered by a thick steel door which provides shielding and protection against tornado missiles. The door assembly includes a solid concrete core which acts as a combined gamma and neutron shield. The door is attached to the front wall using four bolted clamps.

For increased shielding, the entrance to HSM Model 102 is provided with a 2-foot thick reinforced concrete door with ½ inch thick steel liner at the rear. The door is attached to the front wall by four 1½ inch diameter bolts.

- Design Specification certified by a professional engineer
- Formal Over Pressurization report
- Design performed by a firm(s) holding an N-stamp
- Fabrication performed by a firm(s) holding an N-stamp
- No Authorized Nuclear Inspector (ANI) certified inspector sign-off

These items have little affect on the functionality of the component DSCs but directly affect its ability to comply with the requirements of the ASME Code. The qualifications of the firms and personnel, procedures used to develop the design reports and fabrication specifications, and the lack of an N stamped vendor are all exceptions to the requirements of Subsection NCA. Technically, wherever the Code requires the Certificate Holder to perform some function, neither the designer nor the fabricator can comply since they are not formally functioning as the Certificate Holder. Hence Subsection NCA does not apply.

Technical Compliance

Technical compliance is compliance with the design rules, material specifications, fabrication processes, joint configurations, etc. that would allow the DSCs to comply with the Code. Tables 4.8-1 and 4.8-2 provide a discussion of exceptions made to the technical requirements of the Code for materials, fabrication, examination, and testing. Most of the exceptions occur because the DSC configuration is different from the classical pressure vessel addressed by the Code. If an Owner generated, certified ASME Design Specification had been available, then in accordance with the Code, each of these exceptions could be evaluated and possibly accepted. This Design Specification acceptance constitutes a Code interpretation by the certifying professional engineer and could permit stamping of the DSCs.

Fabrication and Inspection of Components

Permitting a non ASME Code certified fabricator to build the DSCs is an exception made to the Inspection of Components section of the Code. Neither an ANI nor a Code certified shop is required by the procurement documents to fabricate or inspect these components or by 10CFR72 and associated NUREG's. Therefore, the role of the Certificate Holder is missing from the fabrication and inspection process. These exceptions are provided in Table 4.8-1and Table 4.8-2.

Table 4.8-1
ASME Code Exceptions List for NUHOMS®-24P (Standard and Long Cavity), 24PT2 and 52B DSC Pressure Boundary Components

Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
NB-1100	Requirements for Code Stamping of Components	The 24P (standard and long cavity) and 52B, DSC shell assemblies are designed and fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical as described in here, but Code Stamping is not required. Therefore, Code Stamping is not required, the fabricator is not required to hold an ASME N or NPT stamp or be ASME Certified.
NB-2130	Material must be supplied by ASME approved material suppliers	All DSC pressure boundary sub-components designated as ASME on Appendix E DSC drawings are obtained from TN approved suppliers with Certified Material Test Reports (CMTR's). 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. Since the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible.
NB-4121	Material Certification by Certificate Holder	Material traceability and certification are maintained in accordance with TN's approved QA program
NB-4240 NB-5230	Full penetration welds are required for pressure boundary closure joints Weld examination shall be UT or RT with surface PT	DSC Pressure Boundary Welds: The joint details at the top and bottom end of the DSCs are not full penetration welds and thus do not comply with the requirements of figure NB-4243-1 for Category C flat head closure pressure and containment boundary welds. Volumetric weld inspection (RT or UT) is not practical due to the DSC geometry at the top and bottom closures and due to high radiation at the top closure after fuel loading (ALARA consideration). The inner and outer cover plate closure welds provide redundant closure welds, which are required by the 10CFR72 license. These welds are partial penetration welds that have been designed using a conservative "weld efficiency" factor of 0.6. Breach of the DSC confinement barriers due to an undetected flaw in any single weld layer is implausible due to the requirement for multi-layer welds. The top and bottom outer cover plate to shell welds and the inner bottom cover plate to shell weld receive a root and final PT. The top inner cover plate to shell weld, which is leak tested, has a final PT only. The final assurance against containment breach is that the normal operating pressure inside the DSC is low (approx. 5 psig compared to over 60 psig for the worst case accident pressure) (See Chapter 8 for the normal and accident pressures). The SER states in Section 3.2.1 that the normal operating pressure is closer to 1 psig. Thus, low pressure stresses result during normal operating conditions.

Table 4.8-1
ASME Code Exceptions List for NUHOMS®-24P (Standard and Long Cavity), 24PT2 and 52B DSC Pressure Boundary Components

(continued)

Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
NB-6111	All completed pressure retaining systems shall be pressure tested	The pressure retaining system of the DSC consists of the following components: shell, bottom inner and outer cover plates, siphon and vent block, and top inner and outer cover plates. The bottom cover plates are welded to the shell in the fabricator shop, whereas the top cover plates are field welded to the shell in the nuclear power plant following the loading of irradiated nuclear fuel. All other welds made to the pressure boundary, such as the support ring to shell weld are not part of the pressure boundary and thus are not pressure tested. <u>DSC Shell and Bottom Cover Plate Welds:</u> The DSC Shell and inner bottom cover plate are pressure tested during fabrication to the requirements of NB-6000. A helium leak test is performed to demonstrate leakage integrity of this boundary. Since the outer bottom cover plate is installed after the inner bottom cover plate is installed, it cannot be pressure tested. <u>DSC Top Cover Plates Closure Welds:</u> The top closure welds are not completed until the DSC is loaded with irradiated nuclear fuel; therefore, a pressure test is not performed. Multi-layer welds are used for these joints to eliminate potential leakage paths. The inner and outer top containment boundary welds are tested as follows: <u>Inner Top Containment Boundary Welds:</u> The inner top containment boundary welds include the following: (1) field weld of inner cover plate to shell weld (including inner top cover plate to vent and siphon block), (2) top of siphon and vent block to shell weld, and (3) field weld of siphon and vent port cover plates to vent and siphon block ports. Weld (1) is helium leak tested in the field. Weld (2) is made in the fabricator shop under controlled conditions and receives a final PT. A pressure test and helium leak test are not practical because of its location. A field leak test of weld (2) is not performed because the current 10CFR72 license does not require it. Weld (3) is performed in the field with a final PT and without a leak test. A helium leak test cannot be performed on these welds because the vent and siphon ports are covered by the plates. Pressurization would require cutting a hole in the DSC creating a potential leakage point for the long-term storage canister. <u>Outer Top Containment Boundary Weld:</u> The outer top cover plate to shell weld receives a root and final PT. It is not leak tested because it is installed following the inner top cover plate.

Table 4.8-1
ASME Code Exceptions List for NUHOMS®-24P (Standard and Long Cavity), 24PT2 and 52B DSC Pressure Boundary Components

(continued)

Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
NB-7000	Vessels are required to have overpressure protection	No overpressure protection is provided for the DSCs. The function of the DSC is to contain radioactive materials under normal, off normal and hypothetical accident conditions of storage. The DSCs are designed to withstand the maximum credible internal pressure at the maximum accident temperature and remain with ASME Code Level D allowable stresses. (See Section 8.2.9)
NB-8000	Requirements for nameplates, stamping and reports per NCA-8000	The DSC stamping provides the appropriate information required by 10CFR72; however, code stamping is not required. QA Data Packages are prepared in accordance with 10CFR72 and TN's approved QA program.

Table 4.8-2

ASME Code Exceptions List for NUHOMS®-24P (Standard and Long Cavity), 24PT2 and 52B DSC Basket Assembly

Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
NF-2130	Material must be supplied by ASME approved material suppliers	All DSC Basket Assembly sub-components designated as ASME on Appendix E DSC drawings are obtained from TN approved suppliers with Certified Material Test Reports (CMTR's). The DSC basket subcomponents listed below have been designated as non-Code. • Guide Sleeves, Oversleeves, and extraction stops (PWR only) • Neutron Absorber Plates and misc. hardware, such as anti-rotation pin, screws and locknuts, (BWR Only) • Coating for Spacer Discs
NF-4121	Material Certification by Certificate Holder	Material traceability and certification are maintained in accordance with TN's NRC approved QA program
NF-8000	Requirements for nameplates, stamping and reports per NCA-8000	The DSC stamping provides the appropriate information required by 10CFR72; however, code stamping is not required. QA Data Packages are prepared in accordance with 10CFR72 and TN's approved QA program.

4.9 ASME Code Exceptions List for the Transfer Cask

The transfer cask is a nonpressure retaining component that is conservatively designed and fabricated, where practicable and appropriate, in accordance with ASME Code Section III, Subsection NC requirements for a pressure retaining component. The following discussion documents and provides justification for deviations from these ASME Code requirements.

The technical requirements for the transfer cask are based on the following sections of the ASME Code:

- Section II for materials.
- Section III for materials, design, fabrication, testing, inspection, and over pressure protection.
- Section V for non-destructive examination.
- Section IX for welder and procedure qualifications.

Code Exceptions

The areas of possible exceptions to the ASME Code can be broken down into four basic areas. These are:

- Administration of the Code
- Technical design
- Fabrication of the components
- Inspection/Examination of the components

Although each of these areas is interrelated, each section is governed by different authorities.

Administration of the Code

This is generally covered in Section III, Division 1, Subsection NCA and is controlled by the type of contract placed for the design and fabrication of the component. The transfer cask was procured under the premise of following the technical requirements of the Code without requiring the use of an ANI and not applying an N stamp. Hence, many of the administrative items that would allow the cask to be stamped are not formally in place. This includes such things as a Design Specification certified by a professional engineer and a formal Over Pressurization report; and design and fabrication work being done by a

firm(s) holding an N-stamp. These items have little effect on the functionality of the component but directly affect its ability to comply with the requirements of the ASME Code. The qualifications of the firms and personnel, procedures used to develop the design reports and fabrication specifications, and the lack of an N stamped vendor are all exceptions to the requirements of Subsection NCA. Technically, wherever the Code requires the Certificate Holder to perform some function, neither the designer nor the fabricator can comply since they are not formally functioning as the Certificate Holder. Hence, Subsection NCA does not apply.

Technical Compliance

Technical compliance is compliance with the design rules and specification of materials, processes, joint configurations, etc. that allow the transfer cask to comply with the Code. The design is based on compliance with Section III of the ASME Code as modified by 10CFR72, NRC Regulatory Guides and NUREGs as discussed in the Chapters 3 and 8. Table 4.9-1 provides a discussion of technical exceptions to the written Code provisions for the materials, fabrication, examination, and testing. The majority of these exceptions are caused by the deviations in configuration of the cask from the classical pressure vessel addressed by the Code. If an Owner generated, certified ASME Design Specification had been available, then in accordance with the Code, each of these exceptions could be evaluated and possibly accepted. This Design Specification acceptance constitutes a Code interpretation by the certifying professional engineer and could permit stamping of the cask.

Fabrication and Inspection of Components

An exception to the Code is the use of a non-ASME Code certified fabricator for the fabrication of the transfer cask. Neither an ANI nor a Code certified shop is required by the procurement documents or 10CFR72 and associated NUREGs to fabricate or inspect the cask. Therefore, the role of the Certificate Holder is missing from the fabrication and inspection process.

Table 4.9-1

ASME Code Exceptions List for the Transfer Cask (Applies to Cask Structural Components Only, Lead Shielding, Neutron Shielding, and Neutron Shield Jacket of the Cask is Not Addressed by this Table)

Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
NC-1100	Requirements for Code Stamping of Components	As described in Chapters 3 and 8, the cask 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 10CFR72 regulation. Therefore, the fabricator is not required to be ASME Certified.
NC-2000	ASME Code Materials are to be used	The Cask 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 Appendix E drawings are obtained by TN approved suppliers with Certified Material Test Reports (CMTR's). 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-4240	Full penetration welds are required for pressure boundary closure joints.	The joint between the ram access penetration forging and the bottom end plate consists of partial penetration welds, while NC-3200 would require full penetration welds. 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-5250	Category A and B joints shall be fully radiographed.	Appendix E drawing NUH-03-8001 permits weld examination of (a) the circumferential and longitudinal welds for the structural shell and (b) the weld between the bottom end plate and the bottom support ring to be done using radiography (RT) or ultrasound (UT) while NC-5250 allows full penetration welds to be examined by RT only. Since the structural shell is not a pressure boundary, this code exception is acceptable.
NC-6000	All completed pressure retaining systems shall be pressure tested.	With respect to pressure testing requirements, the transfer cask is considered a non 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 transfer cask is considered a non 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.

Reference ASME Code Section/Article	Code Requirement	Exception, Justification and Compensatory Measures
NC-8000	Requirements for nameplates, stamping & reports per NCA-8000	The transfer cask nameplate provides the information required by 10CFR72. Code stamping is not required for the transfer cask. QA Data packages are prepared in accordance with the requirements of 10CFR72 and TN's NRC approved QA program.

4.10 References

4.1 U.S. Government, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," Title 10 Code of Federal Regulations, Part 72, Office of the Federal Register, Washington, D.C.

4.2 Deleted.

4.3 Deleted.

4.4 Deleted.

4.5 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1983 Edition, with Winter 1985 Addenda.

4.6 American Society for Testing and Materials, Annual Book of ASTM Standards, Section 4, Volume 04.02, 1990.

4.7 American Society for Testing and Materials, Annual Book of ASTM Standards, Section 1, Volume 01.04, 1990.

4.8 Deleted.

4.9 "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials", ANSI N14.6-1993, American National Standards Institute, Inc., New York, New York.

4.10 American Concrete Institute, "Building Code Requirement for Reinforced Concrete," ACI-318, 1983.

4.11 American Institute of Steel Construction, (AISC), "Specification for Structural Steel Buildings," Ninth Edition 1990, Chicago, Illinois.

4.12 American National Standards Institute, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," ANSI N14.5, 1977.

7.2.3 Fuel Qualification Tables

The purpose of this section is to document the methodology used to determine the cooling times required for PWR and BWR fuel assemblies, with various burnups and initial enrichments, for storage in the standardized NUHOMS® 24P and 52B systems. An acceptable fuel assembly meets the cladding temperature limits, overall heat generation limit, and design basis HSM and Transfer Cask (TC) surface dose rates presented in the Section 7.3.2. The methodology is based on preserving the following parameters for design basis fuel: the cladding temperature, the total dose rate on the exterior of the HSM and the TC radial surface thereby, assuring that the temperatures and dose rates calculated on and around the HSM and TC, using the design basis fuel source terms, remain bounding. The HSM roof surface dose rate is chosen for this evaluation because it represents the largest contribution to the exposure received by members of the public, both offsite and onsite. The TC radial surface is chosen because it represents the greatest surface area and dose rate on the TC surface to which workers are exposed during fuel loading operations.

For a wide range of assembly burnups and initial enrichments, the OCRWM Characteristics Database (CDB) [7.14] is used to determine the required cooling time to meet the decay heat and surface dose rate criteria described above for this evaluation. The results of this evaluation are provided in Fuel Qualification Table 3.1-8a and Table 3.1-8b.

Methodology

The standard NUHOMS® design basis fuel assemblies have a decay heat of 1.0 kW/assy for a PWR assembly and 0.37 kW/assy for a BWR assembly. A fuel assembly with a decay heat less than these design basis values results in maximum HSM, TC and DSC component temperatures less than those listed in Section 8.1.3. The maximum allowable fuel cladding temperature is a function of both the post irradiation cooling time and the fuel burnup. Allowable decay heats as a function of cooling time and burnups are based on the criteria in Reference [7.15]. These decay heats result in maximum fuel cladding temperatures that are less than the corresponding cladding temperature limit.

Surface neutron dose rates are assumed to be directly proportional to the total neutron sources in the assemblies. The primary neutron source in LWR spent fuel is the spontaneous fission of ^{244}Cm. For the ranges of burnups, initial 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. To account for the fact that the original analyses used a different neutron spectrum, the variation in the spectrum is accounted for by applying a 5% safety

margin to all neutron results. Surface gamma dose rates are determined for the HSM and TC surfaces using the actual gamma spectrum applicable for each case.

The BWR heavy metal weight used is 0.198 MTU per assembly. The PWR heavy metal weight used is 0.475 MTU per assembly to bound existing PWR fuel designs. Note that this is an increase over that used for the design basis shielding analysis in Section 7.3. The increase in heavy metal loading is accounted for by increased cooling time to offset the increase in the source terms.

The design basis HSM roof dose rate from Table 7.3-2 is 48.6 mrem/hr. Although not a regulatory or operational limit, the HSM roof dose rate of 48.6 mrem/hr is an appropriate acceptance criterion for the purposes of this evaluation. The HSM surface dose rate criterion assures the design basis offsite dose rates remain bounding.

The design basis transfer TC radial dose rate from Table 7.3-2 is 591.8 mrem/hr. Like the HSM dose rate, this is not a regulatory or operational limit, but is considered an appropriate acceptance criterion for the purposes of this evaluation. The TC surface dose rate criterion assures the design basis occupational exposures remain bounding.

For conservatism, all required cooling times are rounded to the next higher integral year. None of the safety margins in the design basis analyses are reduced in this evaluation. The acceptance criteria for this evaluation are that the cladding temperature is less than the applicable cladding temperature limit, that the HSM concrete temperatures are maintained, and that the surface dose rates are less than the design basis surface dose rates for both the HSM roof and TC side centerline.

PWR Fuel Evaluation – Decay Heat

The CDB provides decay heats in the units watts/MTIHM. For an assembly heavy metal loading of 0.475 MTU, the per assembly decay heat is determined using the relation,

$$DecayHeat = \frac{0.475 \cdot Q_{CDB}}{1000}$$

where Q_{CDB} is the CDB decay heat in units of watts/MTIHM for at given burnup, initial enrichment and cooling time. The calculated decay heat is checked against the allowable decay heat as given in Table 7.2-5. If the calculated decay heat is too high then the cooling time is increased until the decay heat limit is met.

Table 7.2-5
PWR Allowable Decay Heat Versus Cooling Time and Burnup

Cooling Time (years)	Allowable Decay Heat (kW/assy)	
	≤ 40 GWd/MTU	> 40 GWd/MTU ≤ 45 GWd/MTU
3	1.00	0.97
4	1.00	0.97
5	1.00	0.97
6	0.99	0.93
7	0.86	0.83
8	0.85	0.82
9	0.84	0.81
10	0.84	0.80
11	0.83	0.80
12	0.82	0.79
13	0.81	0.78
14	0.81	0.78
15	0.81	0.78
16	0.80	0.77
17	0.79	0.77
18	0.79	0.76
19	0.78	0.76
20	0.78	0.76
21	0.78	0.75
22	0.77	0.75
23	0.77	0.75
24	0.77	0.75
25	0.77	0.74
26	0.76	0.74
27	0.76	0.74
28	0.76	0.74
29	0.75	0.73
30	0.75	0.73

PWR Fuel Evaluation – Neutron Dose Rate

The CDB provides neutron sources in the units neutrons/s/MTIHM, for a given burnup, initial enrichment and cooling time. Neutron dose rates are determined for both the HSM roof and the TC side surface for every entry in the fuel qualification table. The HSM and TC neutron dose rates are each determined using the relation,

$$DoseRate = 1.05 \cdot \left(Dose_{DesignBasis}\right) \frac{0.475\left(Source_{CDB}\right)}{Source_{Section7.2}}$$

where $Dose_{DesignBasis}$ is the HSM roof or TC sidewall dose rate for design basis fuel (0.4 mrem/hr HSM and 163.9 mrem/hr TC, Table 7.3-2), $Source_{CDB}$ is the neutron source from CDB for a given assembly, and $Source_{Section\ 7.2}$ is the per assembly source for the design basis fuel (2.23E-08 n/s/assy). As discussed above, the calculated dose rates include a 5% safety factor to account for the spectral differences relative to the design basis neutron source listed in Table 7.2-2.

PWR Fuel Evaluation – Gamma Dose Rate

The CDB provides gamma spectra in the units γ/s/MTIHM for each of 18 energy groups for a given burnup, initial enrichment and cooling time. The HSM and TC gamma dose rates are determined by (1) Mapping the CDB gamma source and spectrum into the Cask-81[7.7] gamma energy structure used in the shielding evaluation; (2) Multiplying by the number of assemblies in the DSC and by the heavy metal weight; (3) Dividing by the fuel region volume; (4) Multiplying the resulting source in each energy group by a response function to determine the dose rate contribution from the group to the total surface dose rate; and (5) Summing the dose rate contribution from each energy group. Each of these steps is described in detail below.

The 18 gamma-ray group CDB energy spectrum has been mapped to the Cask-81 18 group energy spectrum used in the ANISN shielding models. The energy group mapping is performed by assuming that the particles in each group are evenly distributed in logarithmic energy space. The total source strength is conserved. The formulae used to map the CDB energy structure into the structure used in the shielding evaluation are shown in Table 7.2-6.

Table 7.2-6
Formulae for Mapping Gamma Source Spectra

CDB Structure Group	CDB Structure E_{mean} (MeV)	Cask-81 Group Structure Group	Cask-81 Group Structure E_{upper} (MeV)	Mapping Formula CDB → Cask-81 Group Structure
a	9.500	23	10.000	a
b	7.000	24	8.000	0.722b
c	5.000	25	6.500	0.278b+0.450c
d	3.500	26	5.000	0.550c
e	2.750	27	4.000	d
f	2.250	28	3.000	e
g	1.750	29	2.500	f
h	1.250	30	2.000	0.648g
i	0.850	31	1.660	0.352g+0.297h
j	0.575	32	1.330	0.703h
k	0.375	33	1.000	0.626i
l	0.225	34	0.800	0.374i+0.349j
m	0.125	35	0.600	0.651j+0.290k
n	0.085	36	0.400	0.710k
o	0.058	37	0.300	0.585l
p	0.038	38	0.200	0.415l+m
q	0.025	39	0.100	n+0.762o
r	0.010	40	0.050	0.238o+p+q+r

The mapped gamma source is multiplied by the number of assemblies in each DSC (24) and by the design heavy metal weight (0.475 MTU) to determine the total source in each energy group inside the DSC for the case being evaluated. This value is then divided by the fuel region volume in the ANISN models, 8,073,120 cm^3 to determine the total volumetric source (γ/s/cm^3) in each energy group. The source-to-dose rate response functions, described below, are then applied to the source spectrum to determine gamma dose rates on the HSM and TC surfaces.

The ANISN discrete-ordinates computer code is used to generate source-to-dose rate response functions to convert the group sources to surface dose rates on the HSM and TC. The ANISN models used are identical to those used in Section 7.3.2. Eighteen runs were performed for the HSM roof geometry and an additional 18 runs were performed for the TC side geometry. Each run includes a unit source (1 γ/s/cm^3) in a single energy group. The other input parameters, including geometry, materials, and flux-to-dose rate factors, are unchanged relative to the design basis analysis models. The gamma dose rate reported at the HSM (or TC) surface in each run represents the contribution from that energy group (per unit volumetric source) to the total gamma dose rate. The ANISN results, shown in Table 7.2-7, represent a response function which allows the gamma

dose rates to be determined for each fuel qualification case, including spectral effects, without the need to perform additional ANISN runs. The total surface gamma dose rate is then calculated by multiplying the source in each group (discussed above) by the response function for that group, and summing the result for all eighteen groups

Table 7.2-7
PWR HSM and TC Unit Gamma Source Response Functions
(mrem/hr per γ/s/cm³

Cask-81 Group	HSM Response Function	TC Response Function
23	6.55E-05	2.39E-05
24	4.86E-05	2.94E-05
25	3.04E-05	3.13E-05
26	1.63E-05	2.96E-05
27	7.67E-06	2.47E-05
28	3.03E-06	1.72E-05
29	1.28E-06	1.04E-05
30	4.45E-07	4.88E-06
31	1.48E-07	1.85E-06
32	3.34E-08	3.51E-07
33	5.32E-09	2.72E-08
34	9.66E-10	1.06E-09
35	9.33E-11	5.29E-13
36	3.06E-12	3.25E-18
37	1.95E-13	5.80E-20
38	1.88E-15	1.13E-32
39	3.42E-27	0.00E+00
40	0.00E+00	0.00E+00

The goal of this evaluation, however, is simply to compare the dose rates for various sets of fuel parameters to the dose rate for the design basis fuel parameters. The design basis dose rates reported in Table 7.3-2, therefore, are scaled by the ratio of the calculated dose rate for each case to the calculated dose rate for the design basis case. The end result of this effort is to specifically account for the gamma spectrum for every case on the fuel qualification table.

$$DoseRate = \left(Dose_{T7.3.2}\right)\frac{0.475\left(Dose_{case}\right)}{0.472\left(Dose_{DesignBasis}\right)}$$

Where $Dose_{T7.3.2}$ is the HSM roof or TC side gamma dose rate for design basis fuel (48.2 mrem/hr for the HSM and 427.9 mrem/hr for the TC) from Table 7.3-2, $Dose_{Case}$ is the gamma dose rate determined by ANISN using the Table 7.2-7 response functions for the

case being evaluated, and $Dose_{DesignBasis}$ is the dose rate determined above for the design basis source. The ratio of 0.475/0.472 scales the design basis dose rate up to a value consistent with a heavy metal weight of 0.475 MTU. The following is an example of the dose rate evaluation methodology for the design basis (40 GWd/MTU, 4.0 wt. %, 5-year cooled) case. The result of this evaluation is $Dose_{DesignBasis}$ as is used in the above equation.

Table 7.2-8 shows the calculation of $Dose_{DesignBasis}$ for both the HSM and TC. The first two columns, "CDB Energy" and "CDB (γ/s/MTIHM)" are the CDB energy group mean energies and gamma source, respectively for 4.0 wt. % ^{235}U, 40,000 MWd/MTU and 5-year cooling time. The "Shielding (γ/s/DSC)" column represents the total gamma source in a single DSC mapped into the 18 energy groups used in the shielding evaluation. The values in this column are determined by multiplying the CDB source by the heavy metal weight (0.475 MTU), the number of assemblies (24) and by the mapping function in Table 7.2-6 for the Cask-81 energy groups as labeled in the "Cask-81 Shielding Energy Group Column". As an example, the third value in the Cask-81 shielding energy group column (Cask-81 Group 25) is equal to (0.475*24)[(.278*1.99e6)+(.450*1.73e7)] = 9.485e7 γ/s/DSC where b=1.99E+06 for CDB energy 7.00 MeV and c=1.73E+07 for CDB energy of 5.00 MeV. The corresponding contribution to the HSM roof dose rate is then equal to the DSC source divided by the source volume and multiplied by the response function, (9.485e7/8,073,120)*3.04e-5 = 3.573e-4. This process is completed for each energy group and the sum shown at the bottom of Table 7.2-8. For the design basis PWR parameters of 4.0 wt. % ^{235}U, 40,000 MWd/MTU, and 5-year cooling the calculated "$Dose_{DesignBasis}$" for the HSM and TC are 82.48 mrem/hr and 846.3 mrem/hr, respectively.

Table 7.2-8
Sample PWR Dose Rate Evaluation

CDB Energy (MeV)	CDB (γ/s/MTIHM)	Cask-81 Shielding Energy Group	Shielding (γ/s/DSC)	HSM Dose	Cask Dose
1.00E-02	3.63E+15	23	2.606E+06	2.115E-05	7.712E-06
2.50E-02	8.55E+14	24	1.638E+07	9.861E-05	5.975E-05
3.75E-02	9.17E+14	25	9.485E+07	3.573E-04	3.674E-04
5.75E-02	7.19E+14	26	1.082E+08	2.188E-04	3.973E-04
8.50E-02	4.66E+14	27	2.016E+11	1.916E-01	6.167E-01
1.25E-01	4.64E+14	28	1.578E+12	5.923E-01	3.352E+00
2.25E-01	3.88E+14	29	5.018E+13	7.931E+00	6.473E+01
3.75E-01	2.29E+14	30	6.439E+13	3.552E+00	3.895E+01
5.75E-01	6.18E+15	31	2.123E+15	3.884E+01	4.851E+02
8.50E-01	1.43E+15	32	4.942E+15	2.041E+01	2.150E+02
1.25E+00	6.17E+14	33	1.023E+16	6.745E+00	3.449E+01
1.75E+00	8.72E+12	34	3.071E+16	3.675E+00	4.017E+00
2.25E+00	4.40E+12	35	4.664E+16	5.391E-01	3.057E-03
2.75E+00	1.38E+11	36	1.852E+15	7.008E-04	7.447E-10
3.50E+00	1.77E+10	37	2.589E+15	6.241E-05	1.861E-11
5.00E+00	1.73E+07	38	7.131E+15	1.657E-06	1.003E-23
7.00E+00	1.99E+06	39	1.155E+16	4.890E-18	0.000E+00
9.50E+00	2.29E+05	40	6.351E+16	0.000E+00	0.000E+00
Total	1.591E+16	-	1.814E+17	8.248E+01	8.463E+02

Similar evaluations were performed for each entry (cooling time) in the fuel qualification table to verify that the decay heat and dose rate determinations for PWR assemblies with burnups ranging from 10,000 MWd/MTU to 45,000 MWd/MTU and initial enrichments from 2.0 wt. % ^{235}U to 4.0 wt. % ^{235}U.

BWR Fuel Evaluation

The BWR fuel evaluations are performed in the same manner to those described above for the PWR fuel. BWR assemblies are evaluated with burnups ranging from 15,000 MWd/MTU to 45,000 MWd/MTU and initial enrichments ranging from 2.0 wt. % to 4 wt. % ^{235}U. The calculated decay heat is checked against the allowable decay heat given in Table 7.2-9. As with the PWR fuel if the calculated decay heat is too high then the cooling time is increased until the decay heat limit is met.

Table 7.2-9
BWR Allowable Decay Heat Versus Cooling Time

Cooling Time (years)	Allowable Decay Heat (kW/assy)
3	0.370
4	0.370
5	0.370
6	0.350
7	0.310
8	0.310
9	0.296
10	0.296
11	0.296
12	0.296
13	0.296
14	0.296
15	0.290
16	0.290
17	0.290
18	0.290
19	0.280
20	0.280
21	0.280
22	0.280
23	0.280
24	0.280
25	0.280
26	0.280
27	0.280
28	0.270
29	0.270
30	0.270

The HSM and TC source-to-dose rate response functions are calculated for BWR fuel similar to the PWR fuel. Because the design basis shielding evaluation in Section 7.3.2 only include fuel region number densities for PWR fuel, number densities for BWR fuel are required. The number densities generated for the 52B DSC evaluation are based on a GE BWR/4-6 7 X 7 GE-2 fuel assembly. This assembly was chosen because its heavy metal weight of 0.1947 MTU is closest to the BWR design basis of 0.198 MTU. The actual fuel assembly chosen has no impact on the results because the calculated dose rates are used only for comparison.

The BWR fuel assembly properties used in the number density calculations are taken from the CDB [7.14]. These include a heavy metal weight of 0.1947 MTU/assy, a zircaloy spacer weight of 2.029 kg/assy, an active fuel length of 144 in, a rod diameter of 0.563 in, a zircaloy cladding thickness of 0.032 in, and 49 fueled rods per assembly. The fuel channels and the DSC basket materials have been neglected. The mass and atom density for BWR fuel region is shown in Table 7.2-10.

Table 7.2-10
BWR Fuel Region Densities

Element	Atomic Mass (g/mol)	Mass Density (g/cc)	Atom Density (atoms/b•cm)
O	15.9994	0.169	6.348E-03
Zr	91.22	0.275	1.818E-03
U-235	235.04	0.050	1.285E-04
U-238	238.05	1.204	3.046E-03

With the exception of the fuel region atom densities, the ANISN source-to-dose rate response function models for the BWR fuel are identical to those for PWR fuel. The HSM and TC gamma dose rate response functions for the BWR fuel are shown in Table 7.2-11. For the $Dose_{DesignBasis}$ calculation, the BWR fuel parameters of 2.65 wt. % ^{235}U, 35,000 MWd/MTU, and 5-years cooling are used and the calculated gamma "$Dose_{DesignBasis}$" for the HSM and TC are 83.94 mrem/hr and 863.2 mrem/hr, respectively. These fuel parameters were also used to lookup the design basis neutron source per assembly from the CDB. This value is used in the evaluation to allow an "apples-to-apples" comparison of the neutron source for different fuel parameters. The $Dose_{DesignBasis}$ neutron and gamma sources for BWR assemblies are 1.83E+08 n/s/assy and 2.63E+15 γ/s/assy, respectively. Note that the design basis neutron source term is calculated using a different methodology than are used herein. The design basis BWR neutron source listed above is taken from the CDB for the design basis fuel parameters. This was done to provide an "apples-to-apples" comparison in the neutron scaling calculations.

Table 7.2-11
BWR HSM and TC Unit Source Response Functions

(mrem/hr per γ/s/cm^3)

Cask-81 Group	HSM Response Function	TC Response Function
23	7.69E-05	2.80E-05
24	5.71E-05	3.45E-05
25	3.58E-05	3.68E-05
26	1.93E-05	3.50E-05
27	9.10E-06	2.93E-05
28	3.61E-06	2.04E-05
29	1.52E-06	1.24E-05
30	5.32E-07	5.83E-06
31	1.76E-07	2.20E-06
32	3.96E-08	4.17E-07
33	6.28E-09	3.22E-08
34	1.13E-09	1.24E-09
35	1.08E-10	6.10E-13
36	3.48E-12	3.70E-18
37	2.20E-13	6.55E-20
38	2.11E-15	1.28E-32
39	3.90E-27	0.00E+00
40	0.00E+00	0.00E+00

Ten year fuel is shown since it is a physical impossibility for a utility to have a facility full of five year fuel. In fact, given the average age of fuel in U.S. storage pools, and the most probable NUHOMS® loading schedules, filled NUHOMS® ISFSIs should have substantially older fuel than indicated in the Figures.

The surface radiation sources used for the direct and air scattered dose calculations are shown in Figure 7.4-5 and Figure 7.4-6. The energy distribution of the neutron and gamma fluxes is taken from the applicable calculation as described in the previous sections. Air-scattered dose rates are determined with the computer code Micro SKYSHINE (7.4); direct dose rates are calculated using the computer code MICROSHIELD (7.11). No credit is taken for shielding by nearby structures or terrain. Initial loading of all HSMs with the ten year cooled fuel is assumed. Dose rates for the PWR DSC are provided since these values bound the BWR DSC dose rates.

The ISFSI is generally surrounded by a large open area for operational and security purposes. Access to the storage modules is restricted such that during storage, no access is allowed except for security and surveillance inspection purposes. There are generally no work areas close to the ISFSI. Additional dose to plant workers due to exposure from the ISFSI is negligible. Inspection of the HSM air vents can be maintained ALARA by keeping inspection personnel back from the HSM front wall a distance which permits adequate inspection.

Since the site dose for an ISFSI is highly site specific, each licensee should perform a dose analysis in accordance with 10CFR72.212. The analysis should consider existing plant conditions, the site specific arrangement of the ISFSI, the characteristics of the spent fuel to be placed in dry storage, and relevant empirical data as appropriate. The on-site dose analysis should demonstrate compliance with the 10CFR 72.104(a) limits for normal conditions and 10CFR72.106 and 10CFR100 for accident conditions.

Table 7.4-1
NUHOMS® System Operations Enveloping Time
for Occupational Dose Calculations

(for information only)

	Number of Workers	Completion Time(3) (hours)
Location: Auxiliary Building and Fuel Pool		
Ready the DSC and Transfer Cask for Service	2	4.0
Place the DSC into the Transfer Cask	3	1.0
Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	2	2.0
Fill the DSC Cavity with Water (borated for PWRs) (1)	1	6.0
Place the Cask Containing the DSC in the Fuel Pool	5	1.0
Verify and Load the Candidate Fuel Assemblies into the DSC	3	8.0
Place the Top Shield Plug on the DSC	3	1.0
Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	5	2.0
Location: Cask Decon Area		
Decontaminate the Outer Surface of the Cask(2)	7	1.0
Drain Water Above DSC Shield Plug	3	1.0
Decon the Top Region of the Cask and DSC	2	1.0
Remove a Small Volume of Water from the DSC Cavity(2)	2	0.5
Remove the Cask/DSC Annulus Seal and Set-up Welder	2	1.5
Weld the Inner Top Cover to the DSC Shell and Perform NDE (PT) (1)	2	6.0
Drain the Cask/DSC Annulus and the DSC Cavity(1)	2	3.0
Vacuum Dry and Backfill the DSC with Helium(1)	2	16.0
Helium Leak Test the Shield Plug Weld	2	1.0
Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE (PT)	2	1.5
Fit-up the DSC Top Cover Plate	2	1.0
Weld the Outer Top Cover Plate to DSC Shell and Perform NDE (PT)(1)	2	16.0
Install the Cask Lid	2	1.0

7.11 Grove Engineering, Inc., "Microshield User's Manual, A Program for Analyzing Gamma Radiation Shielding," Ver. 2.0, 1985.

7.12 American Nuclear Society Standards Committee Working Group ANS-6.1.1, "American National Standard Neutron and Gamma-Ray Flux-to-Dose-Rate Factors" ANSI/ANS-6.1.1-1977, American Nuclear Society, 1977.

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

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

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

The thicker roof and front wall sections qualify as deep flexural members and the allowable shear capacity (V_c) may be calculated in accordance with Section 11.8.6 of ACI 349 using the formula:

$$V_c = \left(3.5 - 2.5\frac{M_u}{V_u d}\right)\left(1.9\sqrt{f'_c} + 2500\,\rho_w \frac{V_u d}{M_u}\right) b_w d \qquad (8.1\text{-}8)$$

but not to exceed

$$6\sqrt{f'_c}\, b_w d$$

8.1.1.6 HSM Door Analyses

The access opening for transferring the DSC into and out of the HSM is protected by a shielded door. The standard door (used in HSM Model 102) is a thick reinforced concrete door with a steel plate at the rear. The standard door design provides for increased shielding compared to the alternate door (used in HSM Model 80). The alternate door is a steel door with a core of concrete shielding material. Both doors are recessed into the HSM front wall and bear against the concrete docking flange as shown in Appendix E drawings. The door is attached to the front wall by four embedded studs with threaded nuts.

Standard Door Analysis: The weight of the 24 inch thick reinforced concrete door (with ½ inch thick steel plate at the rear) is 11.2 kips. The door is designed conservatively for a maximum pressure of 10 psi which envelops the equivalent pressure load due to seismic and tornado wind pressure. The maximum moment and shear forces due to the enveloping load are 26.4 kip-in/ft and 1.98 kips/ft, respectively. These computed moments and shear forces are significantly less than the bending capacity (457 kips-in/ft) and the shear capacity (18.3 kips/ft) of the door.

Alternate Door Analysis: For normal system operation, the door assembly is only subjected to the dead weight which is transmitted by bearing directly into the HSM front wall, and handling loads resulting from installation and removal of the door during DSC transfer operations.

The concrete bearing strength required to support a bearing load on the frame from the door weight of 6556 pounds is a small fraction of the ACI 349 (Section 10.15) permissible concrete bearing strength of $\phi[0.85\ f'_c\ A_1] = 0.7\ [0.85 \times 5 \times 6 \times 80.63] = 1440$ kips. The embedded anchors for the HSM door frame are designed in accordance with ACI 349-85, Appendix B. The governing design load combination for the HSM door embedded anchors is the dead load plus tornado wind load combination. The dead load consists of the weight of the door. The wind load consists of the uniform suction acting on the door. The wind load produces shear and tension on the anchors. The maximum stress in the door subjected to the tornado wind pressure drop load is 3.17 ksi, which is much less than the allowable stress of 33 ksi. The maximum stress in the door due to seismic load is 0.4 ksi, which is less than the allowable stress of 39.4 ksi. The maximum tensile load in the anchors is 18.1 kips which is less than the allowable tension load of 106.0 kips.

8.1.1.7 HSM Heat Shield Analysis

The design of the HSM heat shield assembly is shown in Figure 4.2-7 and on the drawings in Appendix E. The heat shield acts to reduce the HSM concrete temperatures due to the heat rejected by the spent fuel assemblies during normal operating conditions, off-normal conditions, and postulated accident conditions. The only loading that the heat shield assembly is subjected to during normal operation is its own dead weight. Since over-size holes are used at the bolted connections to the HSM no thermal expansion loads are experienced by the heat shield.

The only loading that the HSM heat shield assembly experiences during off normal conditions or a postulated accident is the inertia force associated with a seismic event. The heat shield, studs and the support embedments are analyzed for normal, off-normal, and accident loads. The heat shield is conservatively analyzed as a series of simply supported beams with a span equal to the distance between two adjacent bolts in the longitudinal direction. Also, instead of performing a frequency analysis of the side wall heat shield plates to establish the dynamic amplification factor (DLF) the maximum DLF of 4.25 for 2% damping is selected from the design response spectra curve.

Using these conservative assumptions, the maximum bending stress in the plates obtained for dead weight and seismic load is 5.4 ksi, which is well within the allowable limit of 26.8 ksi, at an operating temperature of 270°F. Maximum axial and bending stresses in the studs are 2.1 ksi and 32.8 ksi, which give an interaction ratio of 0.92. The maximum shear stress in the studs is 0.5 ksi which is less than the allowable shear stress of 18 ksi. The calculated stresses demonstrate that the heat shield is capable of withstanding any normal, off-normal, or postulated accident condition.

8.1.1.8 HSM Axial Retainer for DSC

The design of the HSM axial retainer for the DSC is shown on the drawings in Appendix E. Additional details are provided in Section 8.2.3.2.

8.1.1.9 On-Site Transfer Cask Analysis

The on-site transfer cask is evaluated for normal operating condition loads including:

1. Dead Weight Load
2. Thermal Loads
3. Handling Loads

The NUHOMS® transfer cask is shown in Figure 1.3-6 and on the drawings contained in Appendix E. The transfer cask to be used by a utility may be of the design documented in Appendix E, or any other previously NRC reviewed and approved design such as the transfer cask designs documented in the NUHOMS®-24P Topical Report [4.13], the Oconee Nuclear Station ISFSI Safety Analysis Report [4.16], and the Calvert Cliffs ISFSI Safety Analysis Report [4.17], provided that it is demonstrated prior to use that the limiting conditions of use as described in the COC can be met. Three transfer casks, the standardized, OS197, and OS197H

Table 8.1-3
Mechanical Properties of Materials

(concluded)

(1) Steel data and thermal expansion coefficients are obtained from ASME Boiler and Pressure Vessel Code, Section III-1 Appendices. (8.3)

(2) Concrete and reinforcing steel data were obtained from Handbook of Concrete Engineering, by Mark Fintel. (8.22)

(3) Lead data was obtained from CRC Handbook of Tables for Applied Engineering Science, 2nd Edition, pp. 111 and 118. (8.4)

(4) Data obtained from manufacturers published information.

(5) Age hardened at 1150°F in accordance with notes of ASME Code, Appendix I, Table I-1.1, I-2.1 and I-3.1.

(6) Deleted

(7) Allowable stress intensity values (S_m) and the yield strength (S_y) for ASTM A36 steel are given in Table I-11.1 and Table I-13.1, respectively, of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix I.

(8) Carbon steel DSC basket materials shall be impact tested in accordance with ASME Code Subsection NF Article NF-2300. Impact test temperature is -20°F.

(9) Carbon steel transfer cask structural shell, trunnion sleeves, and trunnion materials shall be impact tested in accordance with ASME Code Subsection NC Article NC-2300. The lowest service temperature is 0°F.

(10) ASME SA479 Type XM-19 material is age hardened/annealed at 1925°F to 1975°F.

(11) ASME SA564 Type 630 material for 52B Support Rods and Spacer Sleeves is age hardened at 1100°F. Support rod material shall be impact tested in accordance with ASME Code, Section II. Impact Test temperature is –20°F.

Table 8.1-4
Estimated NUHOMS®-24P Component Weights

Component Description		Calculated Weight (Pounds)
1.	Dry Shielded Canister Shell Assembly	15,778
2.	DSC Top Shield Plug	7,859
3.	DSC Internal Basket Assembly	12,189
4.	DSC Inner and Outer Top Cover Plates	1,934
5.	24 PWR Spent Fuel Assemblies	≤40,368[4]
6.	Weight of Water in DSC Cavity	14,843
	Total Wet DSC Loaded Weight (w/o DSC inner and outer top cover plates.)	91,038
	Total Dry DSC Loaded Weight (w/ DSC inner and outer top cover plates.)	78,129
7.	Standardized Transfer Cask Empty Weight	107,091[1][3]
8.	Standardized Transfer Cask Max. Loaded Weight	193,642[2][5]
9.	HSM Single Module Weight, Model 80 (empty)	243,000
10.	HSM Single Module Weight, Model 102 (empty)	253,000

(1) Includes weight of cask top cover plate assembly.
(2) Weight includes: DSC dry weight plus fuel, plus water in DSC and cask less DSC and cask top cover plate assemblies.
(3) The as-built empty weight for the OS197 transfer cask is 111,250 lbs, including neutron shield water.
(4) The standard design DSC fuel assembly weight of 1,682 lbs/assembly is used.
(5) The maximum loaded weight for the OS197 transfer cask without DSC and OS197 top cover plates is 199,372 lbs.

Table 8.1-5
Estimated NUHOMS®-52B Component Weights

Component Description		Calculated Weigh (Pounds)
1.	Dry Shielded Canister Shell Assembly	15,658
2.	DSC Top Shield Plug	7,621
3.	DSC Internal Basket Assembly	12,012
4.	DSC Inner and Outer Top Cover Plates	1,934
5.	52 BWR Spent Fuel Assemblies	≤37,700
6.	Weight of Water in DSC Cavity	16,211
	Total Wet DSC Loaded Weight (w/o DSC inner and outer top cover plates.)	89,202
	Total Dry DSC Loaded Weight (w/ DSC inner and outer top cover plates.)	74,925
7.	Standardized Transfer Cask Empty Weight (w/collar)	113,501(1)(3)
8.	Standardized Transfer Cask Max. Loaded Weight	198,294(2)(4)
9.	HSM Single Module Weight, Model 80 (empty)	252,000
10.	HSM Single Module Weight, Model 102 (empty)	263,000

(1) Includes weight of cask top cover plate assembly.
(2) Weight includes: DSC dry weight plus fuel, plus water in DSC and cask less DSC and cask top cover plate assemblies.
(3) The as-built empty weight of the OS197 transfer cask is 111,250 pounds, including water in the neutron shield.
(4) The maximum loaded weight for the OS197 transfer cask without DSC and cask top cover plates is 196,197 lb.

Table 8.1-6
NUHOMS®-24P DSC Operating and Accident Pressures[5] (w/o BPRAs)

Operating Condition	Limiting Case Description	Max DSC Pressure[4] (psia)	Max Total DSC Pressure (psia)	Design Basis Pressure (psia)
Normal	DSC in Cask, 100°F	21.4	21.7[1]	24.7
Off-Normal	DSC in Cask, 100°F	21.4	24.8[2]	24.7
Accident	Blocked HSM vents, 125°F	26.8	73.4[3]	74.7

(1) Normal operating total pressure with 1% of fuel rod cladding failure.
(2) Off-normal operating total pressure with 10% of fuel rod cladding failure. The calculated pressure exceeds the design basis pressure by 0.1 psig. This is acceptable based on paragraph NB-3223(a) of ASME Section III, Subsection NB.
(3) Enveloping accident total pressure with 100% of fuel rod cladding failure.
(4) Total DSC internal pressure without any fuel rod cladding failure.
(5) Maximum 24P DSC internal pressures for Normal, Off-Normal and Accident conditions when storing fuel with BPRAs are presented in Appendix J.

	W_s	=	89.3 kips, Weight of end shield wall
	d	=	58 in. (4.83 ft.), Horizontal distance between center of gravity of HSM to the outer edge of the side wall.
	d_s	=	18 in. (1.50 ft.), Horizontal distance from the module to the shield wall center of gravity.
Therefore:	M_{st}	=	30,758 K-in.

and the overturning moment (M_{ot}) for the free standing module with two end shield walls and a rear shield wall due to DBT wind pressure is:

$$M_{ot} = W_1 A_w h/2 + W_3 A_r (d + d_p) \qquad (8.2\text{-}2)$$

Where:	W_1	=	(0.397 + 0.196) K/ft.^{2}, Wind load (windward plus leeward)
	h	=	180 in. (15.0 ft.) Wall height
	W_3	=	0.357 K/ft.^{2}, Wind uplift on roof
	A_r	=	241.3 ft.^{2}, Roof area (including shield wall)
	A_w	=	297.5 ft.^{2}, Wall area
	d_p	=	15 in. (1.25 ft.) One-half thickness of shield wall plus 6 inch gap
Therefore:	M_{ot}	=	22,168 K-in.

Since the overturning moment is smaller than the stabilizing moment, the free-standing HSM will not overturn. The resulting factor of safety against overturning effects for DBT wind loads is 1.38.

The tornado pressure drop of 3 psi is more controlling than the 357 lb./ft.^{2} DBT negative pressure acting on the HSM door. The pressure drop results in a total load of 18.1 kips on the door, which is reacted by the HSM door attachment anchors. The door attachment embedded anchors have a tensile load capacity that easily exceeds this total load. All other loads acting on the HSM door assembly envelop DBT negative pressure effects. Therefore DBT negative pressure loads have a negligible effect on the HSM door assembly design.

(ii) HSM Sliding Analysis

To evaluate the potential for sliding of a single, free-standing HSM (One HSM module with two end shield walls and a rear shield wall) the sliding force generated by the postulated DBT wind pressure is compared to the sliding resistance provided by friction between the base of the HSM and the ISFSI basemat. The BWR HSM is the critical configuration presented below.

The force (F_{sl}) required to slide a free standing module is:

$$F_{sl} = [W + 2W_s + W_{rs} - 2L(d + T_s + 6)W_3]\mu \qquad (8.2\text{-}3)$$

Where: μ = 0.6, coefficient of friction (ACI 349-85) (8.20)

W, W_s, W_3 and d are defined above. The 6" gap between the side wall and the end shield wall is included in the uplift force calculation.

W_{rs} = 66 kips, Weight of rear shield wall

L = 19 ft, Length of HSM side walls

T_s = 2 ft, Thickness of shield wall

Substituting gives:

$$F_{sl} = 278.6 \text{ kips}$$

The sliding force (F_{hw}) generated by DBT wind pressure for a single HSM is:

$$F_{hw} = W_1[h\,(L + T_s)]$$

Where: W_1, h, L and T_s are as defined above

Substituting gives:

$$F_{hw} = 194.2 \text{ kips}$$

Since the horizontal force generated by the postulated DBT is smaller than the force required to slide the end module in an HSM array, the HSM will not slide. The factor of safety against sliding of the HSM due to DBT wind loads is 1.43.

B. Effects of DBT Wind Pressure Loads on Transfer Cask

The transfer cask design is evaluated for the effects of tornado wind loads in accordance with 10CFR72.122 and ANSI 57.9 criteria. This evaluation is performed for the transfer cask secured horizontally to the transport trailer/skid. Both overall stability and maximum cask stresses are evaluated.

The critical overturning case for the transfer cask stability occurs when the wind loads are applied perpendicular to the cask/skid/trailer.

The stabilizing moment (M_{st}) is given by:

$$M_{st} = W_t d$$

Where: W_t = 225 kips, Minimum weight of cask/skid/trailer

d = 66 in., Half wheel base of trailer

Therefore: M_{st} = 14,800 K-in.

Conservatively assuming if the combined cask/skid/trailer has a solid vertical projected area, and ignoring the reduction in total wind pressure due to the open areas and shape factors, the maximum overturning moment (M_{ot}) for the cask/skid/trailer due to DBT wind pressure is:

$$M_{ot} = (W_1 + W_2)(A)\, h/2$$

Where: A = 232 ft.^{2}, Combined vertical projected area of cask/skid/trailer

W_1 = 0.397 k/ft.^{2}, Wind load windward side

W_2 = 0.196 k/ft.^{2} Wind load leeward side

h = 146 in., Height to top of cask during normal transfer operations

Therefore: M_{ot} = 10,000 k-in.

Since the overturning moment is smaller than the stabilizing moment, the cask/skid/trailer will not overturn. The resulting factor of safety against the overturning effects for DBT wind loads is about 1.5.

The maximum stresses induced in the transfer cask structural shell by DBT wind pressure loads are very conservatively calculated using the correlation presented in Roark (8.16) Table 31 Case 9.c. The wind pressure loads are applied as a line load to the cylindrical shell. Substituting the cask physical dimensions and an equivalent line load of 0.24 k/inch (397 psf x 85.3/144) into the correlation, the maximum calculated shell stress is 3.8 ksi. Similarly, the maximum tornado wind load pressure stresses induced in the top and bottom end plates are calculated using the Roark correlations given in Table 24 Cases 10a and 10b for the simply supported (bolted) top cover and fixed (welded) bottom end plates. The maximum calculated DBT wind pressure stress calculated for these items is 0.5 ksi. Since the resulting DBT transfer cask stresses are a small fraction of the ASME Code Level A allowables, DBT wind loads are not considered further.

The stresses calculated above should be verified by the licensee if the dimensions of the trailer and the skid equipment at the site do not correspond to the value used in this analysis.

C. HSM Missile Impact Analysis

The side walls and roof slab of the reinforced concrete HSM are 18 and 36 inches thick respectively. The side walls of the end HSM in an array are protected from tornado missile impact by 24 inch thick shield walls. The walls and roof are designed to provide adequate biological shielding and easily meet the minimum acceptable barrier thickness requirements for local damage against tornado generated missiles, specified in Table 1 of Section 3.5.3, NUREG-0800. However, in order to demonstrate the adequacy of the HSM design for tornado missiles, a bounding analysis of a free standing module is performed. The items evaluated include the resistance to penetration, scabbing and perforation for a postulated missile impact. For these analyses, a rigid penetration resistant missile consisting of a 280 pound, eight inch diameter blunt nosed hardened steel object is conservatively postulated. The method of analysis is based on the modified NDRC formula as recommended in Section 3.5.3 of NUREG-0800.

The door covering the access opening of the HSM is also evaluated for DBT missile penetration resistance. The HSM Model 80 (alternate) door is constructed of steel plate and filled with concrete. Missile impact on this door is resisted by the composite action of the concrete encased in the inner outer and side casing steel plates. Missile impact on the HSM Model 102 (standard) door is resisted by the 24" thick concrete with ½" thick steel liner. The results of this evaluation indicate that the HSM access door provides sufficient capacity to preclude perforation.

The DBT missile penetration resistance analyses for the HSM are presented in the following paragraphs.

(i) Missile Impact Penetration Resistance Analysis

The modified National Defense Research Committee (NDRC) formula from Kennedy, Holmes and Narver (8.32) is used to predict the HSM wall penetration depth for a postulated DBT missile.

$$x = \sqrt{4KNWd^{-0.8}[\frac{V}{1000}]^{1.8}} \qquad (8.2\text{-}5)$$

When: $\frac{x}{d}$ = ≤ 2.0

Where: x = Total penetration depth (in.)

d = 8 in., Projectile diameter

K = $180/\sqrt{f_c'}$, Concrete penetrability factor

= 2.55 for $f_c' = 5000$

N = 0.84 (blunt nosed), Projectile shape factor

f'_c = 5000 psi, Concrete compressive design strength at 150°F

W = Projectile weight = 280 lbm (conservatively assumed)

V = Striking velocity = 185.0 ft./s

Therefore: x = 4.67 inches

The perforating thickness, or maximum thickness that the postulated DBT missile will completely penetrate, is calculated using the correlation:

$$\frac{e}{d} = 1.32 + 1.24\left(\frac{x}{d}\right) \text{for } 1.35 \leq \frac{x}{d} \leq 13.5 \qquad (8.2\text{-}6)$$

$$(e/d) = 3.19\,(x/d) - 0.718\,(x/d)^2 \quad \text{for } x/d \leq 1.35 \qquad (8.2\text{-}7)$$

$x/d = 4.67/8 = 0.584 < 1.35$

Substituting in equation 8.2-7 yields:

$$e = 3.19(0.584) - 0.718(0.584)^2 = 12.95 \text{ in}$$

Therefore; e, the maximum perforation thickness, is conservative.

The minimum thickness necessary to prevent scabbing of material from the rear face of the target is calculated using:

$$\frac{s}{d} = 2.12 + 1.36\left(\frac{x}{d}\right) \text{ for } \left(0.65 \le \frac{x}{d} \le 11.8\right) \quad (8.2\text{-}8)$$

$$(s/d) = 7.91\,(x/d) - 5.06\,(x/d)^2 \quad \text{for } (x/d \le 0.65) \quad (8.2\text{-}9)$$

Using equation 8.2-9 yields:

s = 23.1 in.

Where: s = Scabbing thickness (in.)

Code requirements for nuclear safety related concrete structures (ACI 349-85) require a minimum of 20% additional wall thickness to prevent perforation and scabbing. Scabbing effects control the minimum required wall thickness. Therefore, the minimum wall thickness required to provide adequate protection for the enveloping DBT missile is:

$1.2s = 27.7$ in.

The specified minimum wall thickness for exterior HSM walls is at least 30 inches including the thickness of the precast shield walls. Consequently, there is adequate protection against local DBT missile impact damage.

The thickness of the end module shield walls (24 inches) may result in some minor scabbing occurring due to the worst case missile impact. Any concrete that is displaced remains within the space between the adjacent module. In the worst possible case, wall debris may block one side of the HSM air inlet or outlet vents. This case is bounded by the blocked inlet and outlet vents case discussed in Section 8.2.7. The combined wall thickness at this location exceeds 36 inches and no possible scabbing of the HSM affects the DSC.

(ii) Local Barrier Impingement Analysis

A composite door comprising of steel plates and concrete is used for the HSM Model 80 and a 24" thick concrete door with ½" inner steel liner is used for the HSM Model 102 to cover the HSM access opening after the DSC is in place. The HSM door is analyzed to verify its adequacy for local barrier impingement of a DBT missile. The 280 pound, eight inch diameter artillery shell is used for this calculation as it envelops effects caused by the postulated one inch diameter solid steel sphere. The minimum thickness of a steel plate that can be perforated by the postulated DBT missile is given in the McDonalds, Mehta, and Minor paper (8.33) as:

$$T = \frac{(0.5 M_m V_s^2)^{2/3}}{672 d_m} = 0.52 \text{ in.} \quad (8.2\text{-}10)$$

Where:	T	=	Perforation thickness (in.)
	M_m	=	Mass of missile $= \frac{W}{g} = 8.7$ lb-sec^2/ft
	W	=	Weight = 280 lb. (conservative)
	g	=	32.2 ft./s^2
	V_s	=	Missile strike velocity = 185.0 ft./s
	d_m	=	Diameter of missile = 8 in.

The steel HSM Model 80 door (alternate door) specified exceeds the minimum required perforation thickness of 0.52 inch by a wide margin. The standard door with 24" thick concrete and ½" steel inner liner of the HSM Model 102 is adequate to prevent any local damage due to missile impact.

(iii) Massive Missile Impact Analysis

The HSM stability and potential damage due to impact of the postulated DBT massive missile consisting of a 4000 lb. automobile, 20 sq. ft. frontal area travelling at 195 ft./sec., is evaluated. The massive missile is assumed to impact the shield wall of free standing HSM module. Using the principles of conservation of momentum with a coefficient of restitution of zero, the analysis presented below demonstrates that the free standing module remains stable and the missile energy is dissipated by sliding or slight tipping of the module.

Using conservation of momentum, the missile impact force equals the change in linear (sliding) or angular (overturning) momentum of the HSM. The HSM velocities immediately after impact are:

Sliding:

$$V = \frac{m\, v_i}{M+m} \qquad (8.2\text{-}11)$$

Overturning:

$$\omega_A = \frac{md\, v_i}{md^2 + I_A} \qquad (8.2\text{-}12)$$

Where:	V	=	Initial linear velocity of module after impact
	v_i	=	195.0 ft./sec., Initial velocity of missile
	ω_A	=	Initial rotational velocity about point A (Figure 8.2-11)
	m	=	4000/386.4 = 10.35 lb-sec ^{2}/in, Mass of missile

M	=	1471.5 lb-sec ^{2} /in, Mass of loaded HSM plus two shield walls. (Conservatively, a lower DSC weight of 72 kips is used.)
d	=	156 in., CG Elevation of the missile above the basemat.
I_A	=	20,075,643 lb-sec^2-in, mass moment of inertia of loaded HSM plus one shield wall about point A

Substituting and solving for V and ω_A produces an initial linear velocity of 1.364 ft./sec. and an angular velocity of 0.18 radians/sec.

The actual ratio between HSM sliding and rotation depends on where the missile impacts the shield wall. A low elevation impact produces mainly sliding while a high elevation impact produces mainly rotation.

For an impact at the bottom of the HSM wall, the kinetic energy imparted to the HSM is absorbed by sliding friction between the concrete of the HSM and the basemat. ACI 349-85 (8.20) recommends a coefficient of friction of 0.6. Assuming that the missile impulse load results in sliding of the HSM and equating the kinetic energy to the sliding friction gives:

$$(M + m)\ \mu g\ \Delta = \frac{1}{2}(M + m)\ V^2 \qquad (8.2\text{-}13)$$

Where:

μ	=	0.6, Coefficient of friction
Δ	=	Linear distance module slides

M, m, g, and V are as defined above

Substituting gives $\Delta = 0.58$ inch

Therefore, a single free-standing module slides a maximum distance of 5/8 inch due to a low elevation tornado missile impact. In an HSM array, the impact force is transmitted to adjacent HSMs through the spacer channels at the base of each HSM. The total velocity decreases very rapidly with the mass of additional modules. A three module array for example will slide a maximum of 0.12 inch. This assumes that all the energy is absorbed in sliding.

At the opposite extreme, when the massive missile impacts at the top of the shield wall most of the missile energy is absorbed in rotation of the shield wall and HSM.

Immediately following missile impact, the tie plates holding the top of the shield wall, which are flexible compared with the adjoining concrete, collapse and the shield wall acts

as a slab simply supported on two edges. Conservatively neglecting the energy absorbed in collapsing the shield wall tie plates and equating the initial kinetic energy of the HSM to the increase in potential energy as the HSM center of gravity rises due to rotation gives:

Loss of Kinetic Energy = Increase in Potential Energy

$$\frac{1}{2}\ I_A \omega_A^2 = Mgd\,[\cos(\beta + \alpha - \frac{\pi}{2}) - \cos\beta] \qquad (8.2\text{-}14)$$

Where:

α and β are defined in Figure 8.2-11

M, g, d, I_A, and ω_A are as defined above

Substituting and solving for α shows that the HSM rotates a maximum of 1.12° about the bottom edge opposite the point of impact. Therefore, the HSM and shield wall provide a stable body as tip over does not occur until the c.g. rotates past the edge (point A in Figure 8.2-11) to an angle of more than 30.5°.

The impact force applied to the shield wall by the massive missile and the behavior of the wall is calculated in accordance with Bechtel Topical Report, "Design of Structures for Missile Impact," BC-TOP-9A (8.51).

The maximum force due to the automobile is given by:

$$F = 0.625\ v_i\ m\ g \qquad (8.2\text{-}15)$$

Where:

v_i, m, and g are defined above.

Substituting for the design basis massive missile parameters, the maximum force due to missile impact is 487.5 kips applied to the end module shield wall. This force is sufficient to collapse the tie plates provided to stabilize the shield wall. The spacer channel at the base of the module is assumed not to crush. The shield wall then acts as a simply supported slab on the top and bottom edges. For design of the shield wall reinforcement, the missile load is applied at the mid-height of the shield wall and a simple yield line failure is considered for the full width of the wall.

For a simply supported slab with a hinge at the slab centerline, the required moment capacity is 1155 in-k/ft. The design capacity of the end shield wall is 2109 in-k/ft., which exceeds the required capacity per ACI 349-85 (8.20).

Thus, loss of bending strength of the shield wall due to a tornado missile impact is acceptable and does not affect the safe operation of the HSM. Recovery from this event can be performed in a planned and deliberate manner to replace the shield wall and tie plates. This requires temporary shielding during removal and replacement of the wall, or removal of the HSM from service. At no time is there any danger of a release of radioactive materials to the general public.

8.2.2.3 Accident Dose Calculation

Each exposed component of the NUHOMS® system is specifically designed to withstand tornado generated missiles as discussed in the preceding paragraphs. The consequence of reduced shielding effects of adjacent HSMs is presented in Section 8.2.1.

8.2.2.4 Transfer Cask Missile Impact Analysis

The effects of a tornado missile impact on the loaded NUHOMS® transfer cask have been addressed in previous licensing correspondence. These documents are included in Appendix C for ease of reference.

8.2.3 Earthquake

8.2.3.1 Cause of Accident

As discussed in Section 3.2.3, enveloping design basis seismic forces are assumed to act on the NUHOMS® system components. For this conservative generic evaluation, the design response spectra of NRC Regulatory Guide 1.60 (8.35) are selected for the seismic analysis of the NUHOMS® system components.

8.2.3.2 Accident Analysis

As discussed in Section 3.2.3, and shown in Figure 8.2-2, the peak horizontal ground acceleration of 0.25g and the peak vertical ground acceleration of 0.17g are utilized for the design basis seismic analysis of the NUHOMS® components. Based on NRC Reg. Guide 1.61 (8.36), a damping value of three percent is used for the DSC seismic analysis. Similarly, a damping value of seven percent for DSC support steel and concrete is utilized for the HSM. An evaluation of the frequency content of the loaded HSM is performed to determine the dynamic amplification factors associated with the design basis seismic response spectra for the NUHOMS® HSM and DSC. The dominant structural frequencies calculated for a loaded HSM in the lateral direction are 19.1 Hz and 30.4 Hz for the DSC on the support structure and HSM concrete structure, respectively. Table 1 of NRC Regulatory Guide 1.60 requires amplification factors for these structural frequencies, which result in conservative horizontal accelerations of 0.40g. The dominant vertical frequencies of the loaded HSM exceed 33 Hz, corresponding to the zero period acceleration of 0.17g vertical.

(ii) HSM Seismic Response Spectrum Analysis

The horizontal and vertical seismic response spectra are applied to the HSM. The horizontal response spectrum is applied in two orthogonal horizontal directions. The response spectra are obtained from Regulatory Guide 1.60 (8.35) at 7% damping, factored by the 0.25g horizontal and 0.17g vertical peak ground accelerations. The horizontal and vertical response spectra utilized in the analysis are shown in Figure 8.2-2. The HSM concrete mass participating in each mode is multiplied by the corresponding spectral acceleration value to determine the applied loads. The mass of the DSC and DSC support structure are also included in the HSM analytical model. The resulting forces and moments in the HSM walls, roof and floor of a single HSM are calculated using the linear finite element modal shown in Figure 8.1-21, and the computer program ANSYS (8.49). The model responses are combined in accordance with Regulatory Guide 1.92 (8.62) using the grouping method for closely spaced modes. The directional responses are then combined by the square root of the sum of the squares (SRSS) method. The combined maximum moments and stress are reported in Table 8.2-3.

(iii) HSM Overturning Due to Seismic

The following conservative analysis is performed to show that a single, free-standing HSM will not overturn due to seismic loads. The HSM stabilizing moment (M_{st}) is:

$$M_{st} = Wd = 19{,}959 \text{ K-in.} \qquad (8.2\text{-}22)$$

Where: W and d are as defined in equation 8.2-1.

The seismic overturning moment is:

$$M_{ot} = Wa_v d + Wa_h h = 17{,}076 \text{ K-in.} \qquad (8.2\text{-}23)$$

Where: M_{ot} = Overturning moment

a_v = 0.17g, Maximum vertical seismic acceleration

a_h = 0.40g, Maximum horizontal seismic acceleration

h = 99.1 in., Vertical height from HSM and DSC center of gravity to base

The result of this analysis indicates that a single free-standing HSM will not overturn during a seismic event. The margin of safety against overturning is 1.17.

(iv) HSM Sliding Due to Seismic

To show that a single free-standing HSM will not slide due to the postulated horizontal and vertical seismic accelerations, the following conservative analysis is performed. The friction force resisting sliding (F_{s1}) is:

F_{s1}	=	$W\mu g$ = 171.4 kips	(8.2-24)
W	=	HSM loaded weight = 345 kips	
μ	=	Coefficient of friction between the HSM concrete walls and the floor slab foundation = 0.6	
g	=	Net downward gravitational force	
	=	(1 - 0.17)g or 0.83g	

The applied horizontal seismic force is:

	F_{hs}	=	Wa_H = 138.0 kips
Where:	F_{hs}	=	Induced horizontal seismic force
	a_H	=	0.40g, Horizontal seismic acceleration

The force required to slide the HSM is larger than the resulting lateral seismic force and therefore, the HSM will not slide. The factor of safety against sliding is 1.24.

C. DSC Support Structure Seismic Evaluation

(i) DSC Support Structure Natural Frequency

The lowest structural frequency of the DSC support structure inside the HSM is dominated by the mass of the DSC. The DSC and support structure are included in the HSM analytical model. The dominant horizontal and vertical frequencies of the DSC/DSC support structure are 19.1 Hz and 41.34 Hz, respectively.

(ii) DSC Support Structure, Seismic Response Spectra Analysis

The horizontal and vertical seismic response spectra accelerations are applied to the support structure as previously described for the HSM. The modal summations and directional summations are also the same. For the support frame cross members, the maximum bending stress is 4.1 ksi and the maximum shear stress is 5.27 ksi. Similarly, the maximum stresses in the support rails are 1.19 ksi and 2.01 ksi, respectively. These

compare with Code allowables of 34.1 ksi for bending and 18.1 ksi for shear and, as a result, have a considerable design margin.

The effect of concentrated anchor bolt forces is included in the design of the DSC support structure connection details. Similarly, each connection of the support rails to the support frame cross members is designed for the resulting seismic loads. This condition envelopes all other loading conditions for the individual bolts or structural elements of the DSC support structure.

The stresses in the support frame columns, cross members and rails due to seismic accelerations are included in the subsequent load combination results reported in Section 8.2.10.

<u>(iii) DSC Axial Retainer Analysis</u>

The DSC axial retainer detail, located inside the HSM access opening, is shown on the Appendix E drawings. The retainer bears against the bottom of the DSC shell and transfers the seismic load to the embedded tube.

The clearance between the DSC axial retainer and the DSC is designed for the maximum DSC thermal growth that occurs during the postulated HSM blocked vent case, as discussed in Section 8.2.7.

The DSC will be subjected to maximum seismic accelerations equal to the rigid range spectral acceleration of 0.40g horizontal. The seismic load acting on the axial retainer is computed as follows:

$$P = 1.5W\{S_a - \mu(1-S_v)/\text{Cos}(30)\} \qquad (8.2\text{-}25)$$

Where,

P = seismic load acting on the axial retainer, kips

W = DSC weight, assumed to be 102 kips

S_a = horizontal rigid range spectral acceleration of 0.40g

S_v = vertical rigid range spectra acceleration = 0.17g

μ = Coefficient of friction between DSC support rail and DSC = 0.25

1.5 = Impact factor

$$P = (1.5)(102 \text{ kips})\{(0.40g) - (0.25)(0.83)/\text{Cos}30\} = 24.6 \text{ kips}$$

The maximum shear and bending stresses in the DSC axial retainer are 6.2 ksi and 8.1 ksi, respectively. The allowable shear and bending stresses are 23.5 ksi and 44.3 ksi, respectively. Therefore, the DSC axial retainer stresses are within allowable values.

D. Transfer Cask Seismic Evaluation

The effects of a seismic event occurring when a loaded DSC is resting inside the transfer cask are conservatively postulated for two conditions that affect the transfer cask. All other conditions that exist during DSC loading or transport operations are enveloped by the two cases postulated. The first case postulates a fully loaded transfer cask standing vertically in the plant's cask decontamination area during closure of the DSC. For this condition it is required that the transfer cask remain upright. The rigid body horizontal acceleration required to overturn a loaded transfer cask at a minimum gross weight of 190 kips is at least 0.40g. Each licensee shall ensure that the transfer cask is not subjected to accelerations greater than this magnitude while in the plant's decontamination area, or provide sufficient lateral restraint to prevent cask overturning.

The second case postulates a seismic event occurring during transport of a loaded DSC, resting inside the transfer cask, in a horizontal position, secured to the support skid/transport trailer. For the standardized cask this load case is conservatively enveloped by the postulated normal transport load accelerations of ± 0.5g acting in the vertical, axial, and transverse directions, applied simultaneously at the center of gravity of the transfer cask, as specified in Section 8.1.1.8. These accelerations envelop those that would result from a seismic event in the highly unlikely event that a design basis earthquake would occur during transport of the loaded DSC to or from the HSM. Therefore, the calculated stress intensities for the normal transport loads case for the cask structural shell are conservatively used as the standardized cask maximum seismic stresses in the load combination results reported in Section 8.2.10.

The analysis of the OS197 and OS197H casks applies peak amplification factors of 3.5 and 3.3 to the 0.25g and 0.17g peak ground spectral accelerations in the horizontal and vertical directions, respectively. A multimode factor of 1.5 is applied, resulting in accelerations of 1.31g and 0.84g in the horizontal and vertical directions, respectively.

The stabilizing moment to prevent overturning of the cask/trailer assembly due to the 0.25g horizontal and 0.17g vertical seismic ground accelerations is calculated and compared to the dead weight stabilizing moment. The results of this analysis show that there is a factor of safety of at least 2.0 against overturning that ensures that the cask/trailer assembly has sufficient margin for the design basis seismic loading.

(i) Transfer Cask Analysis Methodology

The ANSYS axisymmetric finite element models used to perform these analyses are described in Section 8.2.5.2. The loadings due to the postulated vertical drops are applied to the transfer cask analytical models in a symmetric manner. The individual models of the top and bottom cask regions shown in Figure 8.2-9 and Figure 8.2-10 are used for these analyses. The respective top or bottom impacted surface of the transfer cask is assumed to be uniformly supported vertically and the 75g equivalent static decelerations are applied to the models.

(ii) Transfer Cask Stress Analysis

The resulting primary membrane and membrane plus bending stresses due to the postulated end drop are tabulated in Table 8.2-9, Table 8.2-9a, and Table 8.2-9b for the standardized, OS197, and OS197H transfer casks, respectively. For the top end drop analysis, the stresses in the cover plates are relatively small since they arise primarily from bearing of the DSC and its contents on the cask top cover plate. The most critical vertical drop direction for the transfer cask top region is the bottom end drop, since this produced the maximum bending stress in the top cover plate. The maximum (enveloped between the standardized and OS197 casks) primary membrane plus bending stress in the cover plate is 28.6 ksi (36.8 ksi for the OS197H transfer cask). The maximum local membrane stresses in the cask structural shell and inner liner are 9.6 ksi and 12.9 ksi, respectively (the same for the OS197 and OS197H transfer cask). Similarly for the standardized transfer cask bottom region, the most critical drop direction is the top end drop producing a maximum primary membrane plus bending stress of 26.1 ksi in the cask bottom end plate (28.6 ksi and 34.8 ksi for the OS197 and OS197H transfer cask, respectively). These stresses are well below the appropriate ASME Code Service Level D allowables.

(iii) Transfer Cask Collar Analysis

During the postulated vertical end drop, the maximum compressive stress in the cask collar is 13 ksi and the maximum membrane stress intensity is 26.2 ksi. These stresses are well within the ASME Code Service Level D allowables.

E. On-site Transfer Cask/DSC Corner Drop Analyses

The possibility of a drop onto the top or bottom end corners of the transfer cask is extremely remote due to the limited cask handling operations of the NUHOMS® system, as discussed previously. Nevertheless, for this generic evaluation, a cask corner drop is conservatively postulated to occur onto a concrete surface with an equivalent static deceleration of 25g. The orientation of the drop is shown in Figure 8.2-3 as occurring at 30° to the horizontal. This is the largest drop orientation angle that can occur as the center of gravity of the cask passes beyond the back end of the transport trailer and pitches downward. The derivation of this load definition is contained in Appendix C.2.

It is probable that the cask support skid would remain firmly attached to the cask and would absorb considerable energy upon impact, thus reducing the transfer cask deceleration. In addition, this would further reduce the angle of the impact and the drop height. The combined support skid and transfer cask would act as a substantial energy absorbing mechanism thus significantly reducing the effects of impact loads on the DSC and the spent fuel assemblies. Also, for the postulated case of the cask sliding forward, the cask and skid may initially impact the tractor vehicle, prior to pitching onto the ground, with significant reductions in the resulting impact velocity and the energy imparted to the transfer cask and its contents.

<u>(i) Cask/DSC Analysis Methodology</u>

The combined transfer cask/DSC ANSYS linear elastic axisymmetric models used in the side drop and the end drop analyses as shown in Figure 8.2-9 and Figure 8.2-10, are used for the corner drop analyses. The postulated transfer cask corner drop accident results in a very complex loading function because it involves both symmetric and asymmetric load components in both the vertical and horizontal directions. The analysis involved the development of the impact force and the content loading and applying these loads to the axisymmetric model as Fourier harmonics. A complete description of the load development and application of the loads to the ANSYS models is provided in Appendix C.2.

<u>(ii) Cask/DSC Stress Analysis</u>

The resulting local primary membrane and primary bending stresses in the transfer cask due to both the postulated top and bottom corner drop analysis are tabulated in Table 8.2-9, Table 8.2-9a, and Table 8.2-9b for the standardized, OS197, and OS197H transfer casks, respectively. The resulting stresses in the DSC due to a cask corner drop are evaluated and found to be enveloped by those calculated for the 75g end and side drop analyses. As seen from the results, the DSC and transfer cask stress intensities are within the appropriate ASME Code Service Level D allowable limits.

<u>(iii) Transfer Cask Collar Analysis</u>

During the postulated oblique corner drop, the shear forces from the cask lid/DSC/collar are transferred to the cask body by the 3 inch thick shield ring of the cask collar. The tensile forces associated with the corner drop are transferred from the cask collar to the cask body by 16, 1-3/4-inch diameter bolts. The maximum calculated collar membrane stress intensity is 3.2 ksi and the maximum bolt stress is 74.3 ksi. These stresses are well within the ASME Code Service Level D allowables.

8.2.5.3 <u>Loss of Neutron Shield</u>

This accident conservatively postulates loss of neutron shield on the OS197 transfer cask.

Table 8.2-2
Comparison of Total Dose Rates for HSM
with and without Adjacent HSM Shielding Effects

Distance (meters) from Nearest HSM Wall, 2x10 Array	Normal Case Dose Rate[1] (mrem/hr)	Accident Case Dose Rate[1] (mrem/hr)
10	9.0	18
100	0.2	0.4
500	2.1×10^{-3}	4.2×10^{-3}
1000	1.5×10^{-4}	3.0×10^{-4}

(1) Air scattered plus direct radiation.

Table 8.2-3
Maximum HSM Reinforced Concrete Bending Moments and Shear Force for Accident Loads

Structural Section	Force[5] Component	HSM Internal Forces (kip/ft., in.-k/ft.)[1]				
		Tornado Winds	Tornado[3] Missile	Seismic	Flooding	Blocked Vents Thermal[2]
Floor Slab	Shear	3.54	8.68	1.15	3.31	2.74
	Moment	21.98	54.11	6.63	23.08	36.7
Side Wall	Shear	8.72	25.0[4]	7.74	9.53	8.64
	Moment	68.85	118.48	15.89	65.95	266.3
Front Wall	Shear	2.45	38.21	15.53	2.15	17.87
	Moment	74.16	644.65	36.67	55.03	485.84
Rear Wall	Shear	1.61	11.14	1.41	0.81	8.65
	Moment	15.17	36.06	5.01	13.34	169.8
Roof Slab	Shear	2.40	35.05	0.93	0.96	3.22
	Moment	54.89	521.51	6.02	48.09	305.8

(1) Maximum loads shown are irrespective of location.

(2) Maximum moments are calculated using cracked section properties.

(3) The maximum shear on the HSM rear shield wall for the DBT missile is 487.5 kips. The shield wall capacity for punching shear is calculated based on ACI-349 Section 11.11.2.1, and is 1598 kips.

(4) The maximum shear due to tornado missile is the maximum stress d/2 from the back wall inner face.

(5) Out-of-plane shears and moments.

Table 8.2-18
HSM Enveloping Load Combination Results

Load[1] Combination	Loading Combination Description	Governing Load[2][5]		Capacities	
		V_{max} (k/ft.)	M_{max} (k-in./ft.)	V_c (k/ft.)[4]	M_u (k-in./ft.)
1	1.4D + 1.7L	19.09	214.96	40.5	881.00
2	1.4D + 1.7L + 1.7H	19.09	214.96	40.5	881.00
3	0.75(1.4D + 1.7L + 1.7H + 1.7T + 1.7W)	16.61	355.00	22.90	881.00
4	0.75(1.4D + 1.7L + 1.7H + 1.7T)	22.24	304.00	40.50	881.00
5	D + L + H + T + E	27.53	347.77	40.50	881.00
6	D + L + H + T + F	9.67	148.57	22.90	881.00
7[6]	D + L + H + T_a[3]	28.93	533.00	38.40	788.00
8	D + L + H + T + WT	42.73	681.63	44.00	881.00

D = Dead Weight
F = Flood Induced Loads
L = Live Load
T_a = Off-normal or Accident Condition Thermal Load
WT = Tornado Wind and Missile Load
E = Earthquake Load
H = Lateral Soil Pressure Load,
T = Normal Condition Thermal Load
W = Design Basis Wind Load

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(1) Load combinations are based on ANSI-57.9 as shown in Table 3.2-5.

(2) Loads reported have minimum margin to design capacity.

(3) Thermal accident load (T_a) is based on -40°F ambient with air inlets and outlets blocked (See Section 8.1.2.2) for either 40 hours or 5 days (Results of these two blocked vent cases are enveloped).

(4) The shear capacity V_c is calculated using Equation 11-3 of ACI 349-85.

(5) Results of load combinations 3 through 7 are based on cracked section. Others based on uncracked sections.

(6) Material properties taken at 479°F for load combination 7.

Table 8.2-19
<u>DSC Support Structure Enveloping Load Combination Results</u>

Component	Load Combination	Calculated Stress				Interaction (Calc/ Allowable)	Allowable Shear Stress (ksi)
		Axial (ksi)	Strong Axis Bending (ksi)	Weak Axis Bending (ksi)	Shear (ksi)		
Column	Normal Operation $DW_s + DW_c + HL_f + T_n$	3.11	2.85	4.10	0.24	0.55	18.1
	Off-Normal Operation $DW_s + HL_j$	2.9	0.72	3.9	0.22	0.37	19.1
	Accident $DW_s + DW_c + HL_f + DBE + T_n$	6.6	4.2	4.4	0.51	0.87	18.1
	Accident $DW_s + DW_c + T_a$	3.9	10.9	12.6	0.7	0.96	15.3
Cross Beam	Normal Operation $DW_s + DW_c + HL_f + T_n$	1.06	3.81	6.85	3.81	0.53	18.1
	Off-Normal Operation $DW_s + HL_j$	0.12	1.7	6.1	6.5	0.31	19.1
	Accident $DW_s + DW_c + HL_f + DBE + T_n$	1.87	7.04	9.25	14.63	0.88	18.1
	Accident $DW_s + DW_c + T_a$	2.59	8.4	20.6	7.1	0.96	15.3

Table 8.2-21
Standardized Transfer Cask Enveloping Load Combination Results
for Normal and Off-Normal Loads (ASME Service Levels A and B)

Transfer Cask Component	Stress Type	Controlling Load Combination(1)	Stress (ksi)	
			Calculated	Allowable(2)
Structural Shell	Primary Membrane	A4	1.2	21.7
	Membrane + Bending	A4	5.0	32.6
	Primary + Secondary	A4	61.9	65.1
Top Cover Plate	Primary Membrane	A1	0.2	21.7
	Membrane + Bending	A4	7.1	32.6
	Primary + Secondary	A4	20.2	65.1
Bottom End Plate	Primary Membrane	A1	0.2	21.7
	Membrane + Bending	A4	15.6	32.6
	Primary + Secondary	A4	30.3	65.1

(1) See Table 3.2-7 for load combination nomenclature.

(2) See Table 3.2-11 for allowable stress criteria. Material properties were obtained from Table 8.1-3 at a design temperature of 400°F.

Table 8.2-21a

OS197 Transfer Cask Enveloping Load Combination Results for Normal and Off-Normal Loads (ASME Service Levels A and B)

Transfer Cask Component	Stress Type	Stress (ksi)[1]	
		Calculated[1]	Allowable[2]
Structural Shell	Primary Membrane	1.8/7.9[4]	20.0
	Membrane + Bending	14.0/19.5[4]	30.0
	Primary + Secondary	25.4/41.4[4]	60.0
Top Cover Plate[3]	Primary Membrane	0.56	18.7
	Membrane + Bending	4.2	28.1
	Primary + Secondary	10.8	56.1
Bottom End Plate	Primary Membrane	0.56	18.7
	Membrane + Bending	7.2	28.1
	Primary + Secondary	14.0	56.1

(1) The load combination for Levels A and B is dead weight plus thermal plus handling loads.

(2) See Table 3.2-11 for allowable stress criteria. Material properties for all components except the cask structural shell were obtained from Table 8.1-3 at a design temperature of 400°F. The cask structural shell allowables are based on a temperature of 250°F.

(3) Allowable stress values and calculated stress intensities are tabulated for the stainless steel cover plate.

(4) The leftmost stress value listed is for locations remote from the trunnions, while the rightmost stress value occurs in the region of the trunnions.

Table 8.2-21b
OS197H Transfer Cask Enveloping Load Combination Results for Normal and Off-Normal Loads (ASME Service Levels A and B)

Transfer Cask Component	Stress Type	Stress (ksi)[1]	
		Calculated[1]	Allowable[2]
Structural Shell	Primary Membrane	2.1/9.45[4]	20.0
	Membrane + Bending	14.9/21.8[4]	30.0
	Primary + Secondary	25.1/46.8[4]	60.0
Top Cover Plate[3]	Primary Membrane	0.7	18.7
	Membrane + Bending	5.3	28.1
	Primary + Secondary	11.2	56.1
Bottom End Plate	Primary Membrane	1.2	18.7
	Membrane + Bending	8.6	28.1
	Primary + Secondary	14.6	56.1

(1) The load combination for Levels A and B is dead weight plus thermal plus handling loads.

(2) See Table 3.2-11 for allowable stress criteria. Material properties for all components except the cask structural shell were obtained from Table 8.1-3 at a design temperature of 400°F. The cask structural shell allowables are based on a temperature of 250°F.

(3) Allowable stress values and calculated stress intensities are tabulated for the stainless steel cover plate.

(4) The leftmost stress value listed is for locations remote from the trunnions, while the rightmost stress value occurs in the region of the trunnions.

Table 8.2-22
<u>Standardized Transfer Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level C)</u>

Transfer Cask Component	Stress Type	Controlling Load Combination[1]	Stress (ksi)	
			Calculated	Allowable[2]
Structural Shell	Primary Membrane	C1	1.7	26.0
	Membrane + Bending	C1	9.1	39.1
Top Cover Plate	Primary Membrane	C1	0.2	26.0
	Membrane + Bending	C1	14.0	39.1
Bottom End Plate	Primary Membrane	C1	0.1	26.0
	Membrane + Bending	C1	31.0	39.1

(1) See Table 3.2-7 for load combination nomenclature.

(2) See Table 3.2-11 for allowable stress criteria. Material properties were obtained from Table 8.1-3 at a design temperature of 400°F.

Table 8.2-22a

OS197 Transfer Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level C)

Transfer Cask Component	Stress Type	Stress (ksi)	
		Calculated[1]	Allowable[2]
Structural Shell	Primary Membrane	3.4/8.1[4]	24.0
	Membrane + Bending	27.6/19.5[4]	36.0
Top Cover Plate[3]	Primary Membrane	0.56	22.4
	Membrane + Bending	4.5	33.7
Bottom End Plate	Primary Membrane	1.1	22.4
	Membrane + Bending	15.4	33.7

(1) The load combination for Level C include dead weight, thermal, handling and seismic loads

(2) See Table 3.2-11 for allowable stress criteria. Material properties for all components except the cask structural shell were obtained from Table 8.1-3 at a design temperature of 400°F. The cask structural shell allowables are based on a temperature of 250°F.

(3) Allowable stress values and calculated stress intensities are tabulated for the stainless steel cover plate.

(4) The lower stress value listed is for locations remote from the trunnions, while the higher stress value occurs in the region of the trunnions.

Table 8.2-22b
OS197H Transfer Cask Enveloping Load Combination Results
for Accident Loads (ASME Service Level C)

Transfer Cask Component	Stress Type	Stress (ksi)	
		Calculated(1)	Allowable(2)
Structural Shell	Primary Membrane	3.9/9.45(4)	24.0
	Membrane + Bending	28.6/21.8(4)	36.0
Top Cover Plate(3)	Primary Membrane	0.7	22.4
	Membrane + Bending	5.8	33.7
Bottom End Plate	Primary Membrane	2.9	22.4
	Membrane + Bending	22.3	33.7

(1) The load combination for Level C include dead weight, thermal, handling and seismic loads.

(2) See Table 3.2-11 for allowable stress criteria. Material properties for all components except the cask structural shell were obtained from Table 8.1-3 at a design temperature of 400°F. The cask structural shell allowables are based on a temperature of 250°F.

(3) Allowable stress values and calculated stress intensities are tabulated for the stainless steel cover plate.

(4) The leftmost stress value listed is for locations remote from the trunnions, while the rightmost stress value occurs in the region of the trunnions. The maximum stresses in the shell near the trunnions for Service Level C were evaluated against Service Level A/B allowables in Table 8.2-21a.

Table 8.2-23

Standardized Transfer Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level C)

Transfer Cask Component	Stress Type	Controlling Load Combination(1)	Stress (ksi)	
			Calculated	Allowable(2)
Structural Shell	Primary Membrane	C1	1.7	26.0
	Membrane + Bending	C1	9.1	39.1
Top Cover Plate	Primary Membrane	C1	0.2	26.0
	Membrane + Bending	C1	14.0	39.1
Bottom End Plate	Primary Membrane	C1	0.1	26.0
	Membrane + Bending	C1	31.0	39.1

(1) See Table 3.2-7 for load combination nomenclature.

(2) See Table 3.2-11 for allowable stress criteria. Material properties were obtained from Table 8.1-3 at a design temperature of 400°F.

Table 8.2-23a
OS197 Transfer Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level D)

Transfer Cask Component	Stress Type	Stress (ksi)	
		Calculated(1)	Allowable(2)
Structural Shell	Primary Membrane	33.4	48.0
	Membrane + Bending	39.9	68.5
Top Cover Plate(3)	Primary Membrane	28.6	44.9
	Membrane + Bending	28.6	64.4
Bottom End Plate	Primary Membrane	27.0	44.9
	Membrane + Bending	45.5	64.4

(1) The load combination for Level D include dead weight, thermal, handling and cask drop loads.

(2) See Table 3.2-11 for allowable stress criteria. Material properties for all components except the cask structural shell were obtained from Table 8.1-3 at a design temperature of 400°F. The cask structural shell allowables are based on a temperature of 250°F.

(3) Allowable stress values and calculated stress intensities are tabulated for the stainless steel cover plate.

Table 8.2-23b

OS197H Transfer Cask Enveloping Load Combination Results for Accident Loads (ASME Service Level D)

Transfer Cask Component	Stress Type	Stress (ksi)	
		Calculated(1)	Allowable(2)
Structural Shell	Primary Membrane	43.1	48.0
	Membrane + Bending	51.5	68.5
Top Cover Plate(3)	Primary Membrane	36.8	44.9
	Membrane + Bending	36.8	64.4
Bottom End Plate	Primary Membrane	34.8	44.9
	Membrane + Bending	58.6	64.4

(1) The load combination for Level D include dead weight, thermal, handling and cask drop loads.

(2) See Table 3.2-11 for allowable stress criteria. Material properties for all components except the cask structural shell were obtained from Table 8.1-3 at a design temperature of 400°F. The cask structural shell allowables are based on a temperature of 250°F.

(3) Allowable stress values and calculated stress intensities are tabulated for the stainless steel cover plate.

Table 8.2-24
Expanded Load Combinations for DSC Analyses

	Horizontal DW		Vertical DW		Internal	External	Thermal	Lifting	Other	Service
	DSC	Fuel	DSC	Fuel	Pressure(9)	Pressure	Condition	Loads	Loads	Level
FUEL LOADING LOAD CASES										
FL-1 DSC/Cask Filling	–	–	Cask	–	–	Hydrostatic	100°F Cask	–	–	A
FL-2 DSC/Cask Filling	–	–	Cask	–	Hydrostatic	Hydrostatic	100°F Cask	–	–	A
FL-3 DSC/Cask Xfer	–	–	Cask	–	Hydrostatic	Hydrostatic	100°F Cask	–	–	A
FL-4 Fuel Loading	–	–	Cask	X	Hydrostatic	Hydrostatic	100°F Cask	–	–	A
FL-5 Xfer to Decon	–	–	Cask	X	Hydrostatic	Hydrostatic	100°F Cask	–	–	A
FL-6 Inner Cover Plate Welding	–	–	Cask	X	Hydrostatic	Hydrostatic	100°F Cask	–	–	A
FL-7 Fuel Deck Seismic Loading	–	–	Cask	X	Hydrostatic	Hydrostatic	100°F Cask	–	Note 10	C
DRAINING AND DRYING LOAD CASES										
DD-1 DSC Blowdown	–	–	Cask	X	Hydrostatic + 20 psi	Hydrostatic	100°F Cask	–	–	A
DD-2 Vacuum Drying	–	–	Cask	X	0 psia	Hydrostatic + 14.7 psi	100°F Cask	–	–	A
DD-3 Helium Backfill	–	–	Cask	X	12 psi	Hydrostatic	100°F Cask	–	–	A
DD-4 Final Helium Backfill	–	–	Cask	X	5.0 psi	Hydrostatic	100°F Cask	–	–	A
DD-5 Outer Cover Plate Welding	–	–	Cask	X	5.0 psi	Hydrostatic	100°F Cask	–	–	A
TRANSFER TRAILER LOADING										
TL-1 Vertical Xfer to Trailer			Cask	X	11.0 psi(18)	–	0°F Cask	–	–	A
TL-2 "			Cask	X	11.0 psi(18)	–	100°F Cask	–	–	A
TL-3 Laydown	Cask	X			11.0 psi(18)	–	0°F Cask	–	–	A
TL-4 "	Cask	X			11.0 psi(18)	–	100°F Cask	–	–	A

	Horizontal DW		Vertical DW		Internal	External	Thermal	Handling	Other	Service
	DSC	Fuel	DSC	Fuel	Pressure(9)	Pressure	Condition	Loads	Loads	Level
TRANSFER TO / FROM ISFSI										
TR-1 Axial Load - Cold	Cask	X	–	–	11.0 psi(18)	–	0°F Cask	1g Axial	–	A
TR-2 Transverse Load - Cold	Cask	X	–	–	11.0 psi(18)	–	0°F Cask	1g Transverse	–	A
TR-3 Vertical Load - Cold	Cask	X	–	–	11.0 psi(18)	–	0°F Cask	1g Vertical	–	A
TR-4 Oblique Load - Cold	Cask	X	–	–	11.0 psi(18)	–	0°F Cask	½g Axial + ½g Trans + ½g Vert	–	A
TR-5 Axial Load - Hot	Cask	X	–	–	11.0 psi(18)	–	100°F Cask	1g Axial	–	A
TR-6 Transverse Load - Hot	Cask	X	–	–	11.0 psi(18)	–	100°F Cask	1g Transverse	–	A
TR-7 Vertical Load - Hot	Cask	X	–	–	11.0 psi(18)	–	100°F Cask	1g Vertical	–	A
TR-8 Oblique Load - Hot	Cask	X	–	–	11.0 psi(18)	–	100°F Cask	½g Axial + ½g Trans + ½g Vert	–	A
TR-9 75g Top End Drop	–	–	Note 1		60 psi(11)	–	100°F Cask(2)	–	75g TED	D
TR-10 75g Bottom End Drop	–	–	Note 1		60 psi(11)	–	100°F Cask(2)	–	75g BED	D
TR-11 75g Side Drop	Note 1		–	–	60 psi(11)	–	100°F Cask(2)	–	75g Side Drop	D

Notes:

1. 75g drop acceleration includes gravity effects. Therefore, it is not necesary to add an additional 1.0g load.
2. For level D events, only the maximum temperature case is considered. (Thermal stresses are not limited for Level D events and maximum temperatures give minimum allowables).
3. Flood load is an external pressure equivalent to 50 ft. of water.
4. BV = HSM Vents are blocked
5. At temperatures over 100°F, a sunshade is required over the Transfer Cask. Temperatures for these cases are enveloped by the 100°F (without sunshade) case.
6. This pressure assumes release of the fuel cover gas and 30% of the fission gas. Since unloading requires the HSM door to be removed, the pressure and temperatures are based on the normal (unblocked vent) condition. Pressure is applied to the <u>inner</u> pressure boundary.
7. This pressure assumes release of the fuel cover gas and 30% of the fission gas. Although unloading requires the HSM door to be removed, the pressure and temperatures are conservatively based on the blocked vent condition. Pressure is applied to the <u>outer</u> pressure boundary.

Notes continued on following page...

Table 8.2-24
Expanded Load Combinations for DSC Analyses
(concluded)

HSM LOADING	Horizontal DW DSC	Horizontal DW Fuel	Vertical DW DSC	Vertical DW Fuel	Internal Pressure[9]	External Pressure	Thermal Condition	Handling Loads	Other Loads	Service Level
LD-1 Normal Loading - Cold	Cask	X	--	--	11.0 psi[18]	--	0°F Cask	+80 Kip	--	A
LD-2 Normal Loading - Hot	Cask	X	--	--	11.0 psi[18]	--	100°F Cask	+80 Kip	--	A
LD-3	Cask	X	--	--	11.0 psi[18]	--	125°F w/shade[illegible]	+80 Kip	--	A
LD-4 Off-Normal Loading - Cold	Cask	X	--	--	11.0 psi[18]	--	0°F Cask	+80 Kip	--	B
LD-5 Off-Normal Loading - Hot	Cask	X	--	--	11.0 psi[18]	--	100°F Cask	+80 Kip	--	B
LD-6	Cask	X	--	--	11.0 psi[18]	--	125°F w/shade[illegible]	+80 Kip	--	B
LD-7 Accident Loading	Cask	X	--	--	11.0 psi[18]	--	125°F w/shade[illegible]	+80 Kip	--	B
HSM STORAGE										
HSM-1 Off-Normal Storage	HSM	X	--	--	11.0 psi[18]	--	-40°F HSM	--	--	B
HSM-2 Normal Storage	HSM	X	--	--	11.0 psi[18]	--	0°F HSM	--	--	A
HSM-3 Off-Normal Storage	HSM	X	--	--	11.0 psi[18]	--	125°F HSM	--	--	B
HSM-4 Off-Normal Temp. + Failed Fuel	HSM	X	--	--	11.0 psi[18]	--	125°F HSM	--	Failed Fuel	C
HSM-5 Blocked Vent Storage	HSM	X	--	--	60 psi[8,12]	--	125°F HSM/BV[4]	--	--	D
HSM-6 B. V. + Failed Fuel Storage	HSM	X	--	--	60 psi[8,13]	--	125°F HSM/BV[4]	--	Failed Fuel	D
HSM-7 Earthquake Loading - Cold	HSM	X	--	--	11.0 psi[18]	--	0°F HSM	--	Seismic	C
HSM-8 Earthquake Loading - Hot	HSM	X	--	--	11.0 psi[18]	--	100°F HSM	--	Seismic	C
HSM-8a Earthquake Loading - FF	HSM	X	--	--	11.0 psi[17,18]	--	100°F HSM	--	EQ + FF	C
HSM-9 Flood Load (50' H_2O) - Cold	HSM	X	--	--	0 psi	22 psi	0°F HSM	--	Flood[3]	C
HSM-10 Flood Load (50' H_2O) - Hot	HSM	X	--	--	0 psi	22 psi	100°F HSM	--	Flood[illegible]	C

HSM UNLOADING	Horizontal DW DSC	Horizontal DW Fuel	Vertical DW DSC	Vertical DW Fuel	Internal Pressure[9]	External Pressure	Thermal Condition	Handling Loads	Other Loads	Service Level
UL-1 Normal Unloading - Cold	HSM	X	--	--	11.0 psi[18]	--	0°F HSM	-60 Kip	--	A
UL-2 Normal Unloading - Hot	HSM	X	--	--	11.0 psi[18]	--	100°F HSM	-60 Kip	--	A
UL-3	HSM	X	--	--	11.0 psi[18]	--	125°F w/shade	-60 Kip	--	A
UL-4 Off-Normal Unloading - Cold	HSM	X	--	--	11.0 psi[18]	--	0°F HSM	-60 Kip	--	B
UL-5 Off-Normal Unloading - Hot	HSM	X	--	--	11.0 psi[18]	--	100°F HSM	-60 Kip	--	B
UL-6	HSM	X	--	--	11.0 psi[18]	--	125°F w/shade	-60 Kip	--	B
UL-7 Off-Norm. Unloading-FF/Hot[8,16]	HSM	X	--	--	11.0 psi[18]	--	100°F HSM	-80 kip	--	C
UL-8 Accident Unloading - FF/Hot[illegible]	HSM	X	--	--	60 psi[illegible]	--	100°F HSM	-80 kip	--	D

DSC Unloading/Reflood	Horizontal DW DSC	Horizontal DW Fuel	Vertical DW DSC	Vertical DW Fuel	Internal Pressure	External Pressure	Thermal Condition	Handling Loads	Other Loads	Service Level
RF-1 DSC Reflood	--	--	Cask	X	20.0 psi (max)	Hydrostatic	100°F Cask	--	--	B

Notes (continued):

8. This pressure is applied to the outer pressure boundary.
9. Unless noted otherwise, pressure is applied to the inner pressure boundary.
10. Fuel deck seismic loads are assumed enveloped by handling loads.
11. Actual pressure during this event is the normal condition pressure of 10.0 psi.
    Maximum off-normal in-cask pressure (10% failed fuel, 100°F ambient) is less than 11 psi.
12. This is an enveloping pressure, actual pressure under blocked vent conditions (no failed fuel) is less than 11 psig.
13. This is an enveloping pressure, actual pressure under 125°F blocked vent conditions with failed fuel is less than 60 psig.
14. Not Used
15. This is an enveloping pressure, based on 100% failed fuel and 125°F blocked HSM vent conditions.
16. Load Cases UL-7 and UL-8 envelop loading cases where the insertion loading of 80 kips is considered with an accident pressure (the insertion force is opposed by internal pressure).
17. This pressure is based on two accidents occuring simultaneously (failed fuel+seismic).
18. This is an enveloping pressure. Actual off-normal pressure (10% failed fuel) is less than 11 psi for the 24P DSCs and less than 10 psi for the 52B DSCs. This envelops the normal pressure of 10 psi.

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9. CONDUCT OF OPERATIONS

9.1 Organizational Structure

9.1.1 Corporate Organization

The ISFSI may be operated under the same corporate management organization which is responsible for operation of the licensee's plant.

9.1.1.1 Corporate Functions, Responsibilities and Authorities

The various departments within the licensee's organization have responsibility for construction, quality assurance, testing and operation of the ISFSI. TN's corporate functions, responsibilities and authorities for quality assurance, as described in Chapter 11 are applicable for design and procurement of NUHOMS® ISFSI components.

9.1.1.2 Applicant's In-House Organization

The licensee has specific responsibility for design of plant specific structures and systems, specifications, and procurement of materials and equipment, and preparation of construction and installation drawings for the ISFSI. The licensee maintains responsibility for the management of spent fuel and is responsible for operation and maintenance of the ISFSI.

9.1.1.3 Interrelationship with Contractors and Suppliers

The prime contractor for design of the NUHOMS® ISFSI and supply of its components is TN. The ISFSI is to be owned and operated by the licensee. Construction of the ISFSI is the responsibility of an approved construction contractor, to be selected by the licensee. Subsurface investigations at the ISFSI site are to be performed by the licensee's soils engineer.

9.1.1.4 Applicant's Technical Staff

The licensee's technical staff supporting the ISFSI is typically described in the plant's FSAR.

9.1.2 Operating Organization, Management and Administrative Control System

9.1.2.1 On-Site Organization

The licensee's on-site organization is responsible for operation of the ISFSI and typically maintains primary responsibility for spent fuel storage.

9.1.2.2 Personnel Functions, Responsibilities and Authorities

The functions, responsibilities and authorities of major personnel positions, including discussions of specific succession of responsibility for overall operation of the plant are the responsibility of the licensee. These functions, responsibilities and authorities typically extend to the ISFSI.

9.1.3 Personnel Qualification Requirements

The minimum qualification requirements for major operating, technical and maintenance supervisory personnel, as well as the qualification of persons assigned to managerial and technical positions, are the responsibility of the licensee.

9.1.4 Liaison with Other Organizations

Arrangements made with outside organizations are the responsibility of the licensee.

9.2 Pre-Operational Testing and Operation

A comprehensive pre-operational testing program has been carried out at Carolina Power and Light's Robinson plant NUHOMS® ISFSI. This program was developed jointly and conducted by Carolina Power and Light, the Department of Energy, the Electric Power Research Institute and NUTECH (now TN). These pre-operational testing results (9.1), provide sufficient data to demonstrate that the analytical methods described in this SAR provide conservative thermal and radiological results.

Prior to operation of the ISFSI for a particular plant, the licensee should perform functional tests of the in-plant operations, the on-site transport 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® ISFSI at Duke Power Company's Oconee Nuclear Station, Baltimore Gas and Electric Company's Calvert Cliffs Nuclear Power Plant, Pennsylvania Power & Light Company's Susquehanna Steam Supply Electric Station and Toledo Edison's Davis Besse Nuclear Station. Since the loading steps, transfer equipment and license conditions for the site specific and general licenses are similar, licensees using previously licensed and pre-operationally tested transfer equipment, in conjunction with the general license HSMs and DSCs, may limit dry runs to the new interfaces and operational aspects of general license HSMs and DSCs.

9.2.1 Administrative Procedures for Conducting Test Programs

Preoperational testing is typically governed by existing plant procedures for conducting testing.

9.2.2 Test Program Description

The testing program conducted by the licensee utilizes a DSC loaded with mock-up fuel, the transfer cask and associated transfer equipment, and an HSM. The tests should simulate, as nearly as possible, the actual operations involved in preparing a DSC for storage and ensure that they can be performed safely during actual emplacement of spent fuel in the ISFSI. Verification of ALARA practices, which are not completely achievable during dry runs, takes place during the initial fuel loadings. Guidelines for such tests are provided in the following paragraphs.

A. An actual DSC should be utilized for preoperational testing. The DSC should be loaded into the transfer cask to verify fit and adequacy of the cask/DSC annulus

seal. Additionally, the DSC may be used in operational testing of the transfer equipment and HSM.

B. Functional testing is to be performed with the transfer cask and lifting yoke. These tests are to ensure that the transfer cask can be safely lifted from the plant's cask receiving area to the cask washdown area. It should then be, as a minimum, partially lowered into the spent fuel pool and positioned in the cask loading area to verify clearances and travel path.

C. The transfer cask should be placed on the transport trailer, which should then be transported to the ISFSI along a predetermined route and aligned with an HSM. Compatibility of the transport trailer with the transfer cask, verification of the transfer route to the ISFSI, and maneuverability within the confines of the ISFSI should be verified.

D. The transfer trailer should be aligned and docked with the HSM. The hydraulic ram should be used to insert a DSC loaded with test weights in the HSM and then retrieve it. Transfer of the DSC to the HSM should verify that the support skid positioning system and the hydraulic ram system operate safely for both insertion and retrieval of a DSC.

9.2.3 Test Discussion

Implementation of the test program is discussed in the paragraphs which follow.

A. The purpose of the preoperational tests is to ensure that a DSC can be properly and safely placed in the spent fuel pool, loaded with spent fuel, transported to the ISFSI, inserted in the HSM, and retrieved from the HSM. Proper operation of the DSC, transfer cask, and transfer equipment, as well as the associated auxiliary equipment (e.g., automated welding equipment and vacuum drying system), provides such assurance.

B. Detailed procedures should be developed and implemented by licensee's personnel who are responsible for ensuring that the test requirements are satisfied.

C. The expected results of the preoperational tests are the successful completion of the following: placement of a DSC into the transfer cask, placement of the transfer cask into and out of the spent fuel pool, transporting the transfer cask loaded with a DSC and test weights to the ISFSI, and transfer of a DSC to/from the HSM. The tests are deemed successful if the expected results are achieved safely and without damage to any of the components or associated equipment.

10. OPERATING CONTROLS AND LIMITS

The information previously presented in SAR Chapter 10, Operating Controls and Limits is contained in the Technical Specifications of NUHOMS® CoC 1004. Hence, the contents of SAR Chapter 10.0 are being deleted in their entirety.

SAR Chapter 10 requirements are currently referenced in various TN documents. Table 10-1 provides a cross-reference index of these requirements against the corresponding requirements currently listed in the NUHOMS® CoC.

Table 10-1

Index of CoC Requirements v/s Historical SAR References

CoC Section No.	Title of CoC Requirements	Historical SAR Reference
1.2	**Technical Specifications, Functional and Operating Limits**	**10.3**
1.2.1	Fuel Specification	10.3.1
1.2.2	DSC Vacuum Pressure During Drying	10.3.2
1.2.3	DSC Helium Backfill Pressure	10.3.3
1.2.3a	61BT DSC Helium Backfill Pressure	—
1.2.4	DSC Helium Leak Rate of Inner Seal Weld	10.3.4
1.2.4a	61BT DSC Helium Leak Rate of Inner Seal Weld	—
1.2.5	DSC Dye Penetrant Test of Closure Welds	10.3.5
1.2.6	Deleted	10.3.6
1.2.7	HSM Dose Rates	10.3.7
1.2.8	HSM Maximum Air Exit Temperature	10.3.8
1.2.9	Transfer Cask Alignment with HSM	10.3.9
1.2.10	DSC Handling Height Outside the Spent Fuel Pool Building	10.3.10
1.2.11	Transfer Cask Dose Rates	10.3.11
1.2.12	Maximum DSC Removable Surface Contamination	10.3.12
1.2.13	TC/DSC Lifting Heights as a Function of Low Temperature and Location	10.3.13
1.2.14	TC/DSC Transfer Operations at High Ambient Temperatures	10.3.14
1.2.15	Boron Concentration in the DSC Cavity Water (24-P Design Only)	10.3.15
1.2.16	Provision of TC Seismic Restraint Inside the Spent Fuel Building as a function of Horizontal Acceleration and Loaded Cask Weight	10.3.16
1.2.17	Vacuum Drying Duration Limits	
Table 1-1a	PWR Fuel Specifications of Fuel to be Stored in the Standardized NUHOMS®-24P DSC	Table 10.3-1
Table 1-1b	BWR Fuel Specifications of Fuel to be Stored in the Standardized NUHOMS®-24P DSC	Table 10.3-2
Table 1-1c	BWR Fuel Specification of Fuel to be Stored in the Standardized NUHOMS®-61BT DSC	—
Table 1-1d	BWR Fuel Assembly Design Characteristics	—
Figure 1.1	PWR Fuel Criticality Acceptance Curve	Figure 10.3.1
1.3	**Surveillance and Monitoring**	**10.4**
1.3.1	Visual Inspection of HSM Air Inlets and Outlets (Front Wall and Roof Birdscreen)	10.4.1
1.3.2	HSM Thermal Performance	10.4.2
Table 1.3.1	Summary of Surveillance and Monitoring Requirements	Table 10.4-1

11. QUALITY ASSURANCE

The Quality Assurance Program to be applied to the "important-to-safety" and " safety related" activities associated with the standardized NUHOMS® system is as described in the Transnuclear, Inc. (TN) Quality Assurance Program Description unless noted otherwise.

11.1 Introduction

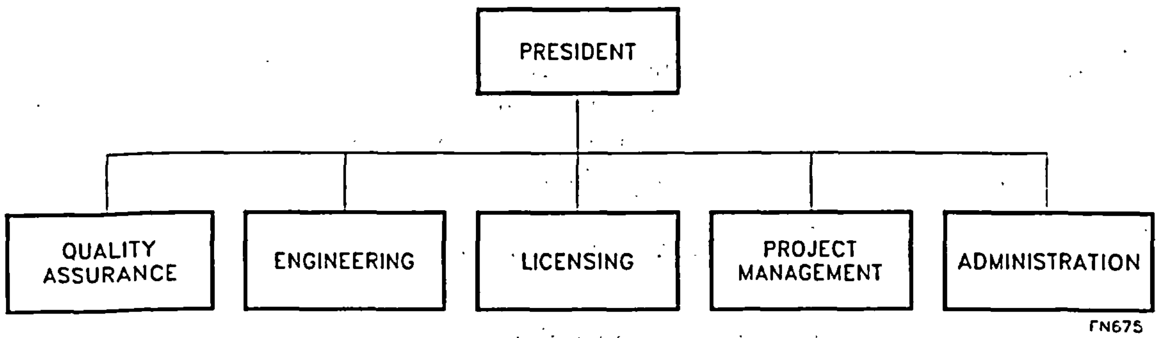
This section provides a brief summary of the quality assurance controls which apply to activities affecting the quality of NUHOMS® parts, components, and systems designated as "important to safety" and "safety related.". System components which are "important to safety" are defined herein. Activities affecting quality are defined by site-specific contract and may include any or all of the following: design, procurement, fabrication, handling, shipping, storage, cleaning, erection, inspection, test, repair, or modification. These activities shall be performed in accordance with a quality assurance program which meets the requirements specified herein.

TN's Quality Assurance Program shall be applied to the important-to-safety activities within TN's scope of responsibility as defined herein. The TN Quality Assurance Program complies with the criteria and requirements of 10CFR72, Subpart G. The complete description and specific commitments of the TN Quality Assurance Program are contained in the TN Quality Assurance Program Description. The Nuclear Regulatory Commission (NRC) has approved the TN QA Program Description for 10CFR72, Subpart G. Changes to the TN QA program description shall be submitted to the NRC for approval within thirty (30) days of implementation. Changes to the TN QA program which decrease or delete previously approved quality assurance commitments shall be submitted to the NRC for approval prior to implementation.

A matrix comparing 10CFR72, Subpart G criteria with the TN QA Program Description is provided in Table 11.1-1.

Table 11.1-1
Quality Assurance Criteria Matrix

10CFR72, Subpart G	TN QA Program Description	
.142	1.0	Organization
.144	2.0	QA Program
.146	3.0	Design Control
.148	4.0	Procurement Document Control
.150	5.0	Procedures, Instructions, and Drawings
.152	6.0	Document Control
.154	7.0	Control of Purchased Items and Services
.156	8.0	Identification and Control of Materials, Parts, and Components
.158	9.0	Control of Special Processes
.160	10.0	Inspection
.162	11.0	Test Control
.164	12.0	Control of Measuring and Test Equipment
.166	13.0	Handling, Storage, and Shipping
.168	14.0	Inspection and Test Status
.170	15.0	Control of Nonconforming Items
.172	16.0	Corrective Action
.174	17.0	Records
.176	18.0	Audits



Notes:

1. Licensing may report to Engineering.
2. Administrative activities may report to various other organizations.

Figure 11.1-1
NUHOMS® Project Organization Chart

11.2 "Important-to-Safety" and "Safety Related" NUHOMS® System Components

TN will apply the TN Quality Assurance Program to those NUHOMS® components for which TN has responsibility and which are "important to safety" and "safety related" as delineated in Section 3.4. These include the DSC with closure weld filler metal, the HSM, and the transfer cask. The lifting yoke is classified as "safety related".

Each item is first identified as "important to safety," "safety related" or "not important to safety." Items that are considered "important to safety" are further categorized using a graded quality approach. When the graded quality approach is used, a list shall be developed for each "important to safety" item which includes an assigned quality category consistent with the item's importance to safety. Quality categories shall be determined based on the guidance from Regulatory Guide 7.10:

Category A items are critical to safe operation. These items include structures, components, and systems whose failure or malfunction could result directly in a condition adversely affecting (1) safe spent fuel storage, (2) integrity of the spent fuel, or (3) public health and safety. This would include conditions as loss of primary containment with subsequent release of radioactive material, loss of shielding or an unsafe geometry compromising criticality control.

Category B items have a major impact on safety. These items include structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting (1) safe spent fuel storage, (2) integrity of the spent fuel, or (3) public health and safety. An unsafe operation could result only if a primary event occurs in conjunction with a secondary event or other failure or environmental occurrence.

Category C items have a minor impact on safety. These items include structures, components, and systems whose failure or malfunction would not significantly reduce the packaging effectiveness and would be unlikely to create a condition adversely affecting (1) safe spent fuel storage, (2) integrity of the spent fuel, or (3) public health and safety.

The Quality Assurance Program as described in paragraph 11.3 is applicable to each "important to safety" graded category and is limited as follows For "safety related" items the program is applied as described in Category A items. Appendix K provides clarification for the procurement of Category A items for the NUHOMS®-61BT DSC. Appendix L provides clarification for the procurement of Category A items for the NUHOMS®-24PT2 DSC.

D. Quality Assurance verification activities shall be performed by personnel qualified and certified in accordance with the requirements of the QA program.

E. Only lead auditor personnel require certification in accordance with the QA program.

Category C

A. Items may be purchased from a catalog or "off-the-shelf".

B. When received, the item shall be identified and checked for compliance with the purchase order and for damage.

The "important to safety" classifications are identified on the drawings (Appendix E).

Additional system components other than those delineated above, such as the ISFSI basemat, the remaining transfer equipment, the auxiliary equipment, and consumables (including the dry film lubricant) are not considered important-to-safety and will be controlled in accordance with good industrial practices.

If a utility elects to perform construction, and has an NRC approved QA program (10CFR50) that is equivalent to or exceeds TN's QA program, then the utility QA program is considered an acceptable substitute.

Category A

A. The design is based on the most stringent industrial codes or standards, and design verification shall be accomplished by prototype testing or formal design review.

B. Vendors for items and services for this category may only be selected from the Approved Suppliers List.

C. TN suppliers and subtier suppliers must have a QA program based on applicable criteria in Subpart G to 10CFR72, or equivalent.

D. Complete traceability of raw materials and the use of certified welders and processes is required.

E. All personnel performing Quality Assurance related inspection, tests, and examinations shall be qualified and certified in accordance with the requirements of the QA program.

F. Only qualified and certified auditors and lead auditors shall perform audits.

G. TN Quality personnel shall be required to inspect and/or approve supplier fabricated components prior to authorizing shipment release.

H. Welding consumables shall be procured as a Category A item if the intended use is unknown. If purchased for a specific B or C application, material must be so identified and its use be restricted to fabrication of the same level.

Category B

A. The design shall be based on the most stringent industrial codes and standards, but design verification may be through use of alternate calculations or computer codes.

B. The procurement of items need not be from the Approved Suppliers List. QA program requirements for the supplier shall be based upon the inspection and test requirements of the procured item.

C. Traceability of materials is not required; however, specified welds require completion by qualified, certified welders.

11.3 Description of TN 10CFR72 Subpart G Quality Assurance Program

11.3.1 Project Organization

The NUHOMS® system has been designed by a dedicated TN project organization.

QA duties are performed by the NUHOMS® project organization, the QA Manager, and QA Engineer.

The organization structure for the NUHOMS® project is presented in Figure 11.1-1. A description of TN's organizational structure, functional responsibilities, levels of authority, and lines of internal and external (client and supplier) communication may be found in the TN Quality Assurance Program Description.

Project QA controls are determined by the Project Manager and approved by the QA Manager. All Project Plans, regardless of the indicated applicability of QA requirements, are reviewed by the QA Manager to assure that QA controls are commensurate with the specific activity, item complexity, importance to safety and client-imposed contractual requirements.

Project personnel are indoctrinated, trained, and qualified in accordance with the TN QA Program.

11.3.2 Quality Assurance Program

TN has established and implemented a QA program for the control of quality in the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and modification of shipping containers for nuclear products. Training and/or evaluation of personnel qualifications are required for all QA functions in accordance with written procedures. The QA program assures that all quality requirements, engineering specifications and specific provisions of any package design approval are met. Those characteristics critical to safety are emphasized.

The TN Quality Assurance Manager regularly evaluates the TN QA program for adherence to the 18 point criteria in scope, implementation and effectiveness. Further, the TN President requires that the Quality Assurance Program, including the QA Program Description Policies and Procedures, be implemented and enforced on all applicable projects at TN.

11.3.3 Design Control

Important-to-safety and safety related NUHOMS® design activities including the performance of design verifications shall be implemented in accordance with the TN Quality Assurance Program.

Errors and deficiencies in the design, including the design process, are documented in the form of Corrective Action Reports.

Typically, valid industry standards and specifications are used for the selection of suitable materials, parts, equipment and processes for important-to-safety and safety related structures, systems, or components. Standard, or off-the-shelf items, and items previously approved for a different application are reviewed for suitability prior to selection.

11.3.4 Procurement Document Control

Procurement documents are prepared in accordance with the TN Quality Assurance Program which delineates the actions to be accomplished in the preparation, review, approval, and control of procurement documents. Review and approval of procurement documents by the QA Manager are documented on the procurement documents prior to release to assure the adequacy of quality requirements stated therein. This review determines that quality requirements are correctly stated, inspectable, and controllable; that there are adequate acceptance and rejection criteria; and that the procurement document has been prepared, reviewed, and approved in accordance with QA program requirements.

The procurement documents shall identify the documentation required to be submitted for information, review, or approval by TN or TN's client. The time of submittal shall also be established. When TN requires the supplier to maintain specific quality assurance records, the retention times and disposition requirements shall be prescribed.

11.3.5 Procedures, Instructions, and Drawings

Activities affecting quality are prescribed and accomplished in accordance with approved, written procedures instructions, or drawings as required by the TN QA Program.

11.3.6 Document Control

The issuance, distribution, and receipt of documents, which prescribe activities affecting quality, are controlled in accordance with the TN Quality Assurance Program. Controlled documents include, but are not limited to, the TN design specifications and criteria documents, drawings, instructions, and test procedures.

The individuals or groups responsible for reviewing, approving, and issuing documents and revisions thereto are identified in the "Responsibilities" sections of the TN QA Program.

11.3.7 Control of Purchased Items and Services

The control of purchased items and services shall be implemented in accordance with the TN Quality Assurance Program.

Surveillance of subcontracted activities is planned and performed in accordance with written procedures to assure conformance to the purchase order. These procedures provide for instructions that specify the characteristics to be witnessed, inspected or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these instructions.

TN suppliers shall furnish documentation that identifies any procurement requirements which have not been met, together with a description of those nonconformances dispositioned as "use-as-is" or "repair."

Documentation from TN suppliers which demonstrates compliance with procurement requirements (such as material test reports, NDE results, performance test results, etc.) is periodically evaluated by audits, independent inspections, or tests as necessary to assure its validity.

11.3.8 Identification and Control of Materials, Parts, and Components

Materials, parts, and components shall be identified and controlled in accordance with the TN Quality Assurance Program. Hardware identification requirements are determined during generation of design drawings and specifications such that the location and method of identification do not affect the form, fit, function, or quality of the item being identified.

11.3.9 Control of Special Processes

The control of special processes, such as nondestructive examination, chemical cleaning, welding, and heat treating shall be performed in accordance with the TN Quality Assurance Program.

11.3.10 Inspection

Receipt inspections, in-process and final inspections of TN fabricated, constructed, or erected items, systems, components, or structures shall be performed in accordance with the TN Quality Assurance Program.

11.3.11 Test Control

Test control shall be accomplished in accordance with the TN Quality Assurance Program.

11.3.12 Control of Measuring and Test Equipment

The TN QA Program defines the requirements for calibration of measuring and testing equipment. Calibration is against certified measurement standards which have known relationships to national standards, where such standards exist. Where such standards do not exist, the basis for calibration shall be documented.

11.3.13 Handling, Storage and Shipping

Handling, storage, and shipping shall be conducted in accordance with the TN Quality Assurance Program. Special handling, preservation, storage, cleaning, packaging, and shipping requirements are established and accomplished by qualified individuals in accordance with predetermined work and inspection instructions.

11.3.14 Inspection and Test Status

The use of inspection and test status tags shall be accomplished in accordance with the TN Quality Assurance Program.

11.3.15 Control of Nonconforming Items

The TN Quality Assurance Program defines the requirements and assigns the responsibilities for the control, identification, segregation, documentation, and close-out of nonconforming items to prevent their inadvertent installation or use in fabrication, construction, or erection.

Nonconformance reports identify the item description and quantity, the disposition of the nonconformance, the inspection requirements, and signature approval of the disposition. They are retained in the Project files and are periodically analyzed to show quality trends and help identify root causes of nonconformances. Significant results are reported to responsible management for review and assessment.

Nonconforming items are segregated from acceptable items and tagged to prevent inadvertent use until properly dispositioned and closed out.

Nonconforming items dispositioned "use-as-is" or “repair” are reported to the client.

11.3.16 Corrective Action

Corrective action for significant conditions adverse to quality shall be taken in accordance with the TN Quality Assurance Program.

11.3.17 Records

The TN Quality Assurance Program defines the scope of the records program such that sufficient records are maintained to provide documentary evidence of the quality of items and the activities affecting quality.

11.3.18 Audits and Surveillances

A comprehensive system of planned and documented audits, including audits of suppliers and site construction activities, verifies compliance with all aspects of the TN Quality Assurance Program and determines the effectiveness of the program.

Audits are performed by certified lead auditors and are planned, performed, and documented in accordance with the TN Quality Assurance Program.

Unannounced QA surveillances may be performed on activities affecting quality by the TN Quality Assurance Manager, or his designee, on an as-needed basis to further assure compliance with QA requirements.

11.4 Conditions of Approval Records

As required by 10CFR72, Subpart L, TN will establish and maintain records for each storage component fabricated under a certificate of compliance as required by §72.234(d). The records will be available for inspection as required by §72.234(e). Written procedures and appropriate tests will be established prior to use of the storage components, which will be provided to each NUHOMS® system user as required by §72.234(f).

Information withheld under 10 CFR 2.390(d)

Figure A.1-3

QAD-CGGP Model of DSC in Cask for Top Axial Gamma Dose Rate

Information withheld under 10 CFR 2.390(d)

Figure A.1-4
QAD-CGGP Model of DSC in Cask for Bottom Axial Gamma Dose Rate

APPENDIX E

DRAWINGS FOR THE STANDARDIZED NUHOMS® SYSTEM

This appendix contains the following items:

E.1 Drawings for NUHOMS® Dry Shielded Canisters

- E.1.1 Standardized NUHOMS®-24P DSC Drawings
- E.1.2 Standardized NUHOMS®-52B DSC Drawings
- E.1.3 Standardized NUHOMS®-24P Long Cavity DSC Drawings

E.2 Drawings for NUHOMS® Horizontal Storage Module

E.3 Drawings for NUHOMS® On-Site Transfer Cask

The drawings for the NUHOMS®-61BT DSC are contained in Appendix K. The drawings for the NUHOMS®-24PT2S and -24PT2L DSCs are contained in Appendix L.

APPENDIX E.1

DRAWINGS FOR NUHOMS® DRY SHIELDED CANISTERS

Appendix E.1.1

This Appendix contains the following drawings of the standardized NUHOMS®-24P system:

Drawing Number	Title
NUH-03-1020NP-SAR	General License NUHOMS® DSC for PWR Fuel Basket Assembly
NUH-03-1021NP-SAR	General License NUHOMS® DSC for PWR Fuel Shell Assembly
NUH-03-1022NP-SAR	General License NUHOMS® DSC for PWR Fuel Basket-Shell Assembly
NUH-03-1023NP-SAR	General License NUHOMS® DSC for PWR Fuel Main Assembly.

NUH-03-1020NP-SAR 1 OF 3

Information withheld under 10 CFR 2.390(d)

DWG NO NUH-03-1020NP-SAR

SHEET 2 OF 3

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

DWG NO NUH-03-1020NP-SAR

SHEET 3 OF 3

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

DWG NO NUH-03-1023NP-SAR SHEET 1 OF 3

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

NUH-03-102SNP-SAR 2 OF 3

Information withheld under 10 CFR 2.390(d)

Appendix E.1.2

This Appendix contains the following drawings of the standardized NUHOMS®-52B system:

Drawing Number	Title
NUH-03-1029NP-SAR	General License NUHOMS® DSC for Channeled BWR Fuel Shell Assembly
NUH-03-1030NP-SAR	General License NUHOMS® DSC for Channeled BWR Fuel Basket-Shell Assembly
NUH-03-1031NP-SAR	General License NUHOMS® DSC for Channeled BWR Fuel Main Assembly
NUH-03-1032NP-SAR	General License NUHOMS® DSC for Channeled BWR Fuel, BWR Fuel Basket Assembly

Information withheld under 10 CFR 2.390(d)

NUH-03-1029NP-SAR 1 OF 1

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

NUH-03-1031NP-SAR

1 OF 3

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Appendix E.1.3

This Appendix contains the following drawings of the standardized NUHOMS®-24P Long Cavity system:

Drawing Number	Title
NUH-03-1050NP-SAR	General License NUHOMS® 24P Long Cavity DSC Basket Assembly
NUH-03-1051NP-SAR	General License NUHOMS® 24P Long Cavity DSC Shell Assembly
NUH-03-1052NP-SAR	General License NUHOMS® 24P Long Cavity DSC Basket-Shell Assembly
NUH-03-1053NP-SAR	General License NUHOMS® 24P Long Cavity DSC Main Assembly

DWG NO NUH-03-1050NP-SAR
SHEET 1 OF 3

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

DWG NO. NUH-03-1050NP-SAR

SHEET 3 OF 3

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

DWG NO. NUH-03-1051NP-SAR

SHEET 2 OF 2

NUH-03-1052NP-SAR

1 OF 2

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

NUH-03-1052NP-SAR

2 OF 2

Information withheld under 10 CFR 2.390(d)

DWG NO NUH-03-1053NP-SAR

Information withheld under 10 CFR 2.390(d)

NUH-03-1053NP-SAR

Information withheld under 10 CFR 2.390(d)

APPENDIX E.2

DRAWINGS FOR NUHOMS® HORIZONTAL STORAGE MODULE

Appendix E.2

This Appendix contains the following drawings for the standardized NUHOMS® horizontal storage module:

Drawing Number	Title
NUH-03-6008NP-SAR	Standardized NUHOMS® ISFSI Horizontal Storage Module ISFSI General Arrangement
NUH-03-6009NP-SAR	Standardized NUHOMS® ISFSI Horizontal Storage Module Main Assembly
NUH-03-6016NP-SAR	Standardized NUHOMS® ISFSI Horizontal Storage Module DSC Support Structure

DWG NO NUH-03-6008NP-SAR

SHEET 1 OF 3

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

DWG NO. NUH-03-6016NP-SAR

SHEET 1 OF 1

APPENDIX E.3

DRAWINGS FOR NUHOMS® ON-SITE TRANSFER CASK

Appendix E.3

This Appendix contains the following drawings for the standardized NUHOMS® On-site Transfer Cask:

Drawing Number	Title
NUH-03-8000NP-SAR	General License NUHOMS® ISFSI On-Site Transfer Cask Overview
NUH-03-8001NP-SAR	General License NUHOMS® ISFSI On-Site Transfer Cask Structural Shell Assembly
NUH-03-8002NP-SAR	General License NUHOMS® ISFSI On-Site Transfer Cask Inner and Outer Shell Assembly
NUH-03-8003NP-SAR	General License NUHOMS® ISFSI On-Site Transfer Cask Main Assembly

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

NUH-03-8001NP-SAR 1 OF 4

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

NUH-03-8001NP-SAR

SHEET 3 OF 4

Information withheld under 10 CFR 2.390(d)

NUH-03-8001NP-SAR

4 OF 4

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

J.4 Thermal Evaluation

J.4.1 Discussion

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

J.4.2 Summary of Thermal Properties of Materials

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

J.4.3 Technical Specifications of Components

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

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

Decay Heat

The addition of the BPRA components adds a small amount of decay heat to the fuel assemblies, which needs to be addressed. The maximum heat generation of the BPRA components is calculated to be 8 watts, as described in Section J.5.2. A new fuel qualification table (Table 1-2c of Fuel Specification 1.2.1) has been added to address the addition of the heat generated by the BPRAs. The same methodology as presented in Chapter 8 of the SAR is used. The total decay heat of each assembly is taken to be that generated by the fuel plus the decay heat generated by the BPRAs. The criteria for fuel cladding temperature limit remains the same, but the allowable decay heat from the fueled rods in an assembly is reduced by 8 watts to accommodate the BPRAs. Therefore, the results from Chapter 8 of the SAR for normal and off-normal conditions remain valid for the maximum design basis decay heat of 1 kW per assembly, including the BPRA contribution.

Pressure Evaluation

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

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

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

$$n_{He} = \frac{(14.7\,psia)(6894.8\,Pa/psi)(24 \cdot 16 \cdot 3.55\,in^3)(1.6387x10^{-5}\,m^3/in^3)}{(8.314\,J/gmol\text{-}K)(293.15\,K)} \cdot \frac{kg/m \cdot s^2}{Pa} \cdot \frac{J}{kg \cdot m^2/s^2}$$

$$= 0.93\,gmoles$$

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

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

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

Thermal Expansion

As described in FSAR section 8.1.1.3B, adequate space is provided in the 24P standard DSC cavity between the basket assembly and the shield plug assemblies for free thermal expansion. To verify that for the 24P long cavity DSC adequate provision for free axial

expansion of the spent fuel assemblies and other internal components of the basket is included, the differential expansion of each DSC component is calculated using the methodology described in FSAR section 8.1.1.3B. The –40° F ambient condition for PWR fuel inside the long cavity DSC represents the worst case for differential thermal growth situation analyzed.

For fuel assemblies with BPRAs, using a maximum unirradiated initial length of 171.71 inch and a maximum burnup of 45,000 MWd/MTU, the calculated hot length of fuel assembly is 172.061 inch. Using a minimum cold length value of 172.94 inch for the 24P long cavity DSC, the calculated hot length of the DSC cavity is 173.046 inch. Hence the clearance available for irradiation growth at hot conditions is 0.985 inch (173.046 – 172.061). Taking thermal expansion of Zircalloy into consideration, the minimum cold clearance available to accommodate irradiation growth is 0.983 inch..

For fuel assemblies with BPRAs which are 0.25 inch longer (a maximum unirradiated initial length of 171.96 inch), the corresponding maximum burnup is determined to be 32,000 MWd/MTU, and the minimum cold clearance available to accommodate irradiation growth is 0.733 inch (0.983 –0.250).

The thermal expansion between the basket assembly components and the DSC shell .in the radial direction remains bounded by the analysis presented in FSAR section 8.1.1.3B.

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J.6 Criticality Evaluation

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

J.6.1 Discussion and Results

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

J.6.2 Package Fuel Loading

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

Table J.6–1 B&W 15x15 Fuel Assembly

Fuel Assembly Design:	**B&W 15x15**		
Parameter	Value		Reference
Maximum number of fuel rods	208		[J-5]
Fuel density, % theoretical	95%		[J-5]
Quantity of guide tubes	16		[J-5]
Quantity of instrument tubes	1		[J-5]
Parameter	inches	cm	Reference
Pellet diameter	0.3686 & 0.370	0.9362 & 0.9398	[J-5]
Active fuel length	141.8	360.1720	[J-5]
Cladding thickness	0.0265	0.0673	[J-5]
Fuel rod OD	0.43	1.0922	[J-5]
Fuel rod pitch	0.568	1.4427	[J-5]
Guide tube OD	0.53	1.3462	[J-6]
Guide tube thickness	0.016	0.0406	[J-6]
Instrument tube OD	0.493	1.2522	[J-6]
Instrument tube thickness	0.026	0.0660	[J-6]

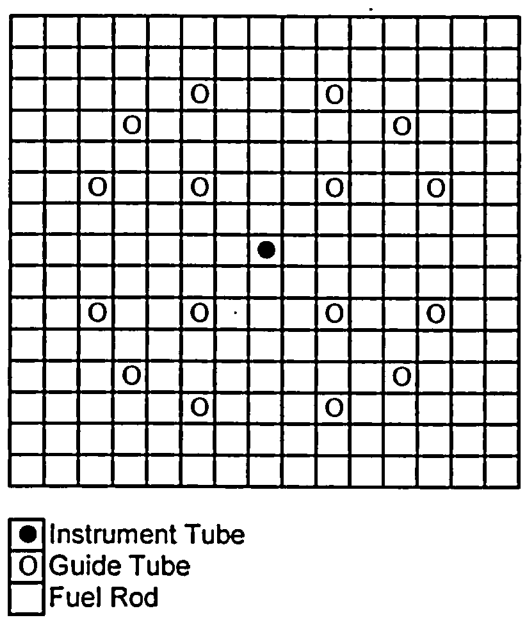

Figure J.6-1 B&W 15x15 Fuel Assembly

APPENDIX K

NUHOMS® 61BT DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

K.1.4 Generic Cask Arrays

No change.

K.1.5 Supplemental Data

The following TN drawings are enclosed:

1. NUHOMS® -61BT Transportable Canister for BWR Fuel General Arrangement, Drawing NUH-61B-1060NP-SAR.

2. NUHOMS® -61BT Transportable Canister for BWR Fuel Shell Assembly, Drawing NUH-61B-1061NP-SAR.

3. NUHOMS® -61BT Transportable Canister for BWR Fuel Canister Details, Drawing NUH-61B-1062NP-SAR.

4. NUHOMS® -61BT Transportable Canister for BWR Fuel Basket Assembly, Drawing NUH-61B-1063NP-SAR.

5. NUHOMS® -61BT Transportable Canister for BWR Fuel Basket Details, Drawing NUH-61B-1064NP-SAR.

6. NUHOMS® -61BT Transportable Canister for BWR Fuel Parts List, Drawing NUH-61B-1065NP-SAR.

7. NUHOMS® -61BT Transportable Canister Top & Bottom Cap Details for Failed BWR Fuel, Drawing NUH-61B-1066NP-SAR.

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

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8 7 6 5 4 3 2 1

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Information withheld under 10 CFR 2.390(d)

8 7 6 5 4 3 2 1

Information withheld under 10 CFR 2.390(d)

H G F E D C B A

8 7 6 5 4 3 2 1

K.2.4 Decommissioning Considerations

No change.

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

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

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

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

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

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

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

associated welds. Figure K.3.1-1 provides a graphic representation of the 61BT-DSC confinement boundary.

The welds made during fabrication of the 61BT-DSC that affect the confinement boundary of the DSC include the weld applied to the inner bottom cover plate and the circumferential and longitudinal seam welds applied to the shell. These welds are inspected (radiographic or ultrasonic inspection, and liquid penetrant inspection) according to the requirements of Subsection NB of the ASME Code. The vent and siphon block weld is also made during fabrication and is liquid penetrant inspected in accordance with Subsection NB of the ASME Code.

The welds applied to the vent and siphon port covers and the inner and outer top cover plates (including cover plate test plug) during closure operations define the confinement boundary at the top end of the 61BT-DSC. These welds are applied using a multiple-layer technique with multi-level PT in accordance with Subsection NB of the ASME Code and Code Case N-595-1.

The basis for the allowable stresses for the confinement boundary is ASME Code Section III, Division I, Subsection NB Article NB-3200 [3.1] for normal condition loads (Level A), off normal condition loads (Level B and C) and Appendix F for accident condition loads (Level D). See Section K.2.2 for additional design criteria.

K.3.1.2.2 <u>DSC Basket</u>

The basket is designed to meet the heat transfer, nuclear criticality, and the structural requirements. The basket structure must provide sufficient rigidity to maintain a subcritical configuration under the applied loads. The 304 stainless steel members in the NUHOMS®-61BT basket are the primary structural components. The neutron poison plates are the primary heat conductors, and provide the necessary criticality control.

The stress analyses of the basket for normal and accident conditions do not take credit for the poison plates except for through-thickness-compression. However, the weight of the poison plates is included in the stress evaluations.

The basis for the allowable stresses for the 304 stainless steel basket assembly is Section III, Division I, Subsection NG of the ASME Code [3.1]. The hypothetical impact accidents are evaluated as short duration, Level D conditions. The stress criteria are taken from Section III, Appendix F of the ASME Code [3.1]. See Section K2.2 for additional design criteria. The basket stress limits are provided for information in Table K.3.1-1.

The basket holddown ring is set between the top of the basket assembly and inside surface of the DSC top shield plug. The holddown ring is used to prevent the basket assembly from sliding freely in the axial direction during the handling/transfer and operation/storage loading conditions. The basket holddown ring is designed, fabricated and inspected in accordance with the ASME Code Subsection NF [3.1], to the maximum practical extent.

K.3.1.2.3 ASME Code Exception for the 61BT DSC

The primary confinement boundary of the NUHOMS®-61BT DSC consists of the DSC shell, the inner top cover plate, the inner bottom cover plate, the siphon vent block, and the siphon/vent port cover plate. Even though the code is not strictly applicable to the DSC, it is TN's intent to follow Section III, Subsection NB of the Code as closely as possible for design and construction of the confinement vessel. The DSC may, however, be fabricated by other than N-stamp holders and materials may be supplied by other than ASME Certificate Holders. Thus the requirements of NCA are not imposed. TN's quality assurance requirements, which are based on 10CFR72 Subpart G and NQA-1 are imposed in lieu of the requirements of NCA-3800. The SAR is prepared in place of the ASME design and stress reports. Surveillances are performed by TN and utility personnel rather than by an Authorized Nuclear Inspector (ANI).

The basket is designed, fabricated and inspected in accordance with the ASME Code Subsection NG, to the maximum practical extent. The following exceptions are taken:

The poison and aluminum plates are not considered for structural integrity. Therefore, these materials are not required to be code materials. The quality assurance requirements of NQA-1 is imposed in lieu of NCA-3800. The basket is not code stamped. Therefore the requirements of NCA are not imposed. Fabrication and inspection surveillances are performed by TN and utility personnel rather than by an ANI.

A complete list of the ASME Code exceptions and justification for the confinement boundary of the NUHOMS®-61BT DSC and basket is provided in Table K.3.1-2 and Table K.3.1-3.

Table K.3.1-1
Numerical Values of Primary Stress Intensity Limits

(304 SS at 650°F)

Stress Category	Allowable Stresses		
	Normal Conditions (Level A)	Accident Conditions (Level D)	
	Elastic Analysis (ksi)	**Elastic/Plastic Analysis (ksi)**	**Elastic Analysis (ksi)**
Primary Membrane Stress Intensity (P_m)	16.2	44.38	38.88
Local Membrane Stress Intensity (P_L)	24.3	57.06	58.32
Primary Membrane + Bending Stress Intensity ($P_m + P_b$)	24.3	57.06	58.32
Primary Membrane + Secondary Stress Intensity Range ($P_m + P_b + Q$)	48.6	N/A	N/A
Shear	9.72	26.63	26.63
Bearing Stress (S_b)	26.85	N/A	N/A

Table K.3.1-2
ASME Code Exceptions for the NUHOMS®-61BT DSC Confinement Boundary

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NB-1100	Requirements for Code Stamping of Components	The NUHOMS®-61BT DSC shell is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical. However, 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	All materials designated as ASME on the SAR drawings are obtained from ASME approved MM or MS supplier(s) with ASME CMTR's 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 joints between the top outer and inner cover plates and containment shell are designed and fabricated per ASME Code Case N-595-1. The welds are partial penetration welds and the root and final layer are PT examined.
NB-5231	Full penetration corner weld joints require the fusion zone and the parent metal beneath the attachment surface to be UT after welding	The inner bottom cover plate weld joint is full penetration per Fig. NB-4243-1. The required UT inspection is performed on a best efforts basis. The joint is examined by RT and either PT or MT methods.
NB-6100 and 6200	All completed pressure retaining systems shall be pressure tested	The vent and siphon block is not pressure tested due to the manufacturing sequence. The siphon block weld is helium leak tested when fuel is loaded and then covered with the outer top closure plate.
NB-7000	Overpressure Protection	No overpressure protection is provided for the NUHOMS® -61BT DSC. The function of the NUHOMS®-61BT DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The NUHOMS®-61BT DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature. The NUHOMS®-61BT DSC is pressure tested in accordance with ASME Code Case N-595-1.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS®-61BT DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the NUHOMS®-61BT DSC. QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TN's approved QA program.

Table K.3.2-1
Summary of the NUHOMS®-61BT System Component Weights

COMPONENT DESCRIPTION	CALCULATED WEIGHT (KIPS)
DSC Shell Assembly	13.52
DSC Top Shield Plug Assembly	8.95
DSC Internal Basket Assembly	22.92
Total Empty Weight	**45.39**
61 BWR Spent Fuel Assemblies	≤ 43.0
Total Loaded DSC Weight (Dry)	**88.39**
Water in Loaded DSC	13.4
Total Loaded DSC Weight (Wet)	**101.79**
Transfer Cask Empty Weight	111.25
Total Loaded Transfer Cask Weight	**199.64**
HSM Single Module Weight, Model 80 (Empty)	**252.0**
HSM Single Module Weight, Model 102 (Empty)	**263.0**
HSM Single Module Weight, Model 80 (Loaded)	**340.4**
HSM Single Module Weight, Model 102 (Loaded)	**351.4**

K.3.3 Mechanical Properties of Materials

K.3.3.1 Material Properties

The mechanical properties of structural materials used in the 61BT DSC and basket are in accordance with ASME Code Section II, Part D [3.2]. A value of 2.78×10^{-6} used for the thermal coefficient of expansion for zircaloy is taken from reference [3.3] at a temperature of 850°F.

K.3.3.2 Materials Durability

The materials used in the fabrication of the NUHOMS®-61BT system are shown in Table K.3.6-3. Essentially all of the materials meet the appropriate requirements of the ASME Code, ACI Code and appropriate ASTM Standards. The durability of the shell assembly and basket assembly stainless steel components is well beyond the design life of the applicable components. The small amount of aluminum material used in the basket meets ASME Code standards and is relied upon for its thermal conductivity properties only. The poison material selected for criticality control of the NUHOMS®-61BT system has been tested and is currently in use for similar applications. Additionally, the NUHOMS®-61BT basket assembly resides in an inert helium gas environment for the majority of the design life. The specifications controlling the mix of the concrete, specified minimum concrete strength requirements, and fabrication controls ensure durability of the concrete for this application. The materials used in the NUHOMS®-61BT system will maintain the required properties for the design life of the system.

There are no chemical, galvanic or other reactions that could reduce the areal density of boron in the neutron poison plates with either of the poison plate materials.

D. Electroless Nickel Plated Carbon Steel

The carbon steel top shield plug of the DSC is plated with electroless nickel. This coating is identical to the coating used on the 52B DSC. It has been evaluated for potential galvanic reactions in Transnuclear West's response to NRC Bulletin 96-04 [3.9]. In BWR pools, the reported corrosion rates are insignificant and are expected to result in a negligible rate of reaction for the NUHOMS® BWR systems.

Lubricants and Cleaning Agents

Cleaning agents used for final cleaning on the NUHOMS®-61BT DSC are limited to those with chlorine contents of less than 1 ppm chloride. Never-seez or Neolube (or equivalent) is used to coat the threads and bolt shoulders of the closure bolts. The lubricant should be selected for compatibility with the spent fuel pool water and the DSC materials, and for its ability to maintain lubricity under long term storage conditions.

The DSC is cleaned in accordance with approved procedures to remove cleaning residues prior to shipment to the storage site. The basket is also cleaned prior to installation in the DSC. The cleaning agents and lubricants have no significant affect on the DSC materials and their safety related functions.

Hydrogen Generation

During the initial passivation state, small amounts of hydrogen gas may be generated in the 61BT DSC. The passivation stage may occur prior to submersion of the transfer cask into the spent fuel pool. Any amounts of hydrogen generated in the DSC will be insignificant and will not result in a flammable gas mixture within the DSC.

The small amount of hydrogen which may be generated during DSC operations does not result in a safety hazard. In order for concentrations of hydrogen in the cask to reach flammability levels, most of the DSC would have to be filled with water for the hydrogen generation to occur, and the lid would have to be in place with both the vent and drain ports closed. This does not occur during DSC loading or unloading operations.

After loading fuel into the NUHOMS®-61BT DSC, the shield plug is placed in the DSC and the transfer cask and DSC are raised to the pool surface. At this time the DSC is completely filled with water.

An estimate of the maximum hydrogen concentration can be made, ignoring the effects of radiolysis, recombination, and solution of hydrogen in water. Testing was conducted by Transnuclear [3.10] to determine the rate of hydrogen generation for aluminum metal matrix composite in intermittent contact with 304 stainless steel. The samples represent the neutron poison plates paired with the basket compartment tubes. The test specimens were submerged in

deionized water for 12 hours at 70 °F to represent the period of initial submersion and fuel loading, followed by 12 hours at 150 °F to represent the period after the fuel is loaded, until the water is drained. The hydrogen generated during each period was removed from the water and the test vessel and measured.

The test results were:

	12 hour @ 70 °F		12 hour @ 150 °F	
	$cm^3hr^{-1}dm^{-2}$	$ft^3hr^{-1}ft^{-2}$	$cm^3hr^{-1}dm^{-2}$	$ft^3hr^{-1}ft^{-2}$
aluminum MMC/SS304	0.517	1.696E-4	0.489	1.604E-4

The total surface area of the aluminum/stainless steel interface at the neutron absorber/compartment wall interface is 1462 ft^2. This surface area, combined with the test data at 150 °F above result in a hydrogen generation rate of

$$(1.6 \times 10^{-4}\ ft^3/ft^2hr)(1462\ ft^2)] = 0.23\ ft^3/hr$$

in the 61BT DSC. During welding of the top inner plate, the DSC is partially filled with water. The minimum free volume of the DSC is 120 cu. feet (based on 60 inches of space between the top inner plate and the water). The following assumptions are made to arrive at a conservative estimate of hydrogen concentration:

- All generated hydrogen is released instantly to the plenum between the water and the shield plug, that is, no dissolved hydrogen is pumped out with the water, and no released hydrogen escapes through the open vent port, and
- 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 $(0.23\ ft^3\ H_2/hr)*(8\ hr) / (120\ ft^3) = 1.5\ \%$. Monitoring of the hydrogen concentration before and during welding operations will be performed to ensure that the hydrogen concentration does not exceed 2.4%. If the concentration exceeds 2.4%, welding operations will be suspended and the DSC will be 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 K.4. The emissivity of the fuel compartment is 0.3, 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 0.8, 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.

C. Thermal Stress Analysis of Basket Plate Inserts

Basket plate inserts are welded to the top and bottom of the of the NUHOMS® 61BT basket to prevent the aluminum poison plates from sliding in the axial direction. The geometry of the basket plate inserts is shown on drawing NUH-61B-1064, provided in Section K1.5. The critical locations with respect to thermal stress are in the insert weld locations, since the weld is used to hold the basket plate insert and basket outer wrappers together.

In the basket plate insert regions (at the top and bottom of the basket), there are no poison plates. Therefore, the only thermal stress generated in the insert welds is caused by the differential thermal expansion of the outer wrappers due to the radial temperature gradient of the basket.

In the analysis below, the average temperature of the adjacent 4 compartment, and 9 compartment outer wrappers are found from the thermal analysis from Section K.4 and used to compute the difference in thermal expansion between the two outer wrappers. The highest temperature load cases for the handling/transfer and operation/storage conditions are the vacuum drying condition and blocked vent condition, respectively. These two cases are analyzed below since they are the bounding load cases.

1. Vacuum Drying Case

From the ANSYS results file generated in Section K.4, the average outer wrapper temperatures are computed below.

9 compartment wrapper location average temperature is 632°F.

4 compartment wrapper location average temperature is 623°F.

α_S = Stainless steel coefficient of thermal expansion = 9.61×10^{-6} in./in.°F
E = Stainless steel modulus of elasticity = 25.05×10^6 psi

The length of the outer wrapper analyzed, L, is,

$$L = 3\times6 + 3 + 4\times0.12 + 3\times0.105 + 3\times0.31 + 3\times0.135 = 23.13 \text{ in.}$$

Differential thermal growth, δL, is

$$\delta L = L \times [\,(632\text{-}70) - (623\text{-}70)\,]\,(\alpha_s)$$
$$= 23.13 \times [\,(632\text{-}70) - (623\text{-}70)\,]\,(9.61 \times 10^{-6}) = 2001\times10^{-6} \text{ in.}$$

The pressure generated in the outer wrappers by this differential growth, P, is,

$$P = \varepsilon E = \delta L/L \times E = 2001\times10^{-6} / 23.13 \times 25.05\times10^{6} = 2166.6 \text{ psi.}$$

Assuming that the pressure in the outer wrapper acts over an area equal to 0.105 in. thick × 3.50 in. tall (size of weld insert), then the force applied to the basket plate insert welds, F, is

$$F = (0.105 \times 3.50) \times 2166.6 = 796.2 \text{ lb.}$$

The shear area of the welds, $A = (0.125 \text{ in.} \times \sin(45)) \times 3$ in./insert $\times 4$ inserts = 1.061 in.^{2}
Therefore the shear stress in the weld, $\tau = 796.2$ lb. / 1.061 in.^{2} = 750 psi. $\approx$ 0.75 ksi

2. Blocked Vent Condition

From the ANSYS results file generated in Section K.4, the average outer wrapper temperatures are computed below.

9 compartment wrapper location average temperature is 681°F

4 compartment wrapper location average temperature is 682°F

α_S = Stainless steel coefficient of thermal expansion = 9.69×10^{-6} in./in.°F
E = Stainless steel modulus of elasticity = 24.8×10^6 psi at 700°F

The length of the outer wrapper analyzed, L, is,

$$L = 3 \times 6 + 3 + 4 \times 0.12 + 3 \times 0.105 + 3 \times 0.31 + 3 \times 0.135 = 23.13 \text{ in.}$$

Differential thermal growth, δL, is
$\delta L = L \times [\,(682\text{-}70) - (681\text{-}70)\,]\,(\alpha_s)$
$= 23.13 \times [\,(682\text{-}70) - (681\text{-}70)\,]\,(9.69 \times 10^{-6}) = 224 \times 10^{-6}$ in.

The pressure generated in the outer wrappers by this differential growth, P, is,
$$P = \varepsilon E = \delta L/L \times E = 224\times10^{-6} / 23.13 \times 24.8\times10^{6} = 240.31 \text{ psi.}$$

Assuming that the pressure in the outer wrapper acts over an area equal to 0.105 in. thick × 3.50 in. high (size of weld insert), then the force applied to the basket plate insert welds, F, is

$$F = (0.105 \times 3.50) \times 240.31 = 88.31 \text{ lb.}$$

The shear area of the welds, $A = (0.125 \text{ in.} \times \sin(45)) \times 3$ in./insert $\times 4$ inserts = 1.061 in.^{2}

Therefore the shear stress in the weld, $\tau = 88.31$ lb. / 1.061 in.^{2} = 83.2 psi $\approx$ 0.08 ksi

D. Summary of the Basket Assembly Thermal Stresses

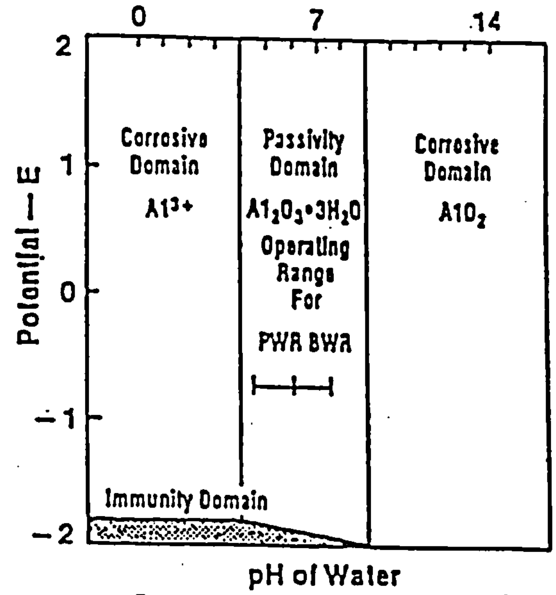
The following table summarizes the basket assembly thermal stresses due to the handling/transfer and storage/operations thermal loads.

Summary of the Basket Assembly Thermal Stresses

THERMAL LOADING	MAXIMUM CALCULATED THERMAL STRESS (KSI)		
	Basket	**Rail**	**Plate Insert**
Handling/Transfer			
-40°F Ambient	11.97	0.81	Enveloped by Vacuum Drying Condition
100°F Ambient	13.25	1.01	Enveloped by Vacuum Drying Condition
Vacuum Drying	18.46	0.83	0.75
Storage/Operations			
-40°F Ambient	8.85	0.91	Enveloped by Blocked Vent Condition
100°F Ambient	11.82	0.78	Enveloped by Blocked Vent Condition
125°F Ambient	12.07	0.79	Enveloped by Blocked Vent Condition
Blocked Vent	20.14	0.89	0.08

These stresses are well below the allowable stresses permitted by the ASME B&PV Code (3 S_m, 3 x 15.2 = 45.2 ksi, S_m at 800°F) and are combined with other loads in Section K.3.6.1.3.3 for handling/transfer loads and Section K.3.6.1.3.4 for operation/storage loads.

At 25°C (77°F):



At 60°C (140°F):

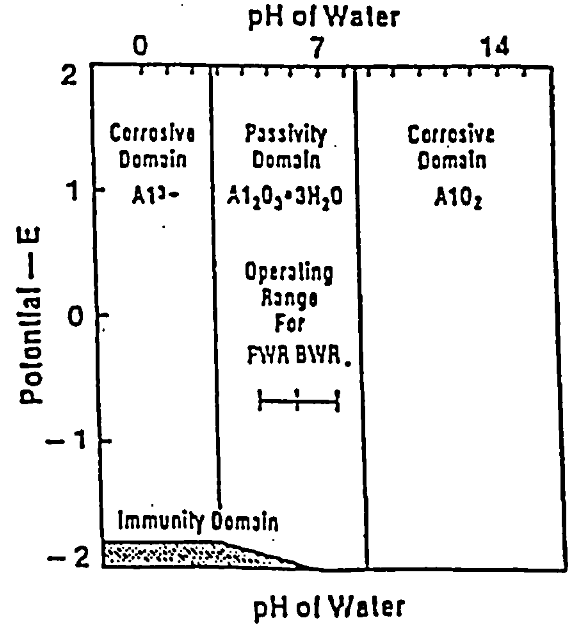


Figure K.3.4-1
Potential Versus pH Diagram for Aluminum-Water System

B. Operation/Storage Loads

The basket loads in the Horizontal Storage Module (HSM) are summarized in the table below. As seen in the table, smaller loads are also conservatively lumped with bigger loads to minimize the analysis effort.

Basket Loads in HSM (Operation/Storage Loads)

Loading	Basket Orientation	Service Level	Load	Enveloped Load for Analysis
Dead Weight	Horizontal	A	1g Down	2g Axial + 2g Trans. + 2g Vertical
Seismic Loads	Horizontal	C	0.37g Axial + 0.37g Trans. + 0.17g Vertical	2g Axial + 2g Trans. + 2g Vertical + Thermal
Thermal(1)	Horizontal	B A B	-40°F Ambient 100°F Ambient 125°F Ambient	-40°F Ambient 100°F Ambient 125°F Ambient
Thermal(1)	Horizontal	D	Blocked Vent	Blocked Vent

(1) The thermal stresses of the basket are addressed in Section K.3.4.

K.3.6.1.3.3 Basket Stress Analysis due to Handling /Transfer Loads

A. Vertical Dead Weight (Basket in Vertical Orientation)

During the1g down loading, the fuel assemblies and fuel compartment are forced against the bottom of the cask. It is important to note that, for any vertical or near vertical loading, the fuel assemblies react directly against the bottom of the canister/cask and not through the basket structure as in lateral loading. It is the dead weight of the basket that causes axial compressive stress during an end drop. Axial compressive stresses are conservatively computed assuming all the weight is taken by the compartment tubes and outer stainless steel wrappers. A conservative basket weight of 23.0 kips (actual weight is 22.92 kips) is used in this analysis.

Compressive Stress at Fuel Compartment Tubes and Outer Wrappers

Total weight = 23.0 kips
Weight excluding hold down ring, SS plate inserts, poison plates, aluminum plates, and rails is calculated to be 12.49 kips

Section area = 12,490 /(164 x 0.29) = 262.62 in^2
Stress due to 1g = -23.0 / 262.62 = - 0.09 ksi

Compressive Stress on Holddown Ring

Weight of hold down ring = 0.94 kips
Section area = 940/(14.5 x 0.29) = 223.5 in^2
Stress due to 1g = -23.0 / 223.5= - 0.1 ksi

This is conservative since for the 1g down case, the basket weight is not applied to the holddown ring.

Shear Stress in Plate Insert Weld

64 (total 128) Inserts support the poison plate weight (3.26 kips)

Load/insert = 3.26 / 64 = 0.051 kips
Weld Shear Area = 0.707 x 3 in. x 0.125 = 0.2651 in^2
Shear stress (1g) = 0.051 / 0.2651 ≈ 0.20 ksi

Shear Stress in Rail Stud

During the 1g down loading, the rail will support its own weight. However, the analysis conservatively assumes that the weight of the rail is supported by the rail studs attached to the fuel compartment tube outer wrappers.

Weight of rails = 5.35 kips
Weld Shear Area = $\pi/4\ (0.5^2 - 0.3^2) = 0.126\ in^2$
Shear stress (1g) = 5.35 / (0.126 × 224) = 0.19 ksi

B. Handling /Transfer Loads – 2g Axial + 2g Transverse + 2g Vertical (Basket in Horizontal Orientation)

The basket finite element model described in Section K.3.6.1.3.1 is used to perform the stress calculations. Since the combined loading (2g axial + 2g transverse + 2g vertical) is non-symmetric, a 360-degree model was used. The canister shell is resting on two rails inside the transfer cask (3" wide x 0.12" thick continuous pad) at 18.5° on either side of basket/canister centerline (see Figure K.3.6-5). The radial contact elements at the two pad locations are assumed closed. The canister nodes at one location of the pad are held in the circumferential direction to avoid rigid-body motion of the model. The contact elements between the pads (between canister and cask from 161.5° to 198.5°) are assumed open with a 0.12" initial gap. The remaining initial gaps are suitably modified (from 0.12"- between 161.5° & 198.5° to 0.63" – at 0°) using the ANSYS macro. The gap elements between the inside surface of the canister and the basket rails are assumed closed at 180° orientation, and remaining initial gaps are suitably modified (from 0 in. at 180°-bottom to 0.25 in. at 0° - top).

Loadings

The 2g vertical load and 2g transverse lateral load resulting from the fuel assembly weight are applied as pressures on the horizontal and vertical faces of plates.

The inertial load due to the basket, rails and canister dead weight is simulated using the density and appropriate 2g acceleration in the vertical and transverse directions. The poison plate weight is included by increasing the basket plate density. Since only a 3" length of the basket assembly is modeled, the acceleration in the axial direction is increased to account for the entire 164" length.

To simulate the axial stress due to the above acceleration, only one side of basket is restrained in the Z – direction.

Analysis and Results

A nonlinear stress analysis is conducted for computing the elastic stresses in the basket model. The nonlinearity of analysis is due to the gaps in the model. The total load is applied in mall steps. The automatic time stepping program option "Autots" is activated. This option lets the program decide the actual size of the load-substep for a converged solution. Displacements, stresses and forces at the final load substep are written to ANSYS result files. Maximum nodal stress intensities in the basket, rails and canister are shown in Figure K.3.6-10 through Figure K.3.6-15 and summarized in the following table.

Stress Summary of the Basket Due to Handling/Transfer Loads

(2g Axial + 2g Transverse + 2g Vertical)

Component	Stress Classification	Stress (ksi)	Reference Figure
Basket	P_m	0.8	K.3.6-10
	$P_m + P_b$	3.67	K.3.6-11
Rail	P_m	1.18	K.3.6-12
	$P_m + P_b$	5.11	K.3.6-13
Canister	P_m	0.7	K.3.6-14
	$P_m + P_b$	7.12	K.3.6-15

C. Summary of Basket Assembly Stress Analysis due to Handling/Transfer Loads

The following table summarizes the basket assembly stress analysis due to the handling/transfer loads. Stresses in the basket assembly due to side drop and end drop accident loads are calculated in Section K.3.7.5.3.

Summary of Basket Structural Analysis due to Handling/Transfer Load Conditions

Loading	Component	Service Level	Stress Classification	Loads	Stress (ksi)	Allowable Stress (ksi)
Vertical Dead Weight	Basket	A	P_m	1g Axial	0.09	16.2
		A	$P_m + P_b$	1g Axial	0.09	24.3
		A	$P_m + P_b + Q$	1g Axial +Thermal	18.55	45.6*
	Plate Insert	A	Shear Stress	1g Axial	0.20	9.72
		A	Shear Stress	1g Axial + Thermal	0.95	45.6
	Rails Stud	A	Shear Stress	1g Axial	0.19	9.72
Horizontal Dead Weight	Basket, Rails, Canister	A	P_m	1g Axial	Enveloped by Handling /Transfer Load	
		A	$P_m + P_b$	1g Axial		
		A	$P_m + P_b + Q$	1g Axial + Thermal		
Handling /Transfer Load	Basket	A	P_m	2g Axial,Vert.,Trans	0.8	16.2
		A	$P_m + P_b$	2g Axial,Vert.,Trans	3.67	24.3
		A	$P_m + P_b + Q$	2g Axial,Vert.,Trans + Thermal	16.92	48.6
	Rails	A	P_m	2g Axial,Vert.,Trans	1.18	16.2
		A	$P_m + P_b$	2g Axial,Vert.,Trans	5.11	24.3
		A	$P_m + P_b + Q$	2g Axial,Vert,,Trans + Thermal	6.12	48.6
	Canister	A	P_m	2g Axial,Vert.,Trans	0.7	16.2
		A	$P_m + P_b$	2g Axial,Vert.,Trans	7.12	24.3
		A	$P_m + P_b + Q$	2g Axial,Vert,,Trans + Thermal	7.12	48.6

*Allowable at temperature during vacuum drying (≈ 800° F)

K.3.6.1.3.4 Basket Stress Analysis due to Operation/Storage Loads

A. Horizontal Dead Weight

The 1g down loading is enveloped by the seismic loads.

B. Seismic Loads

Finite Element Model Analysis of the Basket Due to Seismic Load

The basket finite element model described in Section K.3.6.1.3.1 is used to perform the stress calculations. Since the combined loading (2g axial + 2g transverse + 2g vertical) is non-symmetric, a 360-degree model is used. The canister shell is resting on two rails inside the HSM (3 in. wide x 0.1875 in. thick) at 30° on either side of the basket/canister centerline. The radial contact elements at the two rail locations are assumed closed. The canister nodes at one location of the rail are held in the circumferential directions to avoid rigid-body motion of the model. The gap elements between the inside surface of the canister and the basket rails are assumed closed at the 180° orientation, and remaining initial gaps are suitably modified (from 0 in. at 180° - bottom to 0.25 in. at 0° - top).

increasing their density. Since only lateral modes of vibration are significant, master degree-of-freedoms are applied in the Y-direction only. Figure K.3.6-22 shows the ANSYS finite element model and locations of master degree of freedoms.

Modes and Frequencies From ANSYS Analysis

The first 4 mode frequencies resulting from the ANSYS modal analysis are tabulated below.

Mode	Frequency (Hz.)
1	125.53
2	139.95
3	142.11
4	142.40

The first three (3) mode shapes modal analysis are plotted on Figures K.3.6-23, K.3.6-24 and K.3.6-25.

Results From Hand Calculations

For the first mode shape of each drop, the deformed shape of the central basket panels resembles a simple-simple supported beam.

As an order of magnitude check, the frequency of the fundamental mode of vibration for the simple-simple supported beam is calculated below and compared to the frequency of the first mode of the ANSYS modal analysis results. Reference 3.12, page 369, case 6,"Single span, end supported, uniform load W", lists the following equation for the fundamental frequency:

$$f = 3.55 / (5WL^3/384EI)^{1/2}$$

Where:

$W = 705 \times 3/164 = 12.896$ lbs.
$L = 6.22$ in.
$E = 25.8 \times 10^6$ psi
$I = 2(3.0 \times 0.12^3/12) = 0.000864$ in.^{4}

Substituting the values given above,

$$f = 3.55 / (5 \times 12.896 \times 6.22^3/384 \times 25.8 \times 10^6 \times 0.000864)^{1/2}$$

$$f = 84 \text{ Hz}$$

This value is somewhat lower than that given by ANSYS for the basket. The actual support conditions for the basket are somewhere in between simple-simple and fixed-fixed supports. A fixed-fixed beam's fundamental frequency is approximately double (2.28) that of a simple-simple

supported beam. Therefore, we should expect the ANSYS solution to be somewhere between these values.

Conclusion

Based on the results of modal analysis, it is seen that the lowest natural frequency of the basket is much higher than the threshold frequency of 33 Hz., required for satisfying the rigidity condition. It is also judged that the lowest frequency for other orientations will also be higher than 33 Hz.

E. Summary of Basket Assembly Stress Analysis due to Operation / Storage Loads

The following table summarizes the basket stress analysis results and compares them with the code allowable stresses. The maximum calculated temperature of the basket assembly during storage conditions is less than 550°F (except the blocked vent condition). For conservatism, allowables are taken at a temperature of 650° F. Level A allowables are conservatively used for Level C stresses.

Summary of Basket Structural Analysis due to Operation/Storage Load Conditions

Loading	Component	Service Level	Stress Classification	Loads	Calculated Stress (ksi)	Allowable Stress (ksi)
Horizontal Dead Weight	Basket, Rails, Canister	A	P_m	1g Down	Enveloped by Seismic Loading	
		A	$P_m + P_b$	1g Down		
Horizontal Dead Weight	Basket	A	$P_m + P_b + Q$	Conservatively using 2g Axial + 2g Vert. + 2g Trans.	17.69	48.6
	Rail	A	$P_m + P_b + Q$		11.51	48.6
	Plate Insert	D	Shear	Blocked Vent	0.08	26.38*
Seismic	Basket	C	P_m	2g Axial + 2g Vertical + 2g Transverse	1.46	16.2
		C	$P_m + P_b$		5.62	24.3
	Rails	C	P_m		1.76	16.2
		C	$P_m + P_b$		10.60	24.3
	Canister	C	P_m		4.07	16.2
		C	$P_m + P_b$		12.13	24.3
	Rail Stud	C	Shear	0.37g Axial	3.47	26.63

* Allowable based on 800°F temperature (Vent Block)

K.3.6.1.4 DSC Support Structure Analysis

The DSC support structure is shown in Figures 4.2-6 and 4.2-7. The DSC support rails are supported vertically and horizontally by three moment resisting braced frames anchored to the HSM floor and side wall. The DSC support structure design uses bolted and welded connection details. Normal operating condition loads on the DSC support structure are:

- DSC Dead weight

K.3.7.3.3.1 HSM Frequency Analysis

The lowest horizontal and vertical structural frequencies calculated for a single free standing HSM loaded with a 52B DSC are 21.7 Hz and 47.0 Hz, respectively. An increase in the NUHOMS®-61BT DSC weight of 11% relative to the NUHOMS®-52B DSC results in a conservative frequency shift estimated to be approximately 5%. The adjusted frequencies are 20.7 Hz and 44.7 Hz, respectively. The corresponding horizontal and vertical spectral accelerations are 0.37g and 0.17g.

K.3.7.3.3.2 HSM Seismic Response Spectrum Analysis

The existing HSM structural qualification evaluations provided in Sections 8.1 and 8.2 used a NUHOMS® DSC weight of 80,000 lbs. The weight of the NUHOMS®-61BT DSC is approximately 11% greater. The effects of the increased weight are evaluated by scaling the governing load case stress ratios (or demand/capacity ratios) that are affected by the weight increase to ensure that ratios are less than 1.0. The scaled stress ratios are reported in Table K.3.7-2.

K.3.7.3.3.3 HSM Overturning Due to Seismic

The heavier weight of the NUHOMS®-61BT DSC does not have any effect on the HSM overturning stability due to seismic forces, since the HSM and DSC weight terms cancel out on either side of the overturning equation presented in Section 8.2.3. Thus, the factor of safety against overturning due to seismic remains unchanged at 1.24 as evaluated in Section 8.2.3.

K.3.7.3.3.4 HSM Sliding Due to Seismic

The heavier weight of the NUHOMS®-61BT DSC does not have any effect on the HSM sliding stability due to seismic forces, since the HSM weight terms cancel out on either side of the sliding equation presented in Section 8.2.3. Thus, the factor of safety against sliding due to seismic remains unchanged at 1.34 as evaluated in Section 8.2.3.

K.3.7.3.4 DSC Support Structure Seismic Evaluation

Using the same method discussed in Section K.3.7.3.3, Section 8 results are scaled to account for the heavier NUHOMS®-61BT DSC. The evaluation includes the support frame, cross members, rails, anchor bolts, and cross member connections.

K.3.7.3.4.1 DSC Support Structure Natural Frequency

The lowest structural frequency of the DSC support structure inside the HSM is dominated by the mass of the DSC. The DSC and support structure are included in the HSM analytical model. The dominant horizontal and vertical frequencies of the DSC/DSC support structure reported in Section 8 are 21.7 Hz and 47.0 Hz, respectively. As discussed in Section K.3.7.3.3.1, a conservative frequency shift is estimated to be 5%. The adjusted frequencies are 20.7 Hz and 44.7 Hz.

K.3.7.3.4.2 DSC Support Structure Seismic Response Spectra Analysis

Using the same method discussed in Section K.3.7.3.3.2, the stress ratios in the support frame columns, cross members and rails for the governing load combinations are reported in Table K.3.7-2.

K.3.7.3.5 DSC Axial Retainer Seismic Evaluation

The DSC axial retainer detail, located inside the HSM access opening, is shown on the Appendix E drawings. The retainer bears on the end of the DSC and transfers axial seismic loads to a steel rod/plate assembly that is bolted onto the HSM access opening door. The DSC axial retainer is bolted in place following transfer of the DSC to the HSM and placement of the shielded door.

The clearance between the DSC axial retainer and the DSC is designed for the maximum DSC thermal growth that occurs during the postulated HSM blocked vent case, as discussed in Section 8.2.7. During normal storage there is a small (1/8 to 1/4 inch) gap that will allow movement of the DSC relative to the HSM. This motion produces a small increase in the DSC axial force due to seismic loads, and has been included in the design of the DSC.

The DSC will be subjected to maximum seismic accelerations equal to the rigid range spectral accelerations of 0.37g horizontal and 0.17g vertical. The seismic load acting on the axial retainer is computed considering the spectral accelerations less the friction force between the DSC Support Structure rails and the DSC. The DSC is supported by the DSC Support Structure on rails that are at 30 degrees on either side of the vertical centerline. The rail orientation is considered in determining the normal force in the friction force calculation. The friction force is calculated using the minimum net vertical acceleration, which is the acceleration due to gravity minus the maximum vertical seismic acceleration or (1g – 0.17g) = 0.83g and a coefficient of friction of 0.25. In order to account for the impact load from the DSC onto the axial retainer, an impact factor of 1.50 is considered. The load on the axial retainer is computed as follows:

$$P = (WS_a - F_f)1.50$$

Where,

P	=	seismic load acting on the axial retainer, kips
W	=	DSC weight, conservatively assumed to be 88.4 kips
S_a	=	horizontal rigid range spectral acceleration of 0.37g
N	=	normal force on the rails due to the weight of the DSC = $W_f G_v/\cos 30°$ = (88.4 kips)(0.83g)/cos30° = 84.7 kips
F_f	=	friction force between the DSC and the rails = NC_f = (84.7 kips)(0.25) = 21.2 kips

orientations, membrane stresses were somewhat lower. Accordingly, the maximum shear stress in the rail stud are expected to occur due to the a 90 degree drop orientation. This seems reasonable since during this basket orientation, the fuel weight sits squarely on the largest number of basket panels. The rail stud stresses are therefore computed for a 90-degree side drop orientation. These stresses bound the stud stresses for other basket drop orientations.

The load resulting from the fuel assembly weight was applied as pressure on the basket panels. At the 90° orientation, the pressure acted only on the horizontal plates.

Finite Element Model Description

A three-dimensional finite element model of the basket, rails and DSC were constructed with the following modifications using the finite element model described in Section K.3.6.1.3.

- The couplings at the rail stud locations were replaced with ANSYS Pipe Elements.
- Shear stresses were considered critical in the rail stud weld (O.D. = 0.5" and I.D. = 0.3"). Therefore, the pipe real constant (equivalent thickness) was calculated based on the weld area. The solid stud area is greater than the weld area. Stresses will be lower in the solid area of the stud.
- All material properties, real constants and couplings of the remainder of the model are the same as used for the previous 90° side drop analysis.

The calculated maximum rail stud shear stress for the 90° side drop orientation (75g) is 17.43 ksi. Maximum rail stresses are included in the summary of stresses in Table K.3.7-5.

K.3.7.5.3.2 Basket Assembly Vertical Drop Analysis

During an end drop, the fuel assemblies and fuel compartments are forced against the bottom of the DSC/cask. It is important to note that, for any vertical or near vertical loading, the fuel assemblies react directly against the bottom or top end of the DSC/cask and not through the basket structure as in lateral loading. It is the dead weight of the basket only that causes axial compressive stress during an end drop. Axial compressive stresses are conservatively computed assuming all the weight will be taken by the compartment tubes and wrappers only. A conservative basket weight of 23.0 kips. (actual weight is 22.92 kips) is used in end drop stress calculations.

K.3.7.5.3.2.1 Component Stress Analysis

Compressive Stress At Fuel Compartment Tubes And Outer Wrappers

Total weight = 23.0 kips

Weight excluding holddown ring, SS inserts, poison plates, aluminum plates, and rails is 12.49 kips.

Section area = 12,490 /(164 x 0.29) = 262.62 in^2
Stress due to 1g = -23.0 / 262.62 = - 0.09 ksi

At 75g = - 0.09 ksi × 75 = - 6.75 ksi

Shear Stress in Plate Insert Weld

64 Inserts support the poison plate weight (3.26 kips)

Load/insert = 3.26 / 64 = 0.05 kips
Weld Shear Area = 0.707 x 3 in. x 0.125 = 0.2651 in^2
Shear stress (1g) = 0.05 / 0.2651 = 0.20 ksi
At 75g = 0.20 ksi × 75 = 15.0 ksi

Shear Stress in Rail Stud

During the 75g end drop, the rail will support its own weight. However, the analysis conservatively assumes that the weight of the rail will be supported by the rail studs attached to the compartment outer boxes.

Weight of rails = 5.35 kips
Weld Shear Area = $\pi/4\ (0.5^2 - 0.3^2) = 0.126\ in^2$
Shear stress (1g) = 5.35 / (0.126 × 224) = 0.19 ksi
At 75g = 0.19 ksi × 75 = 14.25 ksi

Compressive Stress On Holddown Ring

Weight of hold down ring = 0.94 kips
Section area = 940/(14.5 x 0.29) = 223.5 in^2
Stress due to 1g = -23.0 / 223.5= - 0.1 ksi
At 75g = - 0.1 ksi × 75 = - 7.5 ksi

Results of Basket End Drop Analysis

Table K.3.7.6 summarizes the basket structural analysis results due to the 75g vertical end drop accident condition.

K.3.7.5.3.2.2 Holddown Ring Buckling Analysis

The buckling of 6.20" x 6.20" box and 12.96" x 12.96" box are evaluated below for 7.5 ksi axial compressive stress.

6.20" x 6.20" Box of Ring

As given in ASME Code, Subsection NF, Paragraph NF-3322-1(c)(2)(a)(Level A Condition) and modified as per Appendix F, Paragraph F-1334 (Level D Condition), the compressive stress limit for the accident condition (Level D) when KL/r is less than 120 and $S_u > 1.2\ S_y$ is:
$F_a = 2 \times S_y\ [0.47 - (KL/r)/444]$

Where:

K = 2.1 as recommended by AISC (Table C1.8.1). The box is

Table K.3.7-5
Stress Summary of the Basket Due to Side Drop Loads – 75G

Drop Orientation	Component	Stress Category	Max. Stress (ksi)	Allowable Stress (ksi)[1]	Reference Figures
45° Side Drop	Basket	P_m	14.54	44.38	Figure K.3.7-6
		$P_m + P_b$	27.12	57.06	Figure K.3.7-7
	Rails	P_m	16.52	44.38	Figure K.3.7-8
		$P_m + P_b$	25.27	57.06	Figure K.3.7-9
	Canister	P_m	2.01	44.38	Figure K.3.7-10
		$P_m + P_b$	19.60	57.06	Figure K.3.7-11
60° Side Drop	Basket	P_m	14.43	44.38	Figure K.3.7-12
		$P_m + P_b$	27.30	57.06	Figure K.3.7-13
	Rails	P_m	20.85	44.38	Figure K.3.7-14
		$P_m + P_b$	28.72	57.06	Figure K.3.7-15
	Canister	P_m	2.44	44.38	Figure K.3.7-16
		$P_m + P_b$	19.57	57.06	Figure K.3.7-17
90° Side Drop	Basket	P_m	18.02	44.38	Figure K.3.7-18
		$P_m + P_b$	22.78	57.06	Figure K.3.7-19
	Rails	P_m	29.03	44.38	Figure K.3.7-20
		$P_m + P_b$	32.79	57.06	Figure K.3.7-21
	Canister	P_m	3.17	44.38	Figure K.3.7-22
		$P_m + P_b$	16.83	57.06	Figure K.3.7-23
	Rail Weld Stud	Shear	17.43	26.63	--
161.5° Side Drop Impact on one Transfer cask Support rail	Basket	P_m	13.47	44.38	Figure K.3.7-24
		$P_m + P_b$	25.76	57.06	Figure K.3.7-25
	Rails	P_m	19.71	44.38	Figure K.3.7-26
		$P_m + P_b$	44.37	57.06	Figure K.3.7-27
	Canister	P_m	3.27	44.38	Figure K.3.7-28
		$P_m + P_b$	23.12	57.06	Figure K.3.7-29
180° Side Drop Impact on two Transfer cask Support rails	Basket	P_m	16.22	44.38	Figure K.3.7-30
		$P_m + P_b$	23.55	57.06	Figure K.3.7-31
	Rails	P_m	28.09	44.38	Figure K.3.7-32
		$P_m + P_b$	34.71	57.06	Figure K.3.7-33
	Canister	P_m	4.72	44.38	Figure K.3.7-34
		$P_m + P_b$	26.13	57.06	Figure K.3.7-35

[1] Allowables are taken at a temperature of 650°F

Table K.3.7-6
Stress Summary of the Basket due to 75g End Drop Load

Drop Orientation	Component	Stress Category	Max. Stress (ksi)	Allowable Stress (ksi)[1]
End Drop	Hold down Ring	P_m	7.5	44.45
End Drop	Basket	P_m	6.75	44.45
	Rail weld Stud	Shear	9.75	26.7
	Plate Insert Weld	Shear	15.0	26.7

[1] Allowable stresses are determined at 650°F.

APPENDIX L

NUHOMS® 24PT2 DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Section L.5 provides the shielding analysis. Section L.6 covers the criticality safety of the NUHOMS®-24PT2 DSC and its contents, listing material densities, moderator ratios, and geometric configurations.

L.1.3 Identification of Agents and Contractors

TN provides the design, analysis, licensing support and quality assurance for the NUHOMS®-24PT2 System. Fabrication of the NUHOMS®-24PT2 System cask is done by one or more qualified fabricators under TN's quality assurance program. TN's quality assurance program is described in Chapter 10. 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 provide training to utility personnel prior to first use of the NUHOMS®-24PT2 System and prepare generic operating procedures.

Managerial and administrative controls, which are used to ensure safe operation of the casks, are provided by the host utility. NUHOMS®-24PT2 System operations and maintenance are 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 [1.4].

L.1.4 Generic Cask Arrays

No change.

L.1.5 Supplemental Data

The following TN drawings are enclosed:

1. General License NUHOMS® 24PT2S-DSC Main Assembly, Drawing NUH-03-1070NP.

2. General License NUHOMS® 24PT2L-DSC Main Assembly, Drawing NUH-03-1071NP.

Information withheld under 10 CFR 2.390(d)

8 7 6 5 4 3 2 1

Information withheld under 10 CFR 2.390(d)

H

G

F

E

D

C

B

A

8 7 6 5 4 3 2 1

Information withheld under 10 CFR 2.390(d)

8 7 6 5 4 3 2 1

Information withheld under 10 CFR 2.390(d)

H G F E D C B A

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

Information withheld under 10 CFR 2.390(d)

L.3.2 Weights

Table L.3.2-1 and Table L.3.2-2 show the weights of the various components of the NUHOMS®-24PT2S and 24PT2L system including basket, DSC, standard HSM, standardized Transfer Cask and OS197 transfer cask. The dead weights of the components are determined based on the nominal dimensions.

Table L.3.2-1
Summary of the NUHOMS®-24PT2S System Component Weights

	Component Description	Calculated Weight (Pounds)
1.	Dry Shielded Canister Shell Assembly	15,778
2.	DSC Top Shield Plug	7,859
3.	DSC Internal Basket Assembly	18,380
4.	DSC Inner and Outer Top Cover Plates	1,934
5.	24 PWR Spent Fuel Assemblies	≤40,368(4)
6.	Weight of Water in DSC Cavity	13,110
	Total Wet DSC Loaded Weight (w/o DSC inner and outer top cover plates.)	95,495
	Total Dry DSC Loaded Weight (w/ DSC inner and outer top cover plates.)	84,319
7.	Standardized Transfer Cask Empty Weight	107,091(1)(3)
8.	Standardized Transfer Cask Max. Loaded Weight	198,099(2)(5)
9.	HSM Single Module Weight, Model 80 (empty)	243,000
10.	HSM Single Module Weight, Model 102 (empty)	253,000

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(1) Includes weight of cask top cover plate assembly.

(2) Weight includes: DSC dry weight plus fuel, plus water in DSC and cask less DSC and cask top cover plate assemblies.

(3) The as-built empty weight for the OS197 transfer cask is 111,250 lbs, including neutron shield water.

(4) The standard design DSC fuel assembly weight of 1,682 lbs/assembly is used.

(5) The maximum loaded weight for the OS197 transfer cask without DSC and OS197 top cover plates is 203,829 lbs (199,249 lbs without neutron shield water).

Table L.3.2-2
Summary of the NUHOMS®-24PT2L System Component Weights

	Component Description	Calculated Weight (Pounds)
1.	Dry Shielded Canister Shell Assembly	14,720
2.	DSC Top Shield Plug	6,526
3.	DSC Internal Basket Assembly	18,420
4.	DSC Inner and Outer Top Cover Plates	1,934
5.	24 PWR Spent Fuel Assemblies	≤40,368(4)
6.	Weight of Water in DSC Cavity	13,350
	Total Wet DSC Loaded Weight (w/o DSC inner and outer top cover plates.)	93,384
	Total Dry DSC Loaded Weight (w/ DSC inner and outer top cover plates.)	81,968
7.	Standardized Transfer Cask Empty Weight	107,091(1)(3)
8.	Standardized Transfer Cask Max. Loaded Weight	195,988(2)(5)
9.	HSM Single Module Weight, Model 80 (empty)	243,000
10.	HSM Single Module Weight, Model 102 (empty)	253,000

(1) Includes weight of cask top cover plate assembly.

(2) Weight includes: DSC dry weight plus fuel, plus water in DSC and cask less DSC and cask top cover plate assemblies.

(3) The as-built empty weight for the OS197 transfer cask is 111,250 lbs, including neutron shield water.

(4) The standard design DSC fuel assembly weight of 1,682 lbs/assembly is used.

(5) The maximum loaded weight for the OS197 transfer cask without DSC and OS197 top cover plates is 201,969 lbs (197,389 lbs without neutron shield water).

L.3.3 Mechanical Properties of Materials

The mechanical properties of structural materials used in the NUHOMS®-24PT2 DSC shell and basket are in accordance with ASME Code Section III, Appendix I [3.1]. The materials used in the fabrication of the 24PT2 DSC shell assemblies are identical to those used in the 24P DSC shell assemblies. The material used in the fabrication of the 24PT2 DSC support rod assemblies is the same as that used in the 52B DSC support rod assemblies. Material properties for the shell and support rod assemblies are provided in Section 8.1. The properties for material used for the 24PT2 spacer discs are presented in Table L.3.6-1. Neutron absorber sheets used in the basket assembly are not relied upon structurally.

L.6.4 Criticality Calculation

L.6.4.1 Calculational Method

L.6.4.1.1 Computer Codes

All calculations are performed using the microcomputer application KENO-5A [6.2] and the Hansen-Roach 16-group (HR-16) cross section working library.

Resonance and heterogeneous effects corrections are performed using the Transnuclear West (TNW) proprietary program PN-HET. PN-HET is used to calculate σ_{peff}, the effective resonance cross section, for each fuel, using the following procedure:

- Calculate σ_{peff} for U235 and U238 in the fuel rods.
- Select H-R library nuclides with σ_{peff} above and below the calculated value.
- Perform a weighted average to accurately represent the resonance nuclide using a mixture of the two selected nuclides.

The benchmark calculations validating this methodology and these codes are documented in Section 3.3.4.2.2.

L.6.4.1.2 Physical and Nuclear Data

The physical and nuclear data required for the criticality analysis are described in Sections L.6.4.1.3 and L.6.4.1.4.

L.6.4.1.3 Fuel Design Parameters

Table L.6-1 summarizes the B&W 15x15 Mark B fuel (most reactive fuel) characteristics used in the criticality analysis.

Table L.6-1
B&W 15x15 Mark B Fuel Characteristics

Fuel assembly design Shorthand description	B&W 15x15 Mark B BW15MB	
Parameter	Value	
Maximum number of fueled rods	208	
Fuel density, % theoretical	95%	
Quantity of guide tubes	16	
Quantity instrument tubes	1	
Parameter	inches	cm
Pellet diameter	0.3686 & 0.370	0.9362 & 0.9398
Active fuel length	141.8	360.1720
Cladding thickness	0.0265	0.0673
Fuel rod OD	0.43	1.0922
Fuel rod pitch	0.568	1.4427
Guide tube OD	0.53	1.3462
Guide tube thickness	0.016	0.0406
Instrument tube OD	0.493	1.2522
Instrument tube thickness	0.026	0.0660

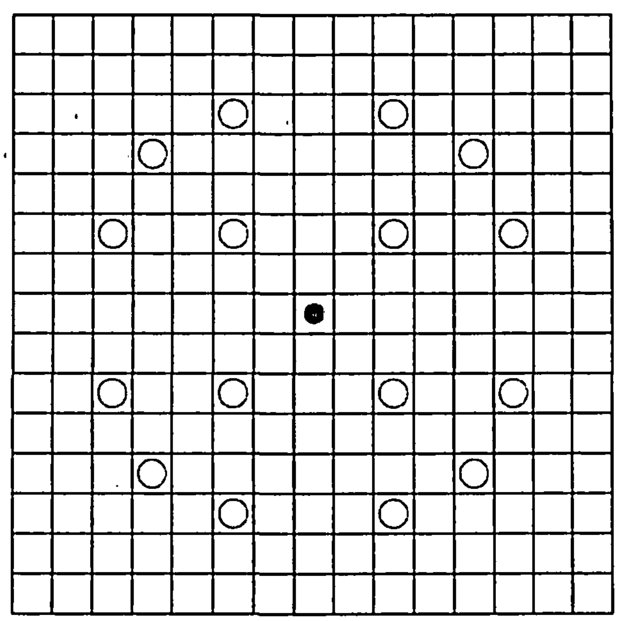

Figure L.6-5
B&W 15x15 Mark B Fuel Assembly