

TRANSNUCLEAR WEST  
STANDARDIZED NUHOMS<sup>®</sup> SYSTEM  
NUHOMS<sup>®</sup>-61BT  
DRY SHIELDED CANISTER  
SAFETY EVALUATION REPORT

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**SAFETY EVALUATION REPORT  
ADDITION OF THE NUHOMS<sup>®</sup>-61BT DRY SHIELDED CANISTER  
AND ADDITIONAL FUEL TYPES**

**DOCKET NO. 72-1004  
MODEL NOS. STANDARDIZED NUHOMS<sup>®</sup>-24P, -52B, AND -61BT  
TRANSNUCLEAR WEST, INC.  
CERTIFICATE OF COMPLIANCE NO. 1004**

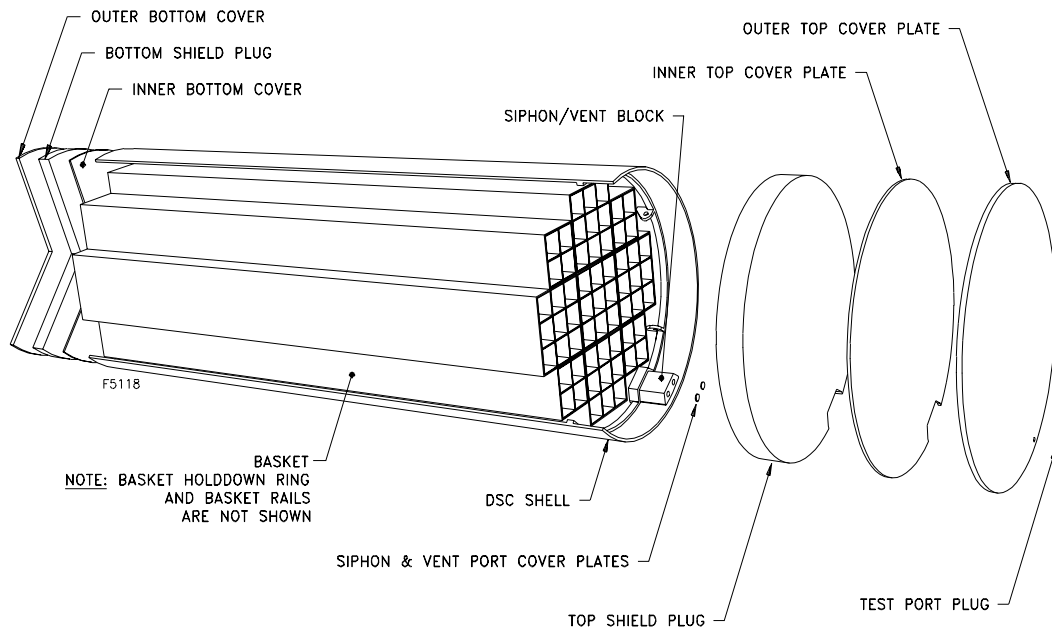
**SUMMARY**

The certificate of compliance (CoC) for the Standardized NUHOMS<sup>®</sup>-24P and -52B System currently limits the number of BWR fuel assemblies per cask to 52. By letter dated July 15, 2000, Transnuclear West, Inc. (TN West) submitted an application to amend Certificate of Compliance 1004 to add the NUHOMS<sup>®</sup>-61BT dry shielded canister (DSC) and additional BWR fuel parameters and damaged fuel to the approved contents of the Standardized NUHOMS<sup>®</sup> System.

The staff performed a detailed safety evaluation of the proposed amendment request which is documented in this safety evaluation report (SER). The staff's evaluation and conclusions regarding the acceptability of the NUHOMS<sup>®</sup>-61BT dry shielded canister for use in the Standardized NUHOMS<sup>®</sup> System are based on information submitted by TN-West on July 15, 2001, as supplemented, requesting an amendment to add the NUHOMS<sup>®</sup>-61BT dry shielded canister to CoC No. 1004. The staff determined that the addition of the NUHOMS<sup>®</sup>-61BT DSC and additional BWR fuel parameters meets the requirements of 10 CFR Part 72, with one exception. The staff determined that the NUHOMS<sup>®</sup>-61BT DSC, as approved in this SER, may not be used to store damaged BWR fuel, that being fuel with greater than hairline cracks and pin hole leaks, as requested in the application.

## 1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the NUHOMS<sup>®</sup>-61BT DSC is to ensure that TN West has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.



NUHOMS<sup>®</sup>-61BT Dry Shielded Canister  
Figure 1

### 1.1 General Description and Operations Features

The NUHOMS<sup>®</sup>-61BT DSC, shown in Figure 1, is a new DSC design which consists of a fuel basket and canister body. Borated aluminum, boron carbide/aluminum metal matrix composite, or Boral<sup>®</sup> plates provide criticality control and heat conduction paths from the fuel assemblies to the cask wall. The canister shell thickness has been reduced from the NUHOMS<sup>®</sup>-52B by 0.125 inches to 0.500 inches and the welded closure has been upgraded to leak tight.

The NUHOMS<sup>®</sup>-61BT DSC will be transferred during loading operations using the previously approved OS-197 transfer cask (TC). Similarly, the NUHOMS<sup>®</sup>-61BT DSC will be stored in the previously approved Standardized NUHOMS<sup>®</sup> System horizontal storage module (HSM). Those components were only reevaluated during this safety evaluation to the extent that they were compatible with the NUHOMS<sup>®</sup>-61BT DSC.

The NUHOMS<sup>®</sup>-61BT DSC was designed to store 61 intact, or a combination of up to 16 damaged and the remainder intact BWR fuel assemblies. However, the staff has determined that the canister may not be used to store damaged fuel as currently designed (as discussed in Section 8 of this SER). The NUHOMS<sup>®</sup>-61BT DSC has been designed but not yet approved for transportation. The new DSC has a maximum heat load of 18.3 kW or 0.3 kW per assembly.

## **1.2 Drawings**

Section K.1 of the SAR contains the non-proprietary drawings for the NUHOMS<sup>®</sup>-61BT DSC, including drawings of the structures, systems, and components (SSC) important to safety. The staff determined that the drawings contain sufficient detail on dimensions, materials, and specifications to allow for a thorough evaluation of the NUHOMS<sup>®</sup>-61BT DSC. Specific SSC are evaluated in Sections 3 through 14 of this SER.

## **1.3 Technical Qualifications of Applicant**

Appendix K, Section K.1.3 of the SAR contains details of the applicant's qualifications and experience regarding its ability to design and fabricate the NUHOMS<sup>®</sup>-61BT DSC in accordance with an approved 10 CFR Part 72 quality assurance program.

## **1.4 Evaluation Findings**

- F1.1 A general description of the NUHOMS<sup>®</sup>-61BT DSC is presented in Appendix K, Section K.1 of the SAR with special attention to design and operating characteristics, unusual or novel design features and principal safety considerations.
- F1.2 Drawings for SSC important to safety are presented in Appendix K, Section K.1 of the SAR. Specific SSC are evaluated in Sections 3 through 12 of this SER.
- F1.3 Specifications for the spent fuel to be stored in the NUHOMS<sup>®</sup>-61BT DSC are provided in SAR Appendix K, Section K.2, Tables K.2-1 and K.2-3. Damaged fuel as specified in Table K.2-2, may not be stored in the NUHOMS<sup>®</sup>-61BT DSC (see Section 8 of this SER for further details). Additional specifications are presented in Appendix K, Chapter K.2 of the SAR and Section 2.0 of the SER.
- F1.4 The technical qualifications of the applicant to engage in the proposed activities are identified in Appendix K, Section K.1.3 of the SAR.
- F1.5 The quality assurance program was previously reviewed and approved for the Standardized NUHOMS<sup>®</sup> System and is referenced in Appendix K, Section 13 of the SAR.
- F1.6 The NUHOMS<sup>®</sup>-61BT DSC has not been certified under 10 CFR Part 71 for use in transportation.
- F1.7 The staff concluded that the information presented in this section of the SAR satisfied the requirements for the general description under 10 CFR Part 72. This finding is



reached on the basis of a review that considered the regulation itself, Regulatory Guide 3.61, and accepted practices.

## **2.0 PRINCIPAL DESIGN CRITERIA**

The objective of evaluating the principal design criteria related to the SSC important to safety is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72.

### **2.1 Structures, Systems, and Components Important to Safety**

The SSCs important to safety are summarized in Appendix K, Table K.2-8 of the SAR. In this table, each component is assigned a safety classification. Only those features that were not previously reviewed and approved by the staff for the Standardized NUHOMS<sup>®</sup> System are addressed in the table.

### **2.2 Design Basis for Structures, Systems, and Components Important to Safety**

The NUHOMS<sup>®</sup>-61BT DSC design criteria summary includes the range of spent fuel types and configurations including the storage of damaged fuel. In addition, the summary includes the enveloping conditions of use and the bounding site characteristics.

#### **2.2.1 Spent Fuel Specifications**

The NUHOMS<sup>®</sup>-61BT DSC is designed to store 61 intact, or up to 16 damaged and the remainder intact, BWR fuel assemblies with or without fuel channels. Appendix K, Tables K.2-1 and K.2-2 provide descriptions of intact and damaged fuel assembly characteristics, and Table K.2-3 provides a list of fuel assembly types. The NUHOMS<sup>®</sup>-61BT has three basket configurations, based on the boron content in the poison plates. Damaged fuel as specified in Table K.2-2, may not be stored in the NUHOMS<sup>®</sup>-61BT (see Section 8 of this SER for further details).

#### **2.2.2 External Conditions**

Appendix K, Section K.2.2 of the SAR identifies the bounding site environmental conditions and natural phenomena for which the NUHOMS<sup>®</sup>-61BT is analyzed. In cases in which the environmental conditions and natural phenomena did not change no descriptions were given. The external conditions are evaluated in Sections 3 through 12 of this SER.

### **2.3 Design Criteria for Safety Protection Systems**

The safety protection systems, a summary of design criteria for the NUHOMS<sup>®</sup>-61BT DSC, are described in Appendix K, Sections K.2.2, K.2.3 and K.2.5, respectively.

#### **2.3.1 General**

The NUHOMS<sup>®</sup>-61BT DSC was designed to provide spent fuel storage for at least 40 years. The Standardized NUHOMS<sup>®</sup> System is certified for 20 years. The internal pressure of the NUHOMS<sup>®</sup>-61BT DSC is always above atmospheric pressure during the storage period to protect against in-leakage of air that could damage the fuel. The welded confinement boundary is verified to be leak tight after loading to ensure the gas cannot escape.

### **2.3.2 Structural**

The structural analysis is presented in Appendix K, Section K.3 of the SAR. The NUHOMS®-61BT DSC is designed to protect the cask contents from significant structural degradation, preserve retrievability, provide adequate shielding, and maintain subcriticality and confinement under credible normal, off-normal, and accident conditions and load combinations. The design requirements for credible normal, off-normal, and accident conditions are defined in Appendix K, Section K.2.2.

### **2.3.3 Thermal**

The thermal analysis is presented in Appendix K, Section K.4 of the SAR. The NUHOMS®-61BT DSC is designed to passively remove decay heat. The fuel cladding integrity is assured by the DSC design which limits fuel cladding temperature and maintains a nonoxidizing environment inside of the canister.

### **2.3.4 Shielding/Confinement/Radiation Protection**

The shielding analysis, confinement analysis and radiological protection capabilities of the NUHOMS®-61BT DSC are discussed in Appendix K, Sections K.5, K.7, and K.10 respectively. The DSC's confinement is obtained with redundant welded closures and is verified through non-destructive examinations at the completion of welding. Radiation exposure is minimized through the shielding capabilities of the previously approved OS-197 transfer cask and the HSM.

### **2.3.5 Criticality**

The criticality analysis is presented in Appendix K, Section K.6 of the SAR. The design criterion for criticality safety is that the effective neutron multiplication factor, including the benchmark bias and modeling bias, does not exceed 0.95 under normal, off-normal and accident conditions. The control method used to prevent criticality is incorporation of poison material in the DSC and a favorable fuel basket geometry.

### **2.3.6 Operating Procedures**

Generic operating procedures are described in Appendix K, Section K.8 of the SAR. This section outlines the loading, unloading, and recovery operations and provides the basis and general guidance for more detailed, site-specific procedures.

### **2.3.7 Acceptance Tests and Maintenance**

The acceptance test and maintenance program for the NUHOMS®-61BT DSC are described in Appendix K, Section K.9 of the SAR, including the commitments, industry standards, and regulatory requirements used to establish the acceptance, maintenance, and periodic surveillance tests.

### **2.3.8 Decommissioning**

There was no change in the decommissioning evaluation due to the addition of the NUHOMS®-61BT DSC.

### **2.4 Evaluation Findings**

The staff concluded that the principal design criteria for the NUHOMS®-61BT DSC are acceptable with regard to meeting the regulatory requirements of 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. A more detailed evaluation of design criteria and an assessment of compliance with those criteria as presented in Sections 3 through 12 of the SER. Damaged fuel as specified in Table K.2-2, may not be stored in the NUHOMS®-61BT DSC (see Section 8 of this SER for further details). Additional specifications are presented in Appendix K, Chapter K.2 of the SAR and Section 2 of the SER.

### **3.0 STRUCTURAL EVALUATION**

This section presents the results of the structural design review of the amendment request for the NUHOMS<sup>®</sup> -61BT DSC to the CoC and the SAR submitted under 10 CFR Part 72, Subpart L<sup>1</sup>. The review was conducted to assess the safety analysis of the structural design features, the structural design criteria, and the structural analysis methodology to evaluate the expected structural performance capabilities under normal operations, off-normal operations, accident conditions and natural phenomena events for those SSCs important to safety. The NUHOMS<sup>®</sup> -61BT DSC is to be utilized in the Standardized NUHOMS<sup>®</sup> System, consisting of the OS 197 TC and the NUHOMS<sup>®</sup> HSM. The evaluation considers only the canister since the TC and the HSM have been previously evaluated and approved for use. The compatibility of the NUHOMS<sup>®</sup> -61BT DSC for use with the TC and the HSM is included in the evaluation.

The review was conducted against the appropriate regulations as described in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified as required by 10 CFR 72.236(a) so that the design basis and design criteria that must be provided for the structures, systems and components important to safety can be assessed under the requirements of 10 CFR 72.236(b). The structural evaluation of the structures, systems and components important to safety must also consider and be compatible with the other specific applicable requirements of 10 CFR 72.236 for maintaining the spent fuel in a subcritical condition, providing adequate radiation shielding and confinement, providing redundant sealing of the confinement system, providing adequate passive heat removal, providing wet or dry transfer capabilities, providing for ease of decontamination, providing for a minimum design life of 20 years, providing for testing or other appropriate means to demonstrate acceptable performance under the design conditions. The structural systems are also evaluated to determine if the DSC is compatible, to the extent possible, for handling and retrievability of the stored spent fuel. The evaluation also must address whether or not the design, fabrication and testing are conducted under a quality assurance program meeting 10 CFR Part 72, Subpart G, as required by 10 CFR 72.234.

#### **3.1 Structural Design of the 61BT DSC**

##### **3.1.1 Structural Design Features**

The NUHOMS<sup>®</sup>-61BT DSC consists of two main structural components that can be described as the cylindrical stainless steel shell confinement vessel that in the transport or storage mode is supported by two rails on the inner surface of the transfer cask. The basket structure is supported laterally by twelve (12) rails against the inner surface of the canister. The twelve (12) rails are bolted with a sliding joint (to prevent thermal restraint) to the periphery of the basket assembly to establish and maintain basket orientation and laterally support the basket. The end shield plugs are fabricated from carbon steel. The NUHOMS<sup>®</sup>-61BT DSC compartment basket assembly provides the lateral structural support for the fuel assemblies and the primary structural portion of the assembly is also stainless steel. The basket is made up of nominal 6" x 6" tube compartments assembled into either 2x2 (4 compartments) or 3x3 (9 compartments) units. The tube compartments are separated by poison plates and the units are wrapped with thin stainless steel plate material to complete the basket assembly. The longitudinal loads from the fuel assemblies are supported by the canister body end cover plates. Other portions of the basket assembly consisting of poison plates are not considered as structural elements for

carrying primary tensile or bending stresses other than sustaining their integrity under their own weight and are therefore considered as structural loads on the basket assembly. The NUHOMS<sup>®</sup>-61BT DSC is the same as the NUHOMS<sup>®</sup>-52B DSC with some dimensional changes to the canister inner volume by a reduction in the material thickness, is designed as a leak-tight confinement with a top outer cover plate with a test port for leakage testing of the top inner cover plate, and has the bottom cover closure weld for the container that is in conformance with Subsection NB of the ASME Boiler and Pressure Vessel Code, Section III, Division 1<sup>2</sup>. The confinement boundary is illustrated in Figure K.3.1-1 of Appendix K and is a positive, fully welded closure system for the container. The basket used in the NUHOMS<sup>®</sup>-61BT DSC represents a new basket design. The canister is not lifted in a loaded condition by its own lifting lugs which are not important to safety, but is handled in the loaded condition by the lifting fittings of the transfer cask.

The classification of the NUHOMS<sup>®</sup>-61BT DSC canister assembly structural elements are clearly delineated in Table K.2-8 for both of the structural components. The individual structural elements are identified as either “important to safety” or as “not important to safety.” The elements of the canister that are considered important to safety include the canister cylindrical shell, inner and outer top and bottom cover plates, the top and bottom shield plugs, the siphon vent block, the siphon/vent port cover plate, the test port plug, the support ring segment and the grapple ring and support. All the elements of the storage basket that are considered important to safety include the fuel compartment, the fuel compartment wrap, the basket plates and inserts, the poison plates, the spacer pads, the basket rail, the alignment leg, the basket holddown plate, the weld studs, washers and hex nuts. Weld rod used for fabrication of the important to safety elements will be in the same classification as the appropriate subsection of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, identified under NB-2400 or NG-2400.

### **3.1.2 Structural Design Criteria**

The NUHOMS<sup>®</sup>-61BT DSC design of the canister confinement vessel is based on the ASME Boiler and Pressure Vessel Code (1998), Section III, Division 1, Subsection NB, Class 1 Components, as identified in Sections K.2.2 and K.2.5 of the amendment, with noted exceptions. The 1999 Addenda to the ASME Code are also incorporated into the design criteria. The specific exceptions have been identified and documented in Table K.3.1-2. In addition, the welded joints between the top inner and outer cover plates and the cylindrical shell are being designed and fabricated in accordance with ASME Code Case N-595-1 by having the root and final passes of the partial penetration welds examined by PT. For normal loading conditions the stress limits will be based on NB-3200 for Level A service limits and for accident loading conditions the stress limits will be based on the Level D service limits.

The NUHOMS<sup>®</sup>-61BT DSC basket design is based on the ASME Boiler and Pressure Vessel Code (1998), Section III, Division 1, Subsection NG, for Core Support Structures, as identified in Sections K.2.2 and K.2.5 of the amendment, with noted exceptions. The 1999 Addenda to the ASME Code are also incorporated into the design criteria. The specific exceptions have been identified and documented in Table K.3.1-3. For normal loading conditions, the stress limits will be based on Level A service limits and for accident loading conditions the stress limits will be based on the Level D service limits. Section III, Division 1, Appendix F may be used

instead of the Level D limits for accident conditions if the stresses do not meet the elastic limits.

### **3.1.2.1 Individual Loads**

Section K.2.2 and Table K.2-10 identify the relevant individual loads, including those resulting from natural phenomena, that the NUHOMS<sup>®</sup>-61BT DSC cask system is designed to resist.

#### **3.1.2.1.1 Dead Loads**

The weight of the fully loaded (dry condition) NUHOMS<sup>®</sup>-61BT DSC is 88,390 pounds with the design value taken as 88,500 pounds. The fully loaded NUHOMS<sup>®</sup>-61BT DSC with water (wet condition) is approximately 101,800 pounds. These loads are considered for the design of the system in all of its possible orientations.

#### **3.1.2.1.2 Live Loads**

The live loads considered for the design of the NUHOMS<sup>®</sup>-61BT DSC are the normal handling loads associated with lifting the cask, placing the cask in the TC, downending the cask in the TC to a horizontal orientation, moving the cask in the TC with the transport trailer, removal from the transport system, and hydraulic insertion into the HSM or extraction from the HSM. The transfer loads include the following loads: axial load of +/- 1.0g, transverse load of +/- 1.0g, vertical load of +/- 1.0g, and under a combined condition of all loads of +/- 0.5g in each of the three directions. The normal design insertion load into the HMS acting axially on the 61BT DSC is 80,000 pounds and the extraction load is 60,000 pounds. The off-normal design loads for both insertion and extraction are 80,000 pounds acting axially on the 61BT DSC.

#### **3.1.2.1.3 Pressure Loads**

The design internal pressure for normal conditions is 10 psig and for the off-normal conditions is 20 psig. The internal test pressure of 12 psig that is applied without the NUHOMS<sup>®</sup>-61BT DSC outer top cover plate in place. The accident internal pressure is 65 psig. Table K.4-5 provides the maximum internal pressures during normal, off-normal, and accident conditions there were used in the design of the NUHOMS<sup>®</sup>-61BT DSC.

#### **3.1.2.1.4 Thermal Loads**

The thermal loading is based on the NUHOMS<sup>®</sup>-61BT DSC containing spent fuel rejecting 18.3 kW decay heat with the ambient air temperature range of -40°F to 125°F. The thermal evaluation of normal conditions, off-normal conditions and accident conditions are provided in Section K.4 of the amendment, with Tables K.4-1, K.4-2 and K.4-3 providing the calculated temperatures under these various loading conditions. These temperature extremes are expected to occur only for short periods of time, on the order of hours. The range of 0°F to 100°F expected to bound the temperatures that could exist for a period of days. The lifetime average temperature ambient is taken as 70°F. Thermal conditions are also calculated for other conditions of operation as described in Section K.3.4.4. The design is based on providing adequate clearances between the fuel, the basket and the canister shell that experience temperature differentials and allow free thermal expansion.

#### **3.1.2.1.5 Flood Loads**

Flood loading is addressed in Section K.2.2.2 of the amendment. The NUHOMS<sup>®</sup>-61BT DSC cask system is designed for flood water to a depth of 50 feet and water velocity of 15 fps, consistent with the NUHOMS<sup>®</sup>-24P DSC and the NUHOMS<sup>®</sup>-52B DSC systems.

#### **3.1.2.1.6 Tornado Wind and Tornado Missiles**

The NUHOMS<sup>®</sup>-61BT DSC cask system is designed for the same tornado wind loads and tornado missiles as the NUHOMS<sup>®</sup>-24P DSC and NUHOMS<sup>®</sup>-52B DSC systems. The NUHOMS<sup>®</sup>-61BT DSC system is evaluated for a design basis tornado wind velocity of 360 mph with a translational velocity of 70 mph and a pressure drop of 3 psig as discussed in Section 3.2.1 of the Standardized NUHOMS<sup>®</sup> System FSAR. Tornado missiles are listed in Section 3.2.1.2 of the FSAR.

#### **3.1.2.1.7 Seismic**

The design earthquake for the NUHOMS<sup>®</sup>-61BT DSC system is based on an earthquake that produces a horizontal ground acceleration of 0.25g and a vertical acceleration of 0.17g. The location of these accelerations is taken at the top of the concrete pad/basemat of the HSM. NRC Regulatory Guides 1.60 and 1.61 are utilized in the seismic design.

#### **3.1.2.1.8 Snow and Ice**

The environmental loads on the NUHOMS<sup>®</sup>-61BT DSC canister and basket from snow and ice are negligible or zero and do not have to be considered since either the TC or HSM will be the loaded component in the NUHOMS<sup>®</sup>-61BT DSC system from snow or ice. Loads for the HSM are provided in Section 3.2.4 of the FSAR.

#### **3.1.2.1.9 Lightning**

The environmental effect on the NUHOMS<sup>®</sup>-61BT DSC canister and basket from lightning will be negligible and does not have to be considered since either the TC or the HSM will surround and protect the canister and its internals from lightning.

#### **3.1.2.1.10 Fire and Explosion**

The NUHOMS<sup>®</sup>-61BT DSC system contains no flammable material and the concrete and steel used for the system fabrication can withstand any credible fire hazard. No explosive materials are present in the fission products or cover gases. Externally initiated explosions are considered to be bounded by the design basis tornado generated loads and are evaluated in Section K.11 of the SAR. In order to utilize the NUHOMS<sup>®</sup>-61BT DSC, licensees are required by 10 CFR 72, Subpart K, to confirm that no conditions exist near the ISFSI that would result in pressures due to off-site explosions which would exceed those postulated for tornado wind or missile effects.



### 3.1.2.2 Loading Combinations

The NUHOMS® 61BT DSC system is subjected to the same loads and load combinations as the existing NUHOMS®-24P DSC or NUHOMS®-52B DSC systems. The loading combinations are provided in Table K.2-5 and Table K.3.7-15. The loading combinations reflect the various operational conditions and events that may occur during the lifetime of the utilization of the NUHOMS®-61BT DSC and the design calculations reflect these combinations. The loading combinations include the following cases:

- Non-operational events
- Fuel loading
- Draining/Drying
- Transfer Trailer Loading
- Transfer to/from ISFSI
- Storage
- Operational events
- Natural Phenomena events

Table K.3.6-1 shows the normal operating loads for which the safety-related/important to safety components are designed. Table K.3.6-2 provides the same information for the off-normal operating loads. The loading combinations represent the design events identified by ANSI/ANS 57.9-1984, "Design Criteria for an Independent Spent Fuel Storage Installation" and are in accordance with NRC Regulatory Guide 3.48. These design events are defined in the Standardized NUHOMS® System FSAR in Section 8.1 and 8.2. For the accident events the considerations to be made are based on the accident analysis scenarios identified in Section K.11

### 3.1.2.3 Allowable Stresses

The allowable stresses for the NUHOMS®-61BT DSC canister shell are based on the ASME Boiler and Pressure Vessel Code (1998), Section III, Division 1, Subsection NB-3200, for normal and off-normal conditions and Appendix F to Section III for accident conditions. The 1999 Addenda is also incorporated into the design bases. Section K.2.2.5.1.1 and Table K.2-6 provide the detailed guidance for the stress allowables under the various loading conditions and events, including normal, off-normal and accident conditions. The stress allowables are identified with respect to each category of stress, whether a primary membrane stress such as induced by internal pressure, primary membrane plus bending that can occur in the shell geometry transition regions, or a bearing stress. Also the various service levels (Level A through Level D) are identified. It is noted that the stress allowables are also based on the temperature conditions of the material that will exist under the specific service conditions. Fatigue considerations are also made for the normal loads that include repetitive loads.

The allowable stresses for the fuel basket assembly are also based on the ASME Boiler and Pressure Vessel Code (1998), Section III, Division 1, Subsection NG-3200, including the 1999 Addenda, for normal and off-normal conditions and Appendix F of Section III for accident conditions. Section K.2.2.5.1.2 and Table K.2-7 provide the detailed guidance for the stress allowables under the various loading conditions and events, including normal, off-normal and accident conditions. The allowable stresses are identified with respect to each category of

stress, whether pure primary shear or a buckling compressive stress. Provisions are identified for addressing such conditions as service temperatures, fatigue, and impact loadings. The numerical values of the stress limits for service at 650 degrees F for normal and accident conditions are provided in Table K.3.1-1.

### **3.1.3 Materials**

#### **3.1.3.1 Fixed Neutron Absorbers**

Four types of fixed neutron absorbers have been included in the SAR. These are designated as follows:

1. Enriched Borated Aluminum Alloy: a wrought aluminum alloy containing boron, which has been isotopically enriched to 95 % B10, as an alloy addition. This material referred to, in Sections 6 and 9, as alloy and borated-aluminum alloy.
2. Three different commercial aluminum matrix composite materials are permitted. Each is formed from a blend of two powders: (1) an aluminum alloy and (2) boron carbide in the form of particles of B<sub>4</sub>C produced from natural boron, i.e. the boron has not been enriched:
  - Boralyn<sup>®</sup>: a hot-vacuum-pressed product that has been formed into plates from a billet of blended aluminum alloy powders and fine particles ( <25 micrometers in diameter) of B<sub>4</sub>C.
  - Metamic<sup>®</sup>: a cold-pressed and sintered product that has been formed into plates from a billet of aluminum alloy powders and fine particles of B<sub>4</sub>C.
  - Boral<sup>®</sup>: a product, containing a blend of aluminum 1100 alloy powders and coarser particles (average diameter of 85 micrometers) of B<sub>4</sub>C. The blend is placed into a box that is formed from sheet material of aluminum 1100 alloy. This box is then formed that is formed into plate or sheet material.

The fixed neutron absorbers serve as a neutron absorber for criticality control and as a heat conduction path; the NUHOMS<sup>®</sup>-61BT DSC safety analyses do not rely upon their mechanical strength, as the basket structural components surround the plates on all sides.

#### **3.1.3.2 Canister Materials**

The material properties used in the structural analyses are in accordance with the ASME Boiler and Pressure Vessel Code (1998), Section II, Part D, with the 1999 Addenda. Tables K.3.6-3 and K.3.7-7 provide the basic mechanical properties of the stainless steel material. In the structural analysis, the bilinear behavior of the SA-240, 304 stainless steel was utilized based on the properties identified in Section K.3.6.1.3.1.B and Table K.3.6-3. The durability of the canister shell, basket, and other assembly components of stainless steel will allow the material to perform its design function beyond the design life of the NUHOMS<sup>®</sup>-61BT DSC system. In addition, the basket and the interior of the canister shell are under a constant inert helium gas

environment once the spent fuel has been loaded and the system sealed with the final structural and confinement welds.

## **3.2 Normal and Off-Normal Conditions**

### **3.2.1 Analysis Methods**

The NUHOMS<sup>®</sup>-61BT DSC assembly was analyzed using the finite element method of the ANSYS software package. The model was developed by creating two separate models, one each of the top and bottom half-length of the canister shell, and utilizing the symmetry of the shell so that only a quadrant was idealized for the model. This is an acceptable modeling technique because of the known stress conditions for most loadings such as internal pressure based on classical unique numerical solutions. With this information it is possible to verify the acceptability of the model in representing the shell halves in the more difficult regions to define stresses without computer capability. Boundary conditions can be imposed to duplicate the prototype behavior. Figures 8.1-14a and 8.1-14b of the Standardized NUHOMS<sup>®</sup> System FSAR present the two models used for the analyses. The models were three-dimensional models that included the details of the cylindrical shell and the closure plates. For one specific loading condition, addressing the loading arising during the horizontal loading and unloading of the NUHOMS<sup>®</sup>-61BT DSC, via the grapple ring, another finite element model was used as shown in Figure 8.1-15.

The NUHOMS<sup>®</sup>-61BT DSC basket assembly was also analyzed using a three-dimensional finite element model using the ANSYS software package. The model represented the canister shell, the rails, and the basket. Only a 3 inches long section or disk thickness of the NUHOMS<sup>®</sup>-61BT DSC system was analyzed as representative of the actual physical prototype. This technique is acceptable based on the boundary conditions that can be imposed at the model edges that can be related to the full physical prototype. The model used is shown in Figure K3.6-1. Figure K.3.6-2 illustrates the basket compartments, Figure K3.6-3 shows the outer wrap of the basket compartments for the 2x2 and 3x3 modules, and Figure K.3.6-4 shows the support rails whose one surface conforms to the inner surface of the cylindrical shell and the other face conforms to the outer perimeter of the basket assembly. The finite element model also included a representation of the gap spaces that would exist between the basket rails and the inner surface of the canister as well as the gaps that would exist between the canister and the transport cask, so that the handling and drop loadings could be analyzed. Figures K.3.6-6 through K.3.6-9 illustrate this analysis capability for the detailed analytical model.

### **3.2.2 Loading Cases Analyzed**

The normal operating load cases analyzed for the canister included the dead weight loads, design internal pressure, design external pressure, design basis thermal loads, operational handling loads and design basis live loads. In order to complete the analysis for the operational handling loads there were actually two situations considered. The first addressed the inertia loads associated with on-site handling and transporting the DSC between the fuel handling/loading area and the HSM, and the second associated with loading the DSC into (or removing the DSC from) the HSM. In addressing the second situation, a conservative coefficient of friction was taken as 0.25 in order to compute projected insertion and extraction loads into the HSM. Each loading case was analyzed for the 61BT DSC in all of the key

orientations. It is noted that the A-36 steel shield plugs are not specifically analyzed since they are free to expand thermally and serve only as a mass for shielding.

The normal loading cases analyzed for the basket assembly included the dead weight loads, the thermal loads, and handling/transfer loads. In addition, individual elements within the basket assembly were analyzed for the loads to determine results such as the compressive stress on the holddown ring; the basket compartment tube and outer wrapper compressive stress, shear stresses in the insert plate weld and the shear stress in the rail stud. Section K.3.6.1.3.2 provides a tabular summary of the basket loads in the transfer cask resulting from the handling/transfer conditions and for the basket loads in the HSM as the operation/storage loads.

### **3.2.3 Analysis Results**

The results of the various analyses are shown graphically in Figures K.3.6-10 through K.3.6-15 for the handling/transfer loads. On page K.3.6-12, the summary of the resulting stresses are provided for the various loadings. In addition, the allowable stresses are shown. All computed stresses are well within the allowable stress values for the Service Level A conditions. The stresses for the operation/storage load conditions are shown in tabular form on page K.3.6-16. The final summary of stress maximums is provided in Table K.3.6-4 for the normal and off-normal loads for the various elements of the canister shell and the basket. Table K.3.7-11 provides the summary results for the enveloping loading cases for the normal and off-normal conditions for Service Levels A and B. The minimum margin against the allowables exists in the canister shell for the primary plus secondary stresses with a 1% margin. The next smallest margin is 12% in the membrane plus bending stress for the outer bottom cover plate. The remainder of the margins are in excess of 30%.

## **3.3 Accident Conditions**

### **3.3.1 Analysis Methods**

The analysis methods include static and dynamic analyses utilizing elastic and elasto-plastic methods, as well as classical methods and numerical methods such as finite element methods. The specific analytical methods used are identified for the particular structural element, component, or assembly being analyzed and the selection of the method for use is influenced by the complexity of the structure, the importance of the structure, the loading conditions, and other characteristics.

The finite element analysis methods described herein in Section 3.2 were also utilized in the analysis of accident conditions, in fact many of the same idealized models were used for evaluating the accident conditions, with the only difference being the loading conditions imposed. All methods of analysis used for evaluation of the NUHOMS<sup>®</sup>-61BT DSC are accepted methods and have been previously used for similar analyses.

### 3.3.2 Loading Cases Analyzed

Section K.3.7 of Amendment 3 addresses the accident loads which also, in this document, encompasses the loads resulting from natural phenomena. The following loading cases have been addressed.

- a. Reduced HSM air inlet and outlet shielding
- b. Debris blockage of HSM air inlet and outlet opening
- c. Accidental transfer cask drop with loss of neutron shield
- d. Pressurization due to fuel cladding failure within the DSC
- e. Postulated DSC leakage
- f. Design basis flood
- g. Tornado winds and tornado generated missiles
- h. Lightning effects
- i. Design basis seismic event

Loading Case a. is bounded by Loading Case b. for the thermal effects on the structural aspects of the scenario. In addition, the thermal effects of Loading Case b. are considered in the NUHOMS<sup>®</sup>-61BT DSC system and the impact on the canister is encompassed in the analyses, and the increased thermal loads on the HSM are bounded by the thermal effects of the NUHOMS<sup>®</sup>-24P DSC system that is already addressed by the current CoC.

Loading Case c., with the loss of the neutron shield has no direct impact on the structure, but the initiating event of the transfer cask drop is considered. The components of the NUHOMS<sup>®</sup>-61BT DSC system that are evaluated due to their influence on structural performance are the canister shell, the basket, and the on-site transfer cask. The drop scenarios for the design are for a horizontal side drop from a height of 80" with the vertical end drop being an 80" drop on the top or the bottom of the transfer cask. The corner drop considered for an 80" drop at 30 degrees to the horizontal was found to be enveloped by the end and side drops. The cask side drops consider the various orientations with respect to the two support rails of the canister so as to bound the possible maximum stress orientation for stresses within the canister shell. In addition, for a conservative assumption, the entire fully loaded weight of the DSC is assumed to be on one support rail. For the vertical drop effects on the canister shell, it is conservatively assumed that no energy is absorbed by the cover plates. Inertia loadings are based on the forces associated with the 75g deceleration value used for the Standardized NUHOMS<sup>®</sup> System. The canister shell was also analyzed for buckling under the vertical drop loads. In addition to the analyses for the canister shell, the basket assembly and its various components were also analyzed under these loading conditions for the dropped transfer cask scenario. The specific components analyzed were the holddown ring, the fuel compartments, the outer wrapper of the fuel compartments, the basket rails, the bolts/studs connecting the basket rails to the fuel compartments and the poison plate support insert welds.

Loading Case d. results in a computed maximum internal pressure of 46.0 psig in the canister shell which remains below the accident design pressure of 65.0 psig.

Loading Case e. has no structural loading implications.

Loading Case f. is the result of specific design bases selected for the cask which must not be exceeded at the location where the 61BT DSC system is used.

Loading Case g. is the result of the specific analysis of the NUHOMS<sup>®</sup> system since the 61BT DSC should not be directly exposed to tornado effects.

Loading Case h. has no structural loading implications.

Loading Case i. Is addressed using Regulatory Guides 1.60 and 1.61. Natural frequencies are determined for the shell bending mode as well as the shell ovaling mode. Spectral accelerations are determined for use in analysis of the seismic loads on the DSC shell and the internal basket. For this loading it was necessary to re-evaluate the HSM for compatibility with the NUHOMS<sup>®</sup>-61BT DSC since it has a larger weight than the NUHOMS<sup>®</sup>-24P DSC and NUHOMS<sup>®</sup>-52B DSC systems.

### **3.3.3 Analysis Results**

Tables K.3.7-12 and K.3.7-13 provide the summary results for the enveloping loading cases for the accident load conditions for Service Levels C and D respectively. Based on the calculated stresses for the various components of the NUHOMS<sup>®</sup>-61BT DSC system, the components with stresses nearly equal to the allowable are the outer bottom cover plates. These components have a safety margin of approximately 5% for the membrane plus bending stress under both Service Levels C and D. The inner bottom cover plate and the canister shell also have a 5% margin on the allowables for the membrane plus bending stress under Service Level D conditions. The remainder of the margins are nearly all in excess of 20%. The results of the analyses performed for the 75g side drop and end drop loading conditions are provided in Tables K.3.7-5 and K.3.7-6 and show a significant margin with respect to the allowable stress limits.

### **3.4 Evaluation Findings**

- F3.1 The SSCs important to safety are described for the NUHOMS<sup>®</sup>-61BT DSC System in sufficient detail to enable an evaluation of their structural effectiveness and are designed to accommodate the combined loads of normal, off-normal, accident and natural phenomena events.
- F3.2 The NUHOMS<sup>®</sup>-61BT DSC System is designed to allow handling and retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage of the system that will prevent retrieval of the stored spent nuclear fuel. However, previously damaged fuel as specified in Table K.2-2, may not be stored in the NUHOMS<sup>®</sup>-61BT DSC (see Section 8 of this SER for further details). Additional specifications are presented in Appendix K, Chapter K.2 of the SAR and Section 2.0 of the SER.
- F3.3 The cask is designed and fabricated so that the spent nuclear fuel is maintained in a subcritical condition under credible conditions. The configuration of the stored spent fuel is unchanged. Additional criticality evaluations are discussed in Section 6 of this SER.

- F3.4 The cask and its systems important to safety are evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.
- F3.5 The staff concludes that the structural design of the NUHOMS®-61BT DSC system is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the NUHOMS®-61BT DSC system will enable safe storage of spent nuclear fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable industry codes and standards, accepted practices and confirmatory analysis.

### **3.5 References**

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72; Subpart L, "Approval of Spent Fuel Storage Casks."
2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, 1998 including the 1999 Addenda.

## **4.0 THERMAL EVALUATION**

The staff reviewed the NUHOMS<sup>®</sup>-61BT DSC thermal design and evaluation to assess whether the cask and fuel material temperatures will remain within their allowable values or criteria for normal, off-normal, and accident conditions as required in 10 CFR Part 72<sup>1</sup>. This amendment was also reviewed to determine whether the NUHOMS<sup>®</sup>-61BT DSC fulfills the acceptance criteria listed in Section 5 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."<sup>2</sup>

### **4.1 Cask System Thermal Design**

The NUHOMS<sup>®</sup>-61BT DSC is designed to store BWR fuel assemblies with a maximum decay heat of 300 watts per assembly, or a total of 18.3 kW. The DSC is inserted and backfilled with helium at the time of loading. The DSC is designed to passively reject decay heat during storage and transfer for normal, off-normal and accident conditions while maintaining temperatures and pressures within specified regulatory limits.

### **4.2 Thermal Models**

The applicant modeled the performance of the NUHOMS<sup>®</sup> -61BT DSC using the ANSYS computer code. The geometry of the DSC was modeled three dimensionally. The three dimensional models represent 90° and 180° symmetrical sections of the DSC and include the geometry and material properties of the basket components, the basket rails, and the DSC. The model simulates the effective thermal properties of the fuel with a homogenized material occupying the active fuel length. For normal and off-normal conditions of storage and transfer, the 90 degrees model is used due to the lack of a large circumferential temperature gradient. The applicant chose the upper 90° quadrant which provided the highest temperatures. For the transfer cases, the highest calculated DSC boundary conditions are applied to the entire DSC. The blocked vent case uses a 180° model to allow the large surface temperature variations to be modeled.

### **4.3 Thermal Analysis**

Normal and off-normal thermal analyses were previously performed by the applicant and reviewed by the staff for the NUHOMS<sup>®</sup> -52B DSC within the HSM. This included the analysis of a) maximum normal ambient temperature of 100°F with insolation; b) minimum off-normal ambient temperature of -40°F without insolation; and c) maximum off-normal ambient temperature of 125°F with insolation. The analysis for the NUHOMS<sup>®</sup> -52B DSC used a decay heat load of 19.2 kW to determine the temperature distributions for the cask. These temperature distributions, which bound the 18.3 kW heat load for the NUHOMS<sup>®</sup>-61BT DSC, are applied as boundary conditions for the finite element models for normal and off-normal conditions of storage.

The analysis for the NUHOMS<sup>®</sup>-61BT DSC within the OS197 TC was performed for the following ambient conditions: a) maximum normal ambient temperature of 100°F with insolation; b) minimum off-normal ambient temperature of -40°F; and c) vacuum drying under an ambient of 100°F without insolation.



Accident analysis for the NUHOMS<sup>®</sup>-61BT DSC is based on the previously analyzed HSM model in the Standardized NUHOMS<sup>®</sup> System FSAR with a maximum ambient temperature of 125°F, with maximum insolation and with the HSM vents totally blocked for 40 hours. The analysis assumed a total decay heat of 18.3 kW per DSC and calculated the surface temperature of the DSC during the blocked vent accident. Additionally, the applicant analyzed a postulated worst case fire accident assuming a 300 gallon diesel fire for a NUHOMS<sup>®</sup> 61BT DSC with a decay heat load of 18.3 kW during transfer in an OS197 TC.

#### **4.4 Evaluation of Cask Performance for Normal Conditions**

The temperatures in the Standardized NUHOMS<sup>®</sup> System HSM and transfer cask are bounded by the existing analysis in the FSAR because of the higher heat load for the NUHOMS<sup>®</sup>-24P DSC or the NUHOMS<sup>®</sup>-52B DSC designs. The maximum calculated temperature of the fuel cladding during storage was 569°F which is below the allowable fuel temperature of 649°F. The maximum calculated temperature of the fuel cladding during transfer was 638°F which is below the allowable fuel temperature of 1058°F. The maximum temperature of the NUHOMS<sup>®</sup>-61BT DSC is 827°F and occurs after 96 hours of vacuum drying during loading and unloading operations. This temperature is below the allowable fuel temperature of 1058°F. The maximum calculated pressure is 9 psig (for transfer) which is below the design pressure of 10 psig for normal conditions of storage and transfer.

#### **4.5 Evaluation of Cask Performance for Off-Normal Conditions**

The temperatures in the NUHOMS<sup>®</sup> HSM and transfer cask are bounded by the existing analysis in the FSAR because of the higher heat load for the NUHOMS<sup>®</sup>-24P DSC or the NUHOMS<sup>®</sup>-52B DSC designs. The maximum calculated temperature of the fuel cladding was 590°F which is below the allowable fuel temperature of 1058°F. The calculated pressure is 11.5 psig (during transfer) which is below the design pressure of 20 psig for off-normal conditions of storage and transfer.

#### **4.6 Evaluation of Cask Performance for Accident Conditions**

The temperatures in the NUHOMS<sup>®</sup> HSM and transfer cask are bounded by the existing analysis in the FSAR because of the higher heat load for the NUHOMS<sup>®</sup>-24P DSC or the NUHOMS<sup>®</sup>-52B DSC designs. The maximum calculated temperature of the fuel cladding was 809°F which is below the allowable fuel temperature of 1058°F. The calculated pressure is 46 psig which is below the design pressure of 65 psig for accident conditions.

The analysis of the hypothetical fire accident shows a maximum DSC surface temperature of 499°F which is below the blocked vent case maximum temperature of 662°F. Therefore, the NUHOMS<sup>®</sup> 61BT DSC temperatures and pressures calculated for the blocked vent case bound the hypothetical fire accident case.

#### **4.7 Evaluation Findings**

F4.1 Appendix K, Section K.7 of the SAR describes SSCs important to safety to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

- F4.2 The NUHOMS®-61BT DSC is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.
- F4.3 The spent fuel cladding is protected against degradation leading to gross ruptures by maintaining cladding temperatures below 1058°F. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.
- F4.4 The staff concludes that the thermal design of the NUHOMS®-61BT DSC is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the cask will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

#### **4.8 References**

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.
2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

## **5.0 SHIELDING EVALUATION**

The staff reviewed the capability of the NUHOMS<sup>®</sup> 61BT DSC to provide adequate protection against direct radiation from the canister contents when used with the Standardized NUHOMS<sup>®</sup> System. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), and 10 CFR 72.236(d)<sup>1,2</sup>. Because 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations, an overall assessment of compliance with these regulatory limits is evaluated in Section 10 of this SER. This amendment was also reviewed to determine whether the NUHOMS<sup>®</sup>-61BT DSC fulfills the acceptance criteria listed in Section 5 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."<sup>3</sup>

### **5.1 Shielding Design Features and Design Criteria**

The applicant requested the addition of a new storage canister, the NUHOMS<sup>®</sup>-61BT DSC, for use with the NUHOMS<sup>®</sup> Horizontal Modular Storage System which includes the Horizontal Storage Module (HSM) and the OS 197 TC. There were no changes to the HSM which is described in Revision 3 of the Standardized NUHOMS<sup>®</sup> System FSAR. Therefore, the HSM and transfer cask are not reviewed here except as to how they relate to the NUHOMS<sup>®</sup>-61BT DSC. The NUHOMS<sup>®</sup>-61BT DSC will be used to store up to 61 BWR fuel assemblies which are described in Table K.5-1 of SAR Appendix K.

#### **5.1.1 Shielding Design Features**

The NUHOMS<sup>®</sup>-61BT DSC, when used with the Standardized NUHOMS<sup>®</sup> System provides both gamma and neutron shielding during loading/unloading, transfer, and storage operations. The 61BT DSC consists of a 0.5-inch thick steel canister with a 5-inch thick steel bottom shield plug, and a 7-inch thick steel top shield plug. The OS 197 TC, as depicted in drawing NUH-03-8000-SAR, consists of a steel shell, lead shielding, and a water jacket. The HSM is constructed of thick concrete walls and a shielded access door. The HSM air inlet paths are designed to preclude radiation streaming.

The staff evaluated the NUHOMS<sup>®</sup>-61BT DSC shielding design features and found them acceptable. The applicant's analysis provides reasonable assurance that the shielding design of the NUHOMS<sup>®</sup>-61BT DSC, when used with the Standardized NUHOMS<sup>®</sup> System, provides reasonable assurance that the shielding design meets the regulatory requirements of 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

#### **5.1.2 Shielding Design Criteria**

The overall radiological protection design criteria are the regulatory dose requirements in 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), and maintaining occupational exposures as-low-as-reasonably-achievable (ALARA). The applicant analyzed the NUHOMS<sup>®</sup>-61BT DSC loaded with spent fuel as described in Section K.2.1 and Table K.5-1 of Appendix K of the SAR.

The SAR analysis provides reasonable assurance that the NUHOMS<sup>®</sup>-61BT DSC can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Dose rates must meet the limits incorporated into the technical specifications.

## **5.2 Source Specification**

The source specification is presented in Section K.5.2 of SAR Appendix K. The gamma and neutron source term calculations were performed with the ORIGEN2 computer code. The fuel types considered in this application are listed in Table K.5-1. The GE 7x7 was chosen as the design basis fuel assembly as it has the highest initial heavy metal loading (0.198 MTU). Source terms were then calculated for the design basis assembly for the four proposed burnup/enrichment combinations. The bounding gamma and neutron source terms were then combined in the shielding models to calculate the dose rates.

To correct for changes in the neutron flux outside the fuel zone during irradiation, the masses of the materials in the bottom end fitting, plenum and top end fitting were multiplied by scaling factors of 0.15, 0.2, and 0.1, respectively. These are the scaling factors recommended in Reference 4 and are considered to provide bounding values.

Axial peaking factors are taken from the TN-68 FSAR which has been approved by NRC to store similar fuel. These peaking factors were determined based on typical axial burnup distributions for BWR assemblies using axial water density distributions found during core operations. The data provided burnup and moderator density for 25 axial locations along the assembly, which the applicant collapsed into 12 axial zones. SAS2H was used to calculate the source terms for each zone of the design basis fuel with the power and water density varied in each zone. The relative source distributions are shown in Figure K.5-1 of the SAR.

### **5.2.1 Gamma Source**

Gamma source terms are calculated for each burnup/enrichment combination and are listed in Tables K.5-7 through K.5-10. The applicant determined that the 27 GWd/MTU, 2.0 wt% U-235, 5-year cooled fuel resulted in the design basis gamma source term. This combination had the largest number of particles in the energy groups between 0.8-2.0 MeV which are the most important energy groups for the surface dose rates.

The hardware activation analysis considered the cobalt impurities in the assembly hardware. The cobalt content is listed in Table K.5-1. The activated hardware source terms are calculated using the hardware masses listed in Table K.5-5. Although cobalt impurities can vary, the applicant's assumed values are reasonable and acceptable. The gamma source also includes the contribution from the fuel channels.

### **5.2.2 Neutron Source**

Neutron source terms are calculated for each burnup/enrichment combination and are listed in Tables K.5-7 through K.5-10. The applicant determined that the 35 GWd/MTU, 2.65 wt% U-235, 8-year cooled fuel resulted in the design basis neutron source term. This combination produced the largest number of particles. The staff notes that the neutron source term in the upper half of the fuel may be underestimated due to the water densities used in the neutron

source calculation. However, this is offset by the applicant's conservative gamma source term calculations which is discussed below. Additionally, the off-site dose rates are dominated by the gamma radiation.

### **5.2.3 Confirmatory Analyses**

The staff reviewed the proposed contents and the assumed hardware cobalt impurities listed in Table K.5-1 of Appendix K of the SAR. The staff has reasonable assurance that the design basis gamma and neutron source terms for the NUHOMS<sup>®</sup>-61BT DSC are acceptable for the shielding analysis. The staff also reviewed the flux scaling factors for the hardware source terms and found them to be appropriate.

The staff performed confirmatory calculations of the source terms for the specified fuel types, burnup conditions, and cooling times. The staff used the OrigenS module of SCALE 4.4 and the associated 27 neutron, 18 gamma group cross section library. The staff's overall source term calculations were in general agreement with the applicant's calculations. While the staff's neutron source terms were slightly higher than the applicant's, the staff's gamma source terms were much lower than the applicant's. Differences are expected due to the use of different codes and assumptions.

The exterior dose rates are adequately controlled by limits in the CoC for maximum burnup, minimum cooling time, and maximum dose rates.

## **5.3 Shielding Model Specifications**

The shielding analysis was performed with DORT, a 2-D discrete ordinates code to calculate the dose rates on and around the HSM and transfer cask. To determine the total off-site dose, the MCNP computer code was used. The off-site dose models include 1) a 2x10 array of HSMs and, 2) two 1x10 arrays (facing front-to-front) loaded with design basis fuel in NUHOMS<sup>®</sup>-61BT DSCs.

### **5.3.1 Shielding and Source Configuration**

The shielding source is divided into 12 axial regions as summarized in Table K.5-14. The top and bottom 10% of the assembly is divided into 2 zones each and the remaining 80% is divided into 8 equal zones. Volumetric sources are developed for all fuel regions. The source is divided into the following regions; active fuel, bottom end fitting, and top end fitting. The axial distribution of the gamma and neutron sources is assumed to follow the relative burnup profile depicted in Figure K.5-1. The fuel channel material and most of the basket materials are conservatively neglected in the shielding model which reduces the amount of actual shielding and results in a bounding dose rate. A number of other simplifications and bounding assumptions that reduce the amount of actual shielding are discussed in Section K.5.4.

The analysis includes streaming paths through the HSM air vents and the transfer cask-DSC gap. The overall design eliminated other potential streaming paths. Evaluation of streaming from narrow and long holes is difficult for DORT. While DORT is subject to ray effects, this tends to over-predict radiation streaming.

### **5.3.2 Material Properties**

The composition and densities of the materials used in the shielding analysis are presented in Tables K.5-15 through K.5-19. The homogenized fuel assembly region accounts for the uranium dioxide; zircaloy cladding and spacers; and steel present in the in-core region of the assembly, the basket inner fuel compartment, and the outer wrapper materials. The BWR fuel channels and all other components of the basket were ignored. The materials used in the HSM were previously reviewed and accepted by the staff.

The staff evaluated the shielding model and found it acceptable. The material compositions and densities used were appropriate and provide reasonable assurance that the 61BT DSC was adequately modeled.

## **5.4 Shielding Analyses**

### **5.4.1 Computer Programs**

The applicant's shielding analysis was performed with DORT and is presented in Section K.5.4 of Appendix K of the SAR. The cross section data used are based on the CASK-81 22 neutron, 18 gamma energy group coupled cross section library.

### **5.4.2 Flux-to-Dose-Rate Conversion**

The SAR uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose rate conversion factors to calculate dose rates.

### **5.4.3 Normal Conditions**

The Appendix K of the SAR presents calculated dose rates for normal condition design-basis dose rates for the HSM in Tables K.5-2 and K.5-3. The dose rates for the HSM are dominated by the gamma component. This is expected due to the thick concrete walls of the HSM. The calculated dose rates are below the dose rate criteria specified in the TS, except for the dose rate at the HSM door centerline. However, due to the conservative assumptions in the shielding analysis, the staff has reasonable assurance that the user will be able to meet the HSM dose rate TS limits.

The Appendix K of the SAR also presents calculated dose rates for the transfer cask in Tables K.5-2 and K.5-4. While the gamma component dominates the dose rates, there is still a significant contribution from neutron radiation. The dose rates for the transfer cask assume that there is four inches of supplemental shielding on top of the DSC during welding. Table K.5-2 also gives the surface peak dose rate at the top of the DSC as approximately 3990 mrem/hr. Exposure from localized peak dose rate may be mitigated by the actual locations of personnel and use of temporary shielding during loading/unloading operations. Figures K.5-12 through K.5-15 of the SAR present dose profiles for the transfer cask at various distances, which show that the dose rates significantly decrease from peak locations to the edges of the top, bottom, and sides of the cask. The calculated average dose rates are below the dose rate criteria specified in the TS, thus the staff has reasonable assurance that the user will be able to meet the TS limits for the transfer cask dose rates.

#### **5.4.4 Accident Conditions**

Appendix K of the SAR does not identify an accident that significantly degrades the shielding of the HSM. The bounding accident condition for the HSM considers sliding of an HSM, which creates a 12-inch gap between the concrete HSMs. SAR table K.11-1 shows that the maximum dose rate for this is approximately  $2.4 \times 10^{-3}$  mrem/hr at 600 meters for a 2x10 array of HSMs. The estimated recovery time for this accident is 5 days. Therefore, the estimated dose to a person at 600 meters from the ISFSI would be approximately 0.3 mrem which meets the requirements of 10 CFR Part 72.

The bounding accident condition for the transfer cask considers loss of water from the transfer cask water jacket combined with damaged fuel. This accident results in an increase by a factor of three in the estimated dose rates. SAR Table K.11-4 shows that the maximum dose rate for this is approximately 1750 mrem/hr at 1 meter from the cask surface. For an 8 hour recovery time, the estimated dose rate to a member of the public at 600 meters is approximately 0.2 mrem which meets the regulatory requirements.

#### **5.4.5 Occupational Exposures**

The analysis in Appendix K of the SAR used the design basis fuel to estimate occupational exposures for the NUHOMS<sup>®</sup>-61BT DSC. Section 10 of the SAR presents the estimated occupational exposures that are based on dose rate calculations in Section 5 of Appendix K to the SAR. The staff's evaluation of the occupational exposures is in Section 10 of this SER.

#### **5.4.6 Off-site Dose Calculations**

Section K.10 of the SAR estimates the offsite dose rates from a 2x10 and two 1x10 arrays. Tables K.10-6 through K.10-8 present the calculated offsite annual doses for these arrays at distances of 6 to 600 meters based on 100% occupancy exposure time. These generic off-site calculations demonstrate that the 61BT DSC is capable of meeting the offsite dose criteria of 10 CFR 72.104(a).

Section 10 of this SER evaluates the overall off-site dose rates from the NUHOMS<sup>®</sup>-61BT DSC. The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by general licensees. The general licensee must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance. The actual doses to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel characteristic, cask-array configurations, topography, demographics, atmospheric conditions. In addition, 10 CFR 72.104(a) includes doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of the general licensee.

A general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public as required, by evaluation and measurements.

An engineered feature for radiological protection, such as a berm, is considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

### **5.4.7 Confirmatory Calculations**

The staff performed confirmatory analyses of selected dose rates using SAS4. The staff based its evaluation on the design features and model specifications presented in Appendix K of the SAR. Limiting fuel characteristics and the burnup and cooling time are included in the TS, as are the dose rates of the transfer cask and HSM. The staff's calculated dose rates were in close agreement with the SAR values and were generally lower due to the applicant combining the worst case source gamma source term from one fuel assembly with the worst case neutron source term from another assembly. The staff found that the SAR has adequately demonstrated that the 61BT DSC is designed to meet the criteria of 10 CFR 72.104(a).

### **5.5 Evaluation Findings**

- F5.1 Appendix K of the SAR sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.
- F5.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.
- F5.3 The NUHOMS<sup>®</sup>-61BT DSC is designed to provide redundant sealing of the confinement system.
- F5.4 The staff concludes that the design of the radiation protection system of the NUHOMS<sup>®</sup>-61BT DSC, when used with the HSM, is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the NUHOMS<sup>®</sup>-61BT DSC will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

### **5.6 References**

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.
2. U.S. Code of Federal Regulations, "Standards for Protection Against Radiation," Title 10, Part 20.
3. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.
4. Luksic, A.T., et al., "Revised Uranium-Plutonium Cycle PWR and BWR Model for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, Oak Ridge, TN, 1978.



## 6.0 CRITICALITY EVALUATION

The staff reviewed the NUHOMS<sup>®</sup>-61BT DSC cask system criticality analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that storage of spent fuel in the -61BT cask system meets the following regulatory requirements: 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g)<sup>1</sup>. Amendment 3 of the SAR was also reviewed to determine whether the NUHOMS<sup>®</sup>-61BT DSC cask system fulfills the following acceptance criteria listed in Section 6 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."<sup>2</sup>

- The multiplication factor ( $k_{eff}$ ), including all biases and uncertainties at a 95% confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.
- At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety under normal, off-normal, and accident conditions should occur before an accidental criticality is deemed to be possible.
- When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.
- Criticality safety of the cask system should not rely on the use of the following credits:
  - burnup of the fuel,
  - fuel-related burnable neutron absorbers, and
  - more than 75% for fixed neutron absorbers when subject to standard acceptance tests. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber as discussed in Section 9 of this SER are required.

### 6.1 Criticality Design Criteria and Features

The design criterion for criticality safety is that the effective neutron multiplication factor,  $k_{eff}$ , including statistical biases and uncertainties, shall not exceed 0.95 for all postulated arrangements of fuel within the cask under normal, off-normal, and accident conditions.

The NUHOMS<sup>®</sup>-61BT DSC cask system design features that are relied upon to prevent criticality during cask loading and unloading are the basket geometry and fixed neutron absorbers, which are present in the form of borated metallic plates. The minimum allowable Boron-10 areal density within these plates as a function of maximum fuel assembly lattice average enrichment are provided in Table 6-1.

**Table 6-1 - Minimum Boron-10 Areal Density as a Function of Maximum Fuel Assembly Lattice Average Enrichment**

| Maximum Fuel Assembly Lattice Average Enrichment (wt% U-235) | Minimum Boron-10 Areal Density for Boral and Metamic (g/cm <sup>2</sup> ) | Minimum Boron-10 Areal Density for Borated Aluminum and Boralyn (g/cm <sup>2</sup> ) | Areal Density Used in the Criticality Evaluation: 90% Credit [75% Credit for Boral and Metamic] (g/cm <sup>2</sup> ) |
|--|---|--|--|
| <b>Intact Fuel Assemblies</b>                                |   |  |  |
| 3.7  | 0.025   | 0.021  | 0.019  |
| 4.1  | 0.038   | 0.032  | 0.029  |
| 4.4  | 0.048   | 0.040  | 0.036  |
| <b>Failed Fuel Assemblies</b>                                |   |  |  |
| 4.0  | 0.048   | 0.040  | 0.036  |

As presented in Table 6-1, the applicant took credit for 75% (Boral and Metamic) to 90% (Borated Aluminum and Boralyn) of the minimum specified Boron-10 areal density in the basket poison material for the criticality calculations. The fabrication requirements and acceptance criteria for the fixed neutron poison, which justify the use of absorber credit exceeding 75% for the basket's fixed neutron poison, are outlined in SAR Section K.9. During storage, the NUHOMS®-61BT DSC is designed to prevent water from entering the cask cavity, which maintains subcriticality.

The staff reviewed the NUHOMS®-61BT DSC design criteria and features discussed in Sections K.1.2, K. 2.3, and K.6 of the SAR and verified that the design features important to criticality safety are clearly identified and adequately described. The staff verified that the amendment contains engineering drawings, figures, and tables that are sufficiently detailed to support an in-depth staff evaluation.

The staff also verified that the design-basis off-normal and postulated accident events would not have an adverse effect on the design features important to criticality safety. Therefore, based on the information provided in SAR, the staff concludes that the NUHOMS® -61BT cask system design with DSC meets the double contingency requirements of 10 CFR 72.124(a).

## **6.2 Fuel Specification**

The NUHOMS® -61BT DSC is designed to store standard boiling water reactor (BWR) fuel assemblies with or without fuel channels. The application requested that the DSC be allowed to store 61 intact BWRs or up to 16 damaged BWRs with the remainder being intact BWR assemblies.

Damaged fuel is defined as spent nuclear fuel with known or suspected cladding defects greater than a hairline crack or a pinhole leak. Per NRC guidance, damaged fuel should be canned for storage. The purpose of canning is to confine gross fuel particles to a known, subcritical volume during off-normal and accident conditions, and to facilitate handling and retrievability. However, the staff determined that, the NUHOMS<sup>®</sup>-61BT DSC as approved in this SER does not meet the guidelines for handling and retrievability as described in Interim Staff Guidance (ISG) -1, "Damaged Fuel."<sup>3</sup> Therefore, the NUHOMS<sup>®</sup>-61BT DSC may not be used to store BWR fuel with greater than hairline cracks and pinhole leaks as requested in the application (see Section 8 of this SER for further details). Notwithstanding, this SER Section assess the SAR's criticality analysis for damaged fuel.

The applicant provides the bounding BWR fuel assembly parameters in Tables K.2-1, K.2-2, K.2-3, and K.2-4 the SAR. The assembly types analyzed are intact 7x7, 8x8, 9x9, and 10x10 and damaged 7x7 and 8x8 BWR fuel assemblies manufactured by General Electric. The maximum initial enrichment is presented in Table 6-1 as a function of minimum allowable Boron-10 areal density. The applicant performed a variety of calculations to verify that criticality safety is maintained for these fuel types in the NUHOMS<sup>®</sup> -61BT DSC.

The staff reviewed the fuel parameters considered in the criticality analysis and verified that they bound the specifications given in Section K.2 of SAR. The staff verified that all fuel assembly parameters important to criticality safety have been adequately presented.

### **6.3 Model Specification**

#### **6.3.1 Configuration**

The applicant initially performed criticality analyses to determine the most reactive intact and damaged fuel assemblies. The applicant determined that the GE12 10x10 BWR fuel assembly is the most reactive intact fuel in the domain of allowed fuel assemblies. The GE2 7x7 and GE9 8x8 fuel assemblies were determined to be the most reactive damaged fuels.

In a second set of analyses, the applicant modeled the full-active fuel height and full radial cross section of the cask and DSC with reflective boundary conditions on all sides. The GE12 10x10 BWR fuel assembly was incorporated into this model. The applicant assumed that the cask neutron shield and stainless steel skin are stripped away and replaced with moderator for the hypothetical accident condition (HAC).

Finally, the applicant modeled 45 intact fuel assemblies and 16 failed fuel assemblies in the four 2x2 compartments in the corners of the DSC basket. The parameters and assumptions utilized in this model are presented in Section K.6.3.1 of Amendment 3. The NRC staff determined that the applicant's treatment of damaged fuel assemblies in the model was appropriate.

The applicant evaluated the variations in system reactivity for the above models as a function of several factors. These factors included moderator density, neutron poison plate thickness, assembly-to-assembly pitch, fuel cladding thickness, canister shell thickness, gap thickness between poison plates, etc. The applicant did not take any credit for fissile depletion due to burnup or fission product poisoning.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in Sections K.1 and K.2, including engineering drawings. The staff also reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. Based on the information presented in Amendment 3, the staff agrees that the most reactive combination of cask parameters and dimensional tolerances was incorporated into the calculation models.

### **6.3.2 Material Properties**

The compositions and densities for the materials used in the computer models are provided in Section K.6.3.2 of the SAR. The minimum required areal density of the Boron-10 is provided as a function of maximum fuel assembly lattice average enrichment (wt% U-235). The applicant's criticality calculations modeled 75% to 90% of the relevant minimum Boron-10 specification. In Section K.9 of SAR, the justification for the use of 90% credit is given, along with acceptance tests for the fabrication of the neutron absorber sheet materials.

There are three types of NUHOMS<sup>®</sup> -61BT DSC baskets, each identical with the exception of minimum Boron-10 content in the poison plates (see Table 6-1). Only one type of plate is utilized in a specific DSC, based on the maximum enrichment of the fuel that will be placed in the DSC. The neutron absorber materials that may be utilized by the DSC include borated aluminum, Boralyn<sup>®</sup>, Metamic<sup>®</sup>, and Boral<sup>®</sup>.

The fabricated plates meet the thermal requirements and can be expected not to have significant erosion or corrosion under ISFSI service. A structural analysis was performed which demonstrates that the basket plates will remain in place during all design basis accident conditions.

The compositions and densities for the materials used in the computer models were reviewed by the staff and found acceptable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

## **6.4 Criticality Analysis**

### **6.4.1 Computer Programs**

The applicant utilized the CSAS25/KENO-V.a module of the SCALE-4.4 computer code<sup>4</sup> and the accompanying 44-group ENDF/B-V cross-section library for the NUHOMS<sup>®</sup> -61BT cask criticality analyses and benchmark calculations. The CSAS25/KENO-V.a code is frequently used by the industry for performing criticality analyses. The NRC staff agrees that the CSAS25/KENO-V.a module and cross-section set used in the subcriticality design evaluation are appropriate for this particular application and fuel system.

### **6.4.2 Multiplication Factor**

The results of the applicant's criticality analyses show that keff of the NUHOMS<sup>®</sup> -61BT cask system with DSC will remain below 0.95 for all allowed fuel loadings. The staff reviewed the applicant's calculated keff values and Upper Subcritical Limit (USL) and agrees that these values have been appropriately calculated to include all biases and uncertainties at a 95%

confidence level or better. The NRC staff reviewed and determined that the applicant's CSAS25/KENO-V.a code, modeling methodology, input parameters, and assumptions provide satisfactory criticality analysis results.

Based on the applicant's criticality evaluation, as reviewed and verified by the NRC staff, the NRC staff concludes that the NUHOMS<sup>®</sup>-61BT cask system with DSC will remain subcritical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

### **6.4.3 Benchmark Comparisons**

The applicant performed benchmark comparisons on 125 uranium oxide critical experiments, which were chosen to bound the variables in the NUHOMS<sup>®</sup>-61BT cask design with DSC. The benchmark problems used to verify the criticality computations are representative of benchmark arrays of commercial light water reactor (LWR) fuels with the following characteristics:

- water moderation;
- boron neutron absorbers;
- unirradiated light water reactor-type fuel (no fission products or "burnup credit") near room temperature; and
- close reflection.

The staff reviewed the benchmark comparisons in Amendment 3 and agrees that the CSAS25/KENO-V.a module of the SCALE-4.4 computer code used for the analysis was adequately benchmarked to representative critical experiments.

An USL of 0.9414 was calculated by the applicant. The USL incorporates the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any keff less than the USL is less than 0.95, which is the design criterion.

The staff reviewed the applicant's method for determining the USL and found it to be acceptable and conservative. The staff also verified that only biases that increase keff have been applied.

### **6.6 Evaluation Findings**

Based on the NRC staff's review of the NUHOMS<sup>®</sup>-61BT DSC SAR, the staff concludes that the NUHOMS<sup>®</sup>-61BT cask system with DSC meets the acceptance criteria specified, for both intact and damaged fuel, in NUREG-1536. Notwithstanding, the storage of damaged fuel in the NUHOMS<sup>®</sup>-61BT DSC does not meet the guidelines of ISG-1 and, therefore, the staff denies the applicant's request to store spent fuel with damage exceeding hairline cracks and pinhole leaks in the NUHOMS<sup>®</sup>-61BT DSC (see Section 8 of this SER for more details). In addition, the staff finds the following:

- F6.1 SSCs important to criticality safety are described in sufficient detail in Sections K.1, K.2, and K.6 of the SAR and on the design drawings to enable an evaluation of their effectiveness.
- F6.2 The NUHOMS®-61BT DSC is designed to be subcritical under all credible conditions.
- F6.3 The criticality design is based on favorable geometry and fixed neutron poisons.
- F6.4 The NRC staff concludes that the criticality design features for the NUHOMS®-61BT DSC are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the NUHOMS®-61BT DSC will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

## **6.7 References**

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.
2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.
3. ISG -1, "Damaged Fuel," Rev 0, May 1999.
4. Scale 4.4, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory, September, 1998.

## 7.0 CONFINEMENT EVALUATION

The staff reviewed the NUHOMS<sup>®</sup>-61BT DSC confinement features and capabilities to ensure a) that any radiological releases to the environment will be within the limits established by the regulations<sup>1</sup>, and b) that the spent fuel cladding will be protected against degradation that might lead to gross ruptures during storage, as required in 10 CFR 72.122(h)(1). This amendment was also reviewed to determine whether the NUHOMS<sup>®</sup>-61BT DSC fulfills the acceptance criteria listed in Section 7 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems"<sup>2</sup>. The staff's conclusions are based on information provided in the NUHOMS<sup>®</sup>-61BT DSC SAR.

### 7.1 Confinement Design Characteristics

A description of the confinement boundary is given in Sections K.1.2.1, K.2.5, K.3.1.2.1, K.7.1.1, and Figure K.3.1-1 of the amendment request. The confinement boundary includes the stainless steel shell, the top and bottom closure assemblies (including the vent and drain system), and the associated welds. The inner top cover plate has two penetrations for the vent and siphon ports which are closed with welded cover plates. The outer top and bottom cover plates provide redundant sealing of the confinement system. The outer top cover plate has a single penetration to leak test the closure welds. This is closed with a welded cover plate after testing to complete the redundant sealing of the confinement boundary. The welds forming the confinement boundary are described in detail in Sections K.3.1.2.1 and K.7.1.3 of the SAR. The redundant closure of the DSC satisfies the requirements of 10 CFR 72.236(e) for redundant sealing of confinement systems.

The DSC is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code Section III, Subsection NB to the maximum extent practicable. Exceptions to the ASME code are listed in SAR Table K.3.1.2.3. The staff concludes that the description of the confinement boundary satisfies the requirements of 10 CFR 72.24(c)(3).

The applicant's proposed procedures for drying and evacuating the cask interior during loading operations were reviewed by the staff to ensure that the design is acceptable for the pressures that may be experienced during storage. The staff finds that this design, if fabricated and tested in accordance with the SAR requirements, will maintain the confinement boundary. Maintaining a stable vacuum pressure of 3 mm Hg for 30 minutes during vacuum drying provides reasonable assurance that an acceptable level of moisture remains in the NUHOMS<sup>®</sup>-61BT DSC.

The NUHOMS<sup>®</sup>-61BT DSC is designed to be leaktight and is tested to a leak rate of  $1 \times 10^{-7}$  atm cm<sup>3</sup>/sec, as defined in "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5-1997. This testing confirms that the amount of helium lost from the NUHOMS<sup>®</sup>-61BT DSC over the approved storage period is negligible. Thus, an adequate amount of helium will remain in the canister to maintain an inert atmosphere and to support the heat transfer during the storage period.

For normal storage conditions, the NUHOMS<sup>®</sup>-61BT DSC uses multiple confinement barriers provided by the fuel cladding (for intact fuel) and the NUHOMS<sup>®</sup>-61BT DSC to assure that the confinement system will reasonably maintain confinement of radioactive material. The canister

is backfilled with an inert gas (helium) to protect against cladding degradation. Section 3 of the SER shows that all confinement boundary components are maintained within their code-allowable stress limits during normal storage conditions. Section 4 of the SER shows that the peak confinement boundary component temperatures and pressures are within the design-basis limits for normal conditions of storage. Weld examinations include the following; multiple surface and volumetric examinations, pneumatic pressure testing, and leakage rate testing on the finished shell and the inner cover plate at the fabricator; leakage rate testing of the closure welds (inner top cover plate and vent and siphon port cover plates) after loading the spent fuel; and multiple surface and dye penetrant examinations on the redundant confinement boundary. The applicant described the canister inspection and test acceptance criteria in Section K.9 of the SAR. The closure weld examination and acceptance criteria are included in Sections 1.2.4.a and 1.2.5 of the TS. The staff finds that this is acceptable provided that all NDE personnel, both at the fabricator and at the loading site, are qualified in accordance with applicable standards and codes such as SNT-TC-1A. This is a requirement of ASME Section V, Article 1, Paragraph T-140.

The all-welded construction of the NUHOMS<sup>®</sup>-61BT DSC with the redundant closure, extensive inspection and testing, ensures that no release of radioactive material for normal storage and on-site transfer will occur.

## **7.2 Confinement Monitoring Capability**

For redundant seal welded closures, continuous monitoring of the closure is not necessary because there is no known plausible, long-term degradation mechanism which would cause the seal welds to fail. Periodic surveillance and monitoring of the storage module thermal performance, as well as the licensee's use of radiation monitors are adequate to ensure the continued effectiveness of the confinement boundary. The staff finds this adequate to enable the licensee to detect any closure degradation and take appropriate corrective actions to maintain safe storage conditions.

## **7.3 Nuclides with Potential Release**

Since the NUHOMS<sup>®</sup>-61BT DSC is designed, fabricated, and tested to meet the leak tight criteria of "American National Standard for Radioactive Materials," ANSI N14.5-1997, there is no contribution to the radiological consequences due to a potential release of canister contents.

## **7.4 Confinement Analysis**

The confinement boundary is welded and tested to meet the leak tight criteria of "American National Standard for Radioactive Materials," ANSI N14.5-1997 and is shown to maintain confinement during all normal, off-normal, and hypothetical accident conditions. Also, the temperature and pressure of the canister are within the design-basis limits. Therefore, no discernable leakage is credible. As discussed in Section 10 of this SER, the staff finds that the 61BT DSC meets the requirements of 10 CFR 72.104(a) and 10 CFR 106(b).



## **7.5 Supportive Information**

Supportive information or documentation includes drawings of the NUHOMS®-61BT DSC confinement boundary and applicable pages from referenced documents.

## **7.6 Evaluation Findings**

- F7.1 Section K.7 of the SAR describes confinement structures, systems, and components important to safety in sufficient detail to permit evaluation of their effectiveness.
- F7.2 The design of the NUHOMS®-61BT DSC adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the SER discusses the relevant temperature considerations.
- F7.3 The design of the NUHOMS®-61BT DSC provides redundant sealing of the confinement system closure joints using dual welds on the canister lid and closure.
- F7.4 The 61BT DSC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. No instrumentation is required to remain operational under accident conditions. Since the 61BT DSC uses an entirely welded redundant closure system, no direct monitoring of the closure is required.
- F7.5 The confinement system is leaktight for normal conditions and anticipated occurrences, thus the confinement system will reasonably maintain confinement of radioactive material. Section 10 of the SER shows that the direct dose from the NUHOMS®-61BT DSC satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).
- F7.6 The confinement system has been evaluated by analysis. Based on successful completion of specified leakage tests and examination procedures, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.
- F7.7 The staff concludes that the design of the confinement system of the NUHOMS®-61BT DSC is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the NUHOMS®-61BT DSC will allow safe storage of spent fuel. This finding considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

## **7.7 References**

- 1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.
- 2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

## **8.0 OPERATING PROCEDURES EVALUATION**

The review of the technical bases for the operating procedures is to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for key operations. The procedures for the NUHOMS<sup>®</sup>-61BT DSC, as described in Section K.8 of the SAR are very similar to those previously approved by the staff for the Standardized NUHOMS System.

### **8.1 Cask Loading**

Detailed loading procedures must be developed by each user.

The loading procedures described in the SAR include appropriate preparation and inspection provisions to be accomplished before cask loading. These include cleaning and decontaminating the transfer cask and other equipment as necessary, and performing an inspection of the NUHOMS<sup>®</sup>-61BT DSC to identify any damage that may have occurred since receipt inspection.

#### **8.1.1 Fuel Specifications**

The procedures described in Section K.8 of the SAR provide for fuel handling operations to be performed in accordance with the general licensee's 10 CFR Part 50 license and requires independent, dual verification, of each fuel assembly loaded into the NUHOMS<sup>®</sup>-61BT DSC.

Section K.2.1 of the SAR states that damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks or with cracked, bulging, or discolored cladding. The SAR further states that missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through a gap created from normal handling. However, the SAR did not fully discuss the effects of any normal or off-normal conditions that may cause the condition of the fuel to deteriorate during 10 CFR Part 72 operations. To store fuel with greater than pinhole leaks and hairline cracks the staff requires an analysis of the effects of stresses on the fuel cladding that would be associated with normal and off-normal conditions. This analysis must include a discussion of the effects of the size of the defect and the condition of the cladding (e.g., amount of oxidation).

Due to the absence of this analysis, the staff cannot determine whether a damaged fuel assembly would remain intact under normal and off-normal conditions. Since the method of containing the damaged fuel is integral to the DSC and does not include a removable failed fuel can, the staff has concluded that the NUHOMS<sup>®</sup>-61BT DSC system for containing damaged fuel does not meet handling and retrievability standards set forth in ISG-1, "Damaged Fuel," dated May 1999. Therefore, the staff denies the applicant's request to store spent fuel with damage exceeding hairline cracks and pinhole leaks in the NUHOMS<sup>®</sup>-61BT DSC.

#### **8.1.2 ALARA**

The ALARA practices utilized during operations are discussed in Section 10.5 of this SER and are found to be acceptable.

### **8.1.3 Draining, Drying, Filling and Pressurization**

Section K.8 of the SAR clearly describes draining, drying, filling and pressurization procedures for the NUHOMS®-61BT DSC that will provide reasonable assurance that an acceptable level of moisture remains in the cask and the fuel is stored in an inert atmosphere. The procedures are similar to those previously approved by the staff for the Standardized NUHOMS System.

### **8.1.4 Welding and Sealing**

Welding and sealing operations of the NUHOMS®-61BT DSC are similar to that previously approved by the staff for other DSCs used with the Standardized NUHOMS System. The procedures include monitoring for hydrogen during welding operations. Unlike previous DSCs approved for use with the Standardized NUHOMS System, as discussed in Section 7.0 of this SER, leak checks performed by TS 1.2.4a for the NUHOMS®-61BT DSC demonstrate that the inner top cover plate is “leak tight” as defined by ANSI N14.5 - 1997. Sealing operations invoke TS 1.2.5 for dye penetrant testing of the closure welds.

## **8.2 Cask Handling and Storage Operations**

All handling and transportation events applicable to moving the NUHOMS®-61BT DSC to the storage location are similar to those previously reviewed by the staff for the Standardized NUHOMS System are bounded by Section K.11 of the SAR. Monitoring operations include daily surveillances of the HSM air inlets and outlets in accordance with TS 1.3.1 and temperature performance is monitored on a daily basis in accordance with TS 1.3.2.

Occupational and public exposure estimates are evaluated in Section K.10 of the SAR. Each cask user will need to develop detailed cask handling and storage procedures that incorporate ALARA objectives of their site-specific radiation protection program.

## **8.3 Cask Unloading**

Detailed unloading procedures must be developed by each user.

Section K.8 provides unloading procedures similar to those previously approved by the staff for use with the Standardized NUHOMS System. The procedures provide a caution on reflooding the DSC to ensure that the cask’s vent pressure does not exceed 20 psig to prevent damage to the cask.

Section K.8 provides a discussion of ALARA practices that should be implemented during unloading operations, however, detailed procedures incorporating provisions to mitigate the possibility of fuel crud particulate dispersal and fission gas release must be developed by each user.

## **8.4 Evaluation Findings**

F8.1 The NUHOMS®-61BT DSC is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Section K.8 of the

applicant's SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

- F8.2 The welded cover plates of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.
- F8.3 The DSC geometry and general operating procedures facilitate decontamination. Only routine decontamination will be necessary after the cask is removed from the spent fuel pool.
- F8.4 No significant radioactive waste is generated during operations associated with the independent spent fuel storage installation (ISFSI). Contaminated water from the spent fuel pool will be governed by the 10 CFR Part 50 license conditions
- F8.5 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading will be governed by the 10 CFR Part 50 license conditions.
- F8.6 The technical bases for the general operating procedures described in the SAR are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed and evaluated on a site-specific basis.
- F8.7 Section 10 of the SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.
- F8.8 The staff concludes that the generic procedures and guidance for the operation of the NUHOMS<sup>®</sup>-61BT DSC are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices. However, the staff denies the applicant's request to store spent fuel with damage exceeding hairline cracks and pinhole leaks in the NUHOMS<sup>®</sup>-61BT DSC.

## **9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM**

### **9.1 Acceptance Tests**

The acceptance tests and inspections to be performed on the NUHOMS® -61BT DSC are discussed in detail in Sections K.3, and K.7; and further summarized in K.9 of the SAR. These inspections and tests are intended to demonstrate that the NUHOMS® -61BT DSC has been fabricated, assembled, and examined in accordance with the design criteria in Section K.2 of the SAR.

#### **9.1.1 Visual and Nondestructive Examination Inspections**

As discussed in Section 3 of this SER, the welded joints between the top inner and outer cover plates and the cylindrical shell are designed and fabricated in accordance with ASME Code Case N-595-1 by having the root and final passes of the partial penetration welds examined by penetrant testing. For normal loading conditions the stress limits will be based on NB-3200 for Level A service limits and for accident loading conditions the stress limits will be based on the Level D service limits.

#### **9.1.2 Leakage Testing**

The NUHOMS® -61BT DSC is designed to be leaktight and is tested to a leak rate of  $1 \times 10^{-7}$  atm  $\text{cm}^3/\text{sec}$ , as defined in "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5-1997<sup>1</sup>. The confinement boundary testing includes; leakage rate testing on the finished shell and the inner cover plate at the fabricator; leakage rate testing of the closure welds (inner top cover plate and vent and siphon port cover plates) after loading the spent fuel. The staff finds that this is acceptable provided that all personnel performing leak rate testing, both at the fabricator and at the loading site, are qualified in accordance with applicable standards and codes such as SNT-TC-1A.

#### **9.1.3 Poison Plate Acceptance Testing**

To assure performance of the plates' Important-to-Safety function, the critical variables that need to be verified for the environment in the cask are durability, thermal conductivity, and B10 areal density as discussed in the following paragraphs.

##### **9.1.3.1 Durability of the Poison Plates**

The application contained data and technical arguments, which were submitted to confirm the durability, for appropriate levels of radiation and temperature, of both the wrought enriched alloy and that of the three composite products. The staff agrees that the environmental conditions of radiation and temperature in the cask are not sufficiently severe, under the conditions of storage service, to damage the aluminum alloys, the aluminum matrix of composite materials, or the boron-containing particles. The staff agrees that these plate materials have been demonstrated to be capable of performing their Important-to-Safety functions within the NUHOMS® -61BT DSC for a licensing period of 20 years.

### 9.1.3.2 Thermal Conductivity Testing

The poison plate material is qualification tested to verify that the thermal conductivity equals or exceeds the values listed in Section K.4.3 for temperatures from 68°F to 650°F. Acceptance testing of the material is verified by measurements on coupons taken from production materials and by thermal tests, at one or more temperatures in the applicable range, to verify that the conductivity equals or exceeds the corresponding value in Section K.4.3. Tests are conducted by ASTM E1225<sup>2</sup>, ASTM E1461<sup>3</sup>, or equivalent methods, performed on a sample of specimens removed from coupons adjacent to the final plates, as described in SAR Section K.9.1.7. The staff agrees that these procedures are adequate to ensure thermal performance in the cask environment.

### 9.1.3.3 B10 Areal Density

There are three types of NUHOMS<sup>®</sup>-61BT DSC baskets (Type A, B, and C), each identical with the exception of the minimum B10 content in the poison plates, as described in Table K.6-1. Only one type of poison plate is used in a specific NUHOMS<sup>®</sup>-61BT DSC, based on the maximum enrichment of the fuel that will be placed in the NUHOMS<sup>®</sup>-61BT DSC. There are two sets of values of the specified minimum B10 content. These values correspond to two levels of demonstrated material effectiveness as stated in Table 6-1 of this SER. The set that applies to Boral<sup>®</sup> and Metamic<sup>®</sup> indicate larger required contents than those that apply to enriched borated aluminum alloy and Boralyn<sup>®</sup>.

These specified minimum areal density values are the amounts that must be shown (by tests) to be present in the production plates used in the NUHOMS<sup>®</sup>-61BT DSC. Each of the three specified minimum values differs from the corresponding values used in analyses of Section K.6. Full credit is not taken for the amount shown to be present. Two different levels of credit are taken for the B10 shown to be present: 90 percent and 75 percent. These levels of credit correspond to two approved levels of demonstrated effectiveness of the product: 100 percent and 83.3 percent. The level of credit is less than the demonstrated effectiveness because, in the analyses, only 90 percent of the demonstrated effectiveness is used due to uncertainties associated with the criticality calculations, i.e. neutronic effects. The uncertainty arises from the difference in the neutronic value of the poison plates in the available benchmark experiments versus that in the cask configurations. A lower neutronic value is computed to be present for a given amount of poison in the configuration of a commercial poison plate material vis-a-vis that of plates used in benchmark experiments. Thus, the levels used in the analyses (90% and 75%), are only 90 percent of the “demonstrated effectiveness” (100% and 83.3%) of the commercial materials.

Acceptance testing as discussed below will be used to ensure that the specified minimum levels are present. For each of four reference cases that correspond to a given loading of the cask (see SER Section 6.1), table K.9-1 gives both the specified minimum B10 value and the B10 value for use in analysis.

The upper level of 100 percent is granted for materials (enriched borated aluminum alloy and Boralyn<sup>®</sup>) that are regarded to have been adequately qualified for this level of credit. The effective B10 content of these materials shall be verified by neutron transmission testing of coupons taken from production materials.

All materials shall be subject to thermal conductivity, dimensional, and visual acceptance testing. The B10 areal density and uniformity of the poison plates shall be verified, using measurements taken on an area of about one centimeter in diameter, based on type, using the following approved procedures. Plate materials may be rejected due to imperfections related to their physical appearance, such as failure to meet thickness or other physical requirements. A coupons that fail to meet the required minimum may be further tested as described by procedures in the SAR. Macroscopic uniformity of B10 distribution is verified by neutron radiography or radiography of selected coupons for each material. The acceptance criterion is that there be uniform luminance across the entire coupon.

#### **9.1.3.4. Borated Aluminum Using Enriched Boron**

Acceptance testing of plate materials fabricated from the enriched borated aluminum alloy involves measurement of the effective B10 content, which is verified by neutron transmission measurements taken through the full thickness of coupons taken from the plates. Calibrated standards are used as reference materials for the measurement system. If the number of neutrons counted in a single measurement is designated as N, then N minus 3 times the square root of N must be greater than or equal to the minimum required value, which is specific to the loading condition, as listed in Table K.9.1.

A statistical analysis of data taken on coupons taken from all plates not rejected for physical (malformation & thickness) reasons is used to ensure that at least 95 percent of the lot meets the required minimum B10 content with 95 percent confidence. Accordingly, a one-sided tolerance limit corresponding to the 95% probability / 95% confidence level for the relevant sample size shall be used. Initial sampling shall be 100 percent of the coupons with reduced (50%) sampling being introduced if all coupons in the first 25% of the lot are acceptable. A rejection during reduced inspection will require a return to 100 % inspection of the lot.

The staff finds that the procedures described in the SAR for the methods used in these acceptance tests (sampling, testing, analysis) are acceptable and adequate to ensure that there is very low likelihood that regions of about 1 cm-diameter on the plates will not meet the minimum required level (of B10) specific to the loading parameters. Only 90% of the required minimum B10 content is assumed to be present in the criticality calculations given in Section K.6. Therefore, the staff agrees that borated aluminum is suitable for use at this level of credit.

The staff takes exception to the SAR's definition of a lot as defined in Section K.9.1.7.A. The staff position is that a lot is defined as all plates rolled from a single cast ingot. The analysis shall be based on a full data set for the lot. For any lot which fails the test, the plate materials of that lot shall not be used for that level of required B10 content but may be used for an alternative level of B10 for which the lot passes this test.

#### **9.1.3.5 Boralyn®**

Acceptance testing of plate from the Boralyn® metal matrix composite (MMC) involves measurement of the effective B10 content, which is verified by neutron transmission measurements taken through the full thickness of coupons taken from the plates. Acceptance testing is as described above for the enriched borated aluminum alloy, except that the acceptance criterion is taken from Table K.9-2. This table shows the same three levels of

required/specified minimum B10 content, except that in Table K.9-2 the content is expressed as a volume percent of natural boron carbide, whereas enriched weight percent boron is given in Table K.9-1.

The Boralyn<sup>®</sup> product has been previously approved for use in the TN-68 DSC and it has been qualified for durability. Extensive neutron transmission data taken on test coupon samples have been submitted to further qualify this material. On the basis of the information submitted for multiple heats of the Boralyn<sup>®</sup> product, the staff concludes that this material has been demonstrated to have sufficient uniformity as to warrant full credit when produced under the procedures described in the SAR. This material is regarded to be 100 percent effective, and the 90 percent level of credit is used in the criticality analyses for the NUHOMS<sup>®</sup>-61BT DSC. Therefore, the staff agrees that Boralyn<sup>®</sup> is suitable for use at this level of credit.

#### **9.1.3.6 Boral<sup>®</sup>**

Acceptance testing of plate from the Boral<sup>®</sup> poison plate materials involves verification of the B10 content, by chemical analysis or by neutron attenuation testing of coupons. The sampling plan verifies that the actual measured value (not some smaller number) for each acceptable coupon meets the specified minimum values of Table K.9-3. A statistical analysis of data taken on coupons taken from all plates not rejected for physical (malformation & thickness) reasons is used to ensure that at least 95 percent of the lot meets the required minimum B10 content with 95 percent confidence. Accordingly, a one-sided tolerance limit corresponding to the 95% probability / 95% confidence level for the relevant sample size shall be used. Initial sampling shall be 100 percent of the coupons with reduced (50%) sampling being introduced if all coupons in the first 25% of the lot are acceptable. A rejection during reduced inspection will require a return to 100 % inspection of the lot.

The staff finds that test methods and the procedures for data analysis are acceptable for use of this product at the 83.3 percent level of credit, which corresponds to 75 percent credit in the criticality analyses for the NUHOMS<sup>®</sup>-61BT DSC.

#### **9.1.3.7 Metamic<sup>®</sup>**

The acceptance testing of plate materials fabricated from the Metamic<sup>®</sup> metal matrix composite (MMC) involves measurements and analysis. The measurements of the effective B10 content are conducted using the same neutron absorption test methodology used for the Boralyn<sup>®</sup> metal matrix composite (MMC). The required analysis for Metamic<sup>®</sup> poison plate materials is the same as that for Boral<sup>®</sup> poison plate materials: The actual measured value is used to verify that the coupon meets the specified minimum values of Table K.9-3 and the lot must meet the specified minimum value with 95% probability at the 95% confidence level. The staff finds that these test methods and procedures for data analysis are acceptable for use of this product at the 83.3 percent level of credit, which corresponds to 75 percent credit in the criticality analyses for the NUHOMS<sup>®</sup>-61BT DSC.

## **9.2 Evaluation Findings**

The staff concluded that the proposed acceptance testing and maintenance program meet regulatory requirements. However, the staff does not accept the alternate version of Section



K.9 of the SAR as proposed by the applicant in Attachment 2, of the letter dated March 21, 2001. Other specific findings are as follows:

- F9.1 Sections K.3, K.7, and K.9 of the SAR describe the applicant's proposed program for preoperational testing and initial operations of the NUHOMS®-61BT DSC. Section K.9.2, by reference to the Standardized NUHOMS System FSAR, discusses the maintenance program.
- F9.2 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Section K.2, Tables K.2-8 and -9 of the SAR identifies the safety importance of SSCs and Section K.3 of the SAR presents the applicable standards for their design, fabrication, and testing.
- F9.3 The applicant will examine and test the NUHOMS®-61BT DSC to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section K.3, K.7, and K.9 of the SAR describe this inspection and testing.
- F9.4 Cask marking and data plate information are discussed in the Standardized NUHOMS System FSAR and were not reviewed for this amendment.
- F9.5 The staff concludes that the acceptance tests and maintenance program for the NUHOMS®-61BT DSC are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied with one exception. That exception is stated in in Section 9.1.3.4 of this SER. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

### **9.3 References**

- 1. ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials."
- 2. ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative Longitudinal Heat Flow Technique."
- 3. ASTM E1461 "Thermal Diffusivity of Solids by the Flash Method."

## **10. RADIATION PROTECTION EVALUATION**

The staff reviewed the radiation protection design features, design criteria, and the operating procedures of the NUHOMS<sup>®</sup> -61BT DSC which will be used with the Standardized NUHOMS<sup>®</sup> Horizontal Storage Module to ensure that the DSC will meet the regulatory dose requirements of 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), and 10 CFR 72.236(d). This amendment was also reviewed to determine whether the NUHOMS<sup>®</sup> 61BT fulfills the acceptance criteria listed in Section 10 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems<sup>2</sup>. The staff's conclusions are based on information provided in amendment number 3 to the NUHOMS<sup>®</sup> SAR.

### **10.1 Radiation Protection Design Criteria and Design Features**

#### **10.1.1 Design Criteria**

The radiological protection design criteria are the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. This is consistent with NRC guidance. As required by 10 CFR Part 20 and 10 CFR 72.212, each general licensee is responsible for demonstrating site-specific compliance with these requirements. The Technical Specifications also establish dose limits for the transfer cask and the Horizontal Storage Module (HSM) that are based on calculated dose rate values which are used to determine occupational and off-site exposures. The TS also establish exterior contamination limits for the DSC to keep contamination levels below 2,200 dpm/100 cm<sup>2</sup> for beta and gamma radiation, and 220 dpm/100 cm<sup>2</sup> for alpha radiation.

#### **10.1.2 Design Features**

Sections 3.3.1 and 7.1 of the Standardized NUHOMS<sup>®</sup> System FSAR, and Section K.7 of the amendment request, define the radiological protection design features which provide radiation protection to operational personnel and members of the public. The FSAR is not included in this review except for how it relates to the NUHOMS<sup>®</sup> -61BT DSC radiological protection. The radiation protection design features include the following:

- the thick-walled concrete HSM that provides radiation shielding,
- the design of the HSM air inlets paths which includes sharp bends to preclude radiation streaming,
- A recess in the HSM access opening to dock and secure the transfer cask during DSC transfer to reduce occupational exposure,
- the thick canister shield plug on both ends of the canister and transfer cask that provide occupational shielding during loading/unloading and transfer operations,
- the confinement system that consists of multiple welded barriers to prevent atmospheric release of radionuclides and is designed to maintain confinement of fuel during accident conditions,

- the system design allows for water in the DSC/transfer cask annulus which is then sealed which reduces occupational dose rates and minimizes contamination of the DSC exterior,
- the use of water in the DSC cavity (except when drained to use the crane) to reduce occupational dose rates,
- the low-maintenance design that reduces occupational exposures during ISFSI operation, and
- the implementation of ALARA principles into the cask design and operating procedures that reduce occupational exposures.

No changes were required for this review to the design features that address process instrumentation and controls, control of airborne contaminants, decontamination, radiation monitoring, auxiliary shielding devices and other ALARA considerations. Therefore these were not reviewed.

The staff evaluated the radiation protection design features and design criteria for the NUHOMS<sup>®</sup> -61BT DSC as used with the HSM and found them acceptable. The SAR analysis provides reasonable assurance that use of the NUHOMS<sup>®</sup> -61BT DSC can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Sections 5, 7, and 8 of the SER discuss staff's evaluations of the shielding features, confinement systems, and operating procedures, respectively. Section 11 of the SER discusses staff evaluations of the capability of the shielding and confinement features during off-normal and accident conditions.

## **10.2 Occupational Exposures**

Section K.8 of the amendment request discusses general operating procedures that general licensees will use for fuel loading, DSC/transfer cask operations, DSC transfer into the HSM, and fuel unloading. Table K.10-1 of the amendment request shows the estimated number of personnel, the estimated time, and the tasks involved and the estimated dose to load one canister. The estimated occupational doses are based on direct radiation calculations in Section K.5 of the amendment request, the generic operating procedures in Section K.8 of the request and on operational experience. The dose estimates indicate that the total occupational dose in loading a single canister with design basis fuel into the Horizontal Storage Modules is approximately 2.97 person-rem.

## **10.3 Public Exposures From Normal and Off-Normal Conditions**

Section K.10.2 of the amendment request presents the calculated direct radiation dose rates at distances beyond 100 meters from a sample cask array configuration loaded with design basis fuel. Figure K.10-1 depicts estimated dose rate versus distance curves. Table K.10-2 specifies distances at which the regulatory design limit of 25 mrem/yr can be achieved. An array of 20 NUHOMS<sup>®</sup> -61BT DSCs loaded with design basis fuel and placed in the Horizontal Storage Module is below regulatory limits at approximately 500 meters for two parallel but separated

1x10 arrays and at approximately 600 meters for a 2x10 array. This assumes 100% occupancy for 365 days.

The staff evaluated the public dose estimates during normal and off-normal conditions and found them acceptable. The primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions is from direct radiation (including skyshine). The canister is leaktight and the confinement function is not affected by normal or off-normal conditions therefore, no discernable leakage is credible. A discussion of the staff's evaluation and confirmatory analysis of the shielding calculations are presented in Section 5 of the SER.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each general licensee. The general licensee using the NUHOMS® -61BT DSC with the Horizontal Storage Module must perform a site-specific evaluation, as required by 10 CFR 72.212(b) to demonstrate compliance with 10 CFR 72.104(a). The actual doses to individual beyond the controlled area boundary depend on several site-specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, and use of engineered features (e.g., berm). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 72.104(a) is the responsibility of each applicant for a site license.

The general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required in 10 CFR Part 20, Subpart D by evaluations and measurements.

#### **10.4 Public Exposures From Design-Basis Accidents and Natural Phenomena Events**

Section K.11 of the amendment request summarizes the calculated dose rates for accident conditions and natural phenomena events to individuals beyond the controlled area. The confinement function of the canister is not affected by design-basis accidents or natural phenomena events thus there is no release of contents.

The amendment analysis indicates the worst case shielding consequences results in a dose at the controlled area boundary that meets the regulatory requirements of 10 CFR 72.106(b). Section 11 of the amendment request discusses corrective actions for each design-basis accident.

The staff evaluated the public dose estimates from direct radiation from accident conditions and natural phenomena events and found them acceptable. A discussion of the staff's evaluation and any confirmatory analysis of the shielding and confinement analysis is presented in Sections 5 and 7 of this SER. A discussion of the staff's evaluation of the accident conditions and recovery actions are presented in Section 11 of the SER. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents and natural phenomena will be below the regulatory limits in 10 CFR 72.106(b).

## 10.5 ALARA

Sections K.5, K.7, and K.10 of the SAR presents evidence that the 61BT DSC radiation protection design features and design criteria address ALARA requirements, consistent with 10 CFR Part 20 and Regulatory Guides 8.8 and 8.10. The overall ALARA requirements are discussed in the FSAR and were not reviewed for this amendment. Each site licensee will apply its existing site-specific ALARA policies, procedures, and practices for cask operations to ensure that personnel exposure requirements in 10 CFR Part 20 are met. Because the transfer cask may have to be drained when used with the NUHOMS<sup>®</sup> -61BT DSC and a 100-ton crane, the occupational dose rates may be higher than when loading other approved canisters. Each plant will have to consider the use of this canister with respect to their particular ALARA implementation philosophy.

The staff evaluated the ALARA assessment of the NUHOMS<sup>®</sup> -61BT DSC and found it acceptable. Section 8 of the SER discusses the staff's evaluation of the operating procedures with respect to ALARA principles and practices. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. In addition, the TS establish dose rates and surface contamination limits ensure that occupational exposures are maintained ALARA.

## 10.6 Evaluation Findings

- F10.1 The SAR amendment sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.
- F10.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.
- F10.3 The NUHOMS<sup>®</sup> -61BT DSC is designed to provide redundant sealing of the confinement system.
- F10.4 The NUHOMS<sup>®</sup> -61BT DSC is designed to facilitate decontamination to the extent practicable.
- F10.5 The SAR amendment adequately evaluates the NUHOMS<sup>®</sup> -61BT DSC and its systems important to safety to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.
- F10.6 The SAR amendment sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.
- F10.7 Operational restrictions necessary to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The NUHOMS<sup>®</sup> -61BT DSC is designed to assist in meeting these requirements.
- F10.8 The staff concludes that the design of the radiation protection system of the NUHOMS<sup>®</sup> -61BT DSC, when used with the HSM, is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the

radiation protection system design provides reasonable assurance that the NUHOMS® - 61BT DSC will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

## **10.7 References**

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.
2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.
3. U.S. Nuclear Regulatory Commission, Information Relevant to Ensuring that Occupational Radiation Exposures Will Be As Low As is Reasonably Achievable," Regulatory Guide 8.8, Revision 3, June 1978.
4. U. S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable," Regulatory Guide 8.10, Revision 1-R, May 1977.

## **11.0 ACCIDENT ANALYSIS**

The purpose of the review of the accident analyses is to evaluate the applicant's identification and analysis of hazards, as well as the summary analysis of systems responses to both off-normal and accident or design-basis events. This ensures that the applicant has conducted thorough accident analyses as reflected by the following factors:

- identified all credible accidents
- provided complete information in the SAR
- analyzed the safety performance of the cask system in each review area
- fulfilled all applicable regulatory requirements

### **11.1 Off-Normal Normal Operations**

Off-normal operations are Design Event II as defined by ANSI/ANS 57.9<sup>1</sup>. These events can be expected to occur with moderate frequency or on the order of once per year. The NUHOMS<sup>®</sup>-61BT DSC off-normal operations are described in Section K.11 of the SAR. In several instances, Section K.11 of the SAR takes credit for analyses contained in the Standardized NUHOMS System FSAR which was previously approved by the staff. Three off-normal events are analyzed for the NUHOMS<sup>®</sup>-61BT DSC: inadvertent jamming of the DSC while loading into the HSM; extreme external temperatures; and, a potential release of radionuclides to the environment. The staff reviewed these events and found them to be bounded by evaluations contained in Section K.3 of the SAR and accepted by the staff in Section 3.0 of this SER. There is no adverse impact on the NUHOMS<sup>®</sup>-61BT DSC integrity from any off-normal event.

### **11.2 Accident Events and Conditions**

Accident events and conditions are Design Event III and IV as defined in Reference 1. They include natural phenomena and human-induced low probability events. The applicant provided analyses to demonstrate design adequacy for the accident-level events discussed below. The NUHOMS<sup>®</sup>-61BT DSC postulated accidents are described in Section K.11 of the SAR. In several instances, Section K.11 of the SAR takes credit for analyses contained in the Standardized NUHOMS System FSAR which was previously approved by the staff. The staff concurs that all accident-level events and conditions have been identified and all potential safety consequences considered.

#### **11.2.1 Reduced HSM Air Inlet and Outlet Shielding**

The applicant postulated the partial loss of adjacent HSM shielding which results in an increase in the skyshine and direct doses. Table K.11-1 compares the increased dose rates as a function of distance. As can be seen from the table, the maximum dose rate is approximately  $2.4 \times 10^{-3}$  mrem/hr at 600 meters for a  $2 \times 10$  array of HSMs. The estimated recovery time for this accident is 5 days. Therefore, the estimated dose to a person at 600 meters from the ISFSI would be approximately 0.3 mrem which meets the requirements of 10 CFR Part 72.

### **11.2.2 Natural Phenomena and Human-induced Low Probability Events**

Section K.11 of the SAR considered the following accidents that could affect the structural integrity of the NUHOMS®-61BT DSC: Earthquake; Extreme Wind and Tornado Missiles; Flood; Lightning; Cask Drop; and, accidental pressurization of the DSC.

These accident conditions were evaluated in Section K.3 of the SAR and Section 8 of the FSAR for the Standardized NUHOMS System. The staff reviewed these accident conditions for the NUHOMS®-61BT DSC to the extent that they differed from those conditions previously reviewed by the staff for the Standardized NUHOMS System. In Section 3 of this SER the staff concluded that the DSC and its systems important to safety demonstrate that they will reasonably maintain confinement of radioactive material under these credible accident conditions.

### **11.2.3 Fire and Explosion**

The analysis of the hypothetical fire accident shows a maximum NUHOMS®-61BT DSC surface temperature of 499°F which is below the blocked vent case maximum temperature of 662°F. Therefore, the NUHOMS® 61BT DSC temperatures and pressures calculated for the blocked vent case bound the hypothetical fire accident case.

### **11.2.4 Loss of Neutron Shielding of Transfer Cask**

The applicant assumed that after a drop of the transfer cask, the water in the neutron shield is lost and seven damaged assemblies collect at the bottom of the canister. This accident results in an increase by a factor of three in the estimated dose rates. SAR Table K.11-4 shows that the maximum dose rate for this is approximately 1750 mrem/hr at 1 meter from the cask surface. For an 8 hour recovery time, the estimated dose rate to a member of the public at 600 meters is approximately 0.2 mrem which meets the regulatory requirements of 10 CFR Part 72.

## **11.3 Evaluation of Findings**

- F11.1 Structures, systems, and components of the NUHOMS®-61BT DSC are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.
- F11.2 The spacing of casks is discussed in Sections 1 and 4 of the Standardized NUHOMS System FSAR. The staff has previously reviewed and approved the cask spacing to ensure accessibility of the equipment and services required for emergency response.
- F11.3 The applicant has evaluated the NUHOMS®-61BT DSC to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.
- F11.4 An accident or natural phenomena event will not preclude the ready retrieval of spent fuel for further processing or disposal.



F11.5 The spent fuel will be maintained in a subcritical condition under accident conditions. Neither off-normal nor accident conditions will result in a dose, to an individual outside the controlled area, that exceeds the limits of 10 CFR 72.104(a) or 72.106(b), respectively.

F11.6 The staff concludes that the accident design criteria for the NUHOMS®-61BT DSC are in compliance with 10 CFR Part 72 and the accident design and acceptance criteria have been satisfied. The applicant's accident evaluation of the cask adequately demonstrates that it will provide for safe storage of spent fuel during credible accident situations. This finding is reached on the basis of a review that considered independent confirmatory calculations, the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

#### **11.4 References**

1. American Nuclear Society, ANSI/ANS-57.9 Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), 1992.

## **12.0 Conditions for Cask Use - Technical Specifications**

The purpose of the review of the technical specifications for the cask is to determine whether the applicant has assigned specific controls to ensure that the design basis of the cask system is maintained during loading, storage and unloading operations.

### **12.1 Conditions for Use**

The conditions for use of the NUHOMS®-61BT DSC, in concert with the Standardized NUHOMS System, are clearly defined in the Certificate of Compliance and the Technical Specification.

The staff, on its own initiative, is removing CoC Condition Nos. 9, 10, and 11. Condition Nos. 9 and 11 have been superseded by a change to 10 CFR 72.48 (see 64 FR 53582; October 4, 1999) which permits certificate holders to make certain changes to a cask design without prior Nuclear Regulatory Commission approval. Condition No. 10 has been superseded by the new 10 CFR 72.248 (same October 4, 1999, rulemaking) which requires a certificate holder to periodically update the final safety analysis report associated with the cask design. This update must include any changes to the cask design made under the provisions of 10 CFR 72.48. The change to 10 CFR 72.48 will become effective on April 5, 2001, and the addition of 10 CFR 72.248 was effective on February 1, 2000. Finally, existing Condition No. 12 will be redesignated as Condition No. 6 with no change to the text of the condition.

### **12.2 Technical Specifications**

Based on the addition of the NUHOMS® -61BT DSC to the Standardized NUHOMS® System Technical Specifications (TS) 1.2.1, 1.2.3, and 1.2.4 will be modified and TS 1.2.3a, 1.2.4a, and 1.2.17 will be added to accommodate the new DSC and fuel types that it will contain.

Table 12-1 lists the Technical Specifications for use of the NUHOMS®-61BT DSC in concert with the Standardized NUHOMS System.

### **12.3 Evaluation Findings**

F12.1 Table 12-1 of the SER lists the Technical Specifications for the use of the NUHOMS®-61BT DSC in concert with the Standardized NUHOMS System. These Technical Specifications are contained as part of the Certificate of Compliance.

F12.2 The staff concludes that the conditions for use of the NUHOMS®-61BT DSC, in concert with the Standardized NUHOMS System, identify necessary Technical Specifications to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The Technical Specifications provide reasonable assurance that the cask will provide for safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

TABLE 12-1

Standardized NUHOMS System Conditions for Operation and Technical Specifications  
Applicable for use with the NUHOMS®-61BT DSC

- 1.2 General Requirements and Conditions
  - 1.1.1 Regulatory Requirements for a General License
  - 1.1.2 Operating Procedures
  - 1.1.3 Quality Assurance
  - 1.1.4 Heavy Loads Requirements
  - 1.1.5 Training Module
  - 1.1.6 Pre-operational Testing and Training Exercise
  - 1.1.7 Special Requirements for First Cask System in Place
  - 1.1.8 Surveillance Requirements Applicability
  
- 1.2 Technical Specifications, Functional and Operating Limits
  - 1.2.1 Fuel Specifications
  - 1.2.2 DSC Vacuum Pressure During Drying
  - 1.2.3a 61BT DSC Helium Backfill Pressure
  - 1.2.4a 61 BT DSC Helium Leak Rate of Inner Seal Weld
  - 1.2.5 DSC Dye Penetrant Test of Closure Welds
  - 1.2.7 HSM Dose Rates
  - 1.2.8 HSM Air Exit Temperatures
  - 1.2.9 Transfer Cask Alignment with HSM
  - 1.2.10 DSC Handling Height Outside the Spent Fuel Pool Building
  - 1.2.11 Transfer Cask Dose Rates
  - 1.2.12 Maximum Removable Surface Contamination
  - 1.2.13 TC/DSC Lifting Heights as a Function of Low Temperature and Location
  - 1.2.14 TC/DSC Transfer Operations at High Ambient Temperatures
  - 1.2.16 Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building as a Function of Horizontal Acceleration and Loaded Cask Weight
  - 1.2.17 61 BT DSC Vacuum Drying Duration Limit
  
- 1.3 Surveillance and Monitoring
  - 1.3.1 Visual Inspection of HSM Air Inlets and Outlets
  - 1.3.2 HSM Thermal Performance

### **13.0 Quality Assurance**

The purpose of this review and evaluation is to determine whether Transnuclear West has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G. The staff has previously reviewed and accepted the Transnuclear West QA program. In addition, the staff has performed inspections of the QA program and found that it met regulatory requirements.

## CONCLUSIONS

The staff performed a detailed safety evaluation of the proposed CoC amendment request and found that the addition the NUHOMS<sup>®</sup>-61BT DSC and BWR fuel does not reduce the safety margin for the Standardized NUHOMS<sup>®</sup> System. However, as discussed in the SER, the staff does not agree that the NUHOMS<sup>®</sup>-61BT DSC may be used to store damaged fuel. The remaining areas of review addressed in NUREG 1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997 are not affected by the applicant's amendment request. Based on the statements and representations contained in the applicant's SAR and the conditions in the CoC, the staff concluded that the addition of the NUHOMS<sup>®</sup>-61BT dry shielded canister (DSC) and additional BWR fuel parameters fuel to the approved contents of the Standardized NUHOMS<sup>®</sup> System meets the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1004, Amendment No. 3,  
on \_\_\_\_\_, 2001