SAFETY EVALUATION REPORT

Docket No. 72-1029
Standardized Advanced NUHOMS® Horizontal
Modular Storage System for Irradiated Nuclear Fuel
Certificate of Compliance No. 1029
Amendment No. 1

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SUMMARY

By letter dated April 30, 2003, as supplemented on March 12, July 2, and September 14, 2004, Transnuclear, Inc. (TN) submitted a request to amend Certificate of Compliance (CoC) No. 1029. TN requested approval to add the NUHOMS® 24PT4 dry shielded canister (DSC) to the Standardized Advanced NUHOMS® System. This canister is designed to accommodate 24 intact Pressurized Water Reactor (PWR) fuel assemblies with or without integral burnable poison rods or integral fuel burnable absorber rods (IFBA), or up to 12 damaged fuel assemblies in lieu of an equal number of intact assemblies. It is designed for use with the existing Advanced NUHOMS® Horizontal Storage Module (AHSM) and transfer in the NUHOMS® OS197H transfer cask (TC), with a maximum heat load of 24kW. This safety evaluation report addresses only those changes from the previously approved design.

The application, as supplemented, included the necessary engineering analyses and proposed Safety Analysis Report (SAR) page changes. The proposed SAR revisions will be incorporated into the Final Safety Analysis Report (FSAR).

The U.S. Nuclear Regulatory Commission (NRC) 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 are based on information submitted by TN on April 30, 2003, as supplemented, requesting an amendment to add the NUHOMS® 24PT4-DSC to CoC No. 1029. The staff determined that the addition of the NUHOMS® 24PT4-DSC meets the requirements of 10 CFR Part 72.

1.0 GENERAL INFORMATION

The objective of the review of the general description of the NUHOMS® 24PT4-DSC (Ref. 1) is to ensure that TN has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.1 General Description and Operational Features

The NUHOMS® 24PT4-DSC is a new DSC design which consists of a fuel basket and a canister body, designed to hold 24 Westinghouse - CENP 16 x 16 (CE 16x16) intact or reconstituted PWR assemblies, with or without IFBAs or integral burnable poison rods, and up to 12 damaged fuel assemblies in lieu of an equal number of intact assemblies. Damaged fuel is stored in failed fuel cans. Reconstituted fuel assemblies may have up to eight (8) damaged fuel rods per assembly replaced with stainless steel rods, or any number of damaged fuel rods per assembly replaced with Zircaloy clad uranium rods. The maximum allowable heat load is 24kW. The maximum allowable burnup is 60,000 MWd/MTU. The NUHOMS® 24PT4 DSC is designed to maintain the fuel cladding temperature below allowable limits during long-term storage, short-term loading operations, off-normal and accident conditions.

The NUHOMS® 24PT4-DSC system consists of two different basket configurations. These configurations differ in the boron loading in the Boral® plates. Type A is the designation for the standard loading basket, Type B for the high loading basket. The minimum boron-10 concentration for Type A is 0.025 g/cm^2, and for Type B is 0.068 g/cm^2. Fuel to be stored in these baskets is limited to an initial ^{235}U enrichment of 4.1 wt. % for the Type A basket, and to an initial ^{235}U enrichment of 4.85 wt.% for the Type B basket.

The basket assembly provides structural support for and geometric separation of the spent fuel assemblies (SFA). The basket consists of 24 stainless steel guide sleeve assemblies, 28 carbon steel spacer discs, and four-support rod/spacer sleeve assemblies. The NUHOMS® 24PT4 shell assembly consists primarily of a cylindrical shell, the top and bottom cover plates and shield plug assemblies. Criticality is controlled by the use of fixed borated neutron absorbing material, Boral®, in the basket. Additional criticality control during storage of damaged fuel is obtained by the use of poison rodlets (boron carbide (B_4C) encased in stainless steel tubes). The confinement vessel for the NUHOMS® 24PT4-DSC consists of the shell, the inner cover plates of the top and bottom shield plug assemblies, the vent and siphon block, the vent and siphon cover plates, and the associated welds. The NUHOMS® 24PT4 is designed to be leaktight.

The NUHOMS® 24PT4-DSC will be stored in the Advanced NUHOMS® Horizontal Storage Module (AHSM), and transferred in a OS197H Transfer Cask (TC) with a radial liquid neutron shield. Those components were reevaluated during this safety evaluation to the extent that they were compatible with the NUHOMS® 24PT4-DSC, and with the higher maximum heat load of 24kW.

The applicant updated several sections of the Standardized Advanced NUHOMS® System's FSAR (Ref. 2) to document compliance with the specifications in Interim Staff Guidance (ISG) - 18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as

Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation."

1.2 Drawings

Section A.1 of the SAR contains the non-proprietary drawings for the NUHOMS® 24PT4-DSC, including drawings of the structures, systems and components (SSCs) 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® 24PT4-DSC. Specific SSC are evaluated in other sections of this SER.

1.3 DCSS Contents

The NUHOMS® 24PT4-DSC system is designed to store up to 24 intact or reconstituted CE 16x16 PWR fuel assemblies with or without IFBAs or integral burnable poison rods. The NUHOMS® 24PT4-DSC is also designed for storage of up to 12 damaged fuel assemblies in specially designed failed fuel cans with the balance being loaded with intact fuel. Reconstituted assemblies containing up to eight (8) replacement stainless steel rods (per assembly) in place of damaged fuel rods, or any number of replacement Zircaloy clad uranium rods (per assembly) in place of damaged fuel rods, are acceptable for storage as either intact or damaged assemblies. Each NUHOMS® 24PT4-DSC is designed for a maximum heat load of 24 kW/canister and 1.26 kW per fuel assembly. Maximum initial enrichment is 4.85 wt. % ^{235}U, and maximum allowable burnup is 60,000 MWd/MTU. Fuel specifications are detailed in Section 2.2 of the Technical Specifications (TS).

1.4 Qualification of the Applicant

Appendix A, Section A.1.3 of the SAR contains reference to the applicant's qualifications which has not changed from the previously approved FSAR.

1.5 Evaluation of Findings

F1.1 A general description of the NUHOMS® 24PT4-DSC is presented in Appendix A, Section A.1 of the SAR.

F1.2 Drawings for the SSC important to safety are presented in Appendix A, Section A.1 of the SAR.

F1.3 Specifications for the spent fuel to be stored in the NUHOMS® 24PT4-DSC are provided in the SAR Appendix A, Section A.1.2.3, and TS 2.2.

F1.4 The technical qualifications of the applicant are identified in Appendix A, Section A.1.3 of the SAR, which are unchanged from the FSAR.

F1.5 The quality assurance program was previously approved for the Standardized Advanced NUHOMS® System, and is referenced in Section 13 of the SAR.

F1.6 The NUHOMS® 24PT4-DSC has not been certified under 10 CFR Part 71 for use in transportation.

F1.7 The staff concludes that the information presented in this section of the SAR satisfies the requirements for the general description under 10 CFR Part 72.

1.6 References

1. Amendment 1 to NUHOMS® Certificate of Compliance No. 1029, for Dry Spent Fuel Storage Casks, Revision 0, April 30, 2003, as supplemented.

2. Transnuclear, Inc, Final Safety Analysis Report (FSAR) for the Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 0, February 2003, USNRC Docket Number 72-1029. .

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 (Ref. 1).

2.1 Structures, Systems, and Components Important to Safety

The SSCs important to safety are summarized in Appendix A, Table A.2.5-1. Only those features that were not previously approved by the staff for the Standardized Advanced NUHOMS® System are addressed in the table.

2.2 Design Basis for SSCs Important to Safety

The NUHOMS® 24PT4-DSC design criteria summary includes the range of spent fuel types and configurations to be stored, and design criteria for environmental conditions and natural phenomena.

2.2.1 Spent Fuel Specifications

The NUHOMS® 24PT4-DSC system is designed to store up to 24 intact CE 16x16 PWR fuel assemblies with or without IFBAs or integral burnable poison rods. The NUHOMS® 24PT4 is also designed for storage of up to 12 damaged fuel assemblies in specially designed failed fuel cans with the balance being loaded with intact fuel. Reconstituted assemblies containing up to eight replacement stainless steel rods in place of damaged fuel rods, or any amount of replacement Zircaloy clad uranium rods, are acceptable for storage as either intact or damaged assemblies.

The NUHOMS® 24PT4-DSC system consists of two different basket configurations. These configurations differ in the boron loading in the Boral® plates. Type A is the designation for the standard loading basket, Type B for the high loading basket. Minimum Boron-10 concentration for Type A is 0.025 g/cm^2, and for Type B it is 0.068 g/cm^2. Fuel to be stored in the Type A basket is limited to an initial ^{235}U enrichment of 4.1 wt. %, and is limited to 4.85 wt.% ^{235}U for Type B. Additional basket configurations for storage of damaged fuel are discussed in Table A.2.1-4 of the SAR.

2.2.2 External Conditions

Section A.2.2 of the SAR identifies the bounding site environmental conditions and natural phenomena for which the NUHOMS® 24PT4-DSC is analyzed. In cases where these did not change from the previously approved FSAR, no descriptions were given. External conditions are further evaluated in Sections 3 through 12 of this SER.

2.3 Design Criteria for Safety Protection Systems

A summary of the design criteria for the safety protection systems of the NUHOMS® 24PT4-DSC, is presented in Section A.2.3 of the SAR. Details of the design are provided in Sections A.3 though A.11 of the SAR.

The NUHOMS® 24PT4-DSC design is reviewed for storage of spent fuel for 20 years. The Standardized Advanced NUHOMS® System is licensed for 20 years of storage. The fuel cladding integrity is assured by the NUHOMS® 24PT4-DSC and basket design which limits fuel cladding temperatures and maintains a nonoxidizing environment in the cask cavity. The NUHOMS® 24PT4-DSC is designed to maintain a subcritical configuration during loading, handling, storage, and accident conditions. Criticality is controlled by utilizing the fixed neutron absorbing material, Boral® , in the NUHOMS® 24PT4-DSC basket. When more than four damaged assemblies are stored in the NUHOMS® 24PT4-DSC, depending on the maximum fuel enrichment, poison rodlets (B_4C encased in steel tubes) are required in the inner 12 intact assemblies for criticality control. The NUHOMS® 24PT4-DSC shell assembly is designed, fabricated, examined and tested in accordance with the requirements of Section III, Subsection NB of the ASME Code. The internal basket assembly such as the spacer discs and guide sleeve assemblies are designed to the criteria of ASME Code, Section III, Subsection NG. The support rods and spacer sleeves are designed to the criteria of Section III, Subsection NF of the ASME Code. Alternatives to the Code are noted in Section A.3.1.2.3 of the SAR.

2.4 Evaluation Findings

F2.1 The staff concludes that the principal design criteria for the NUHOMS® 24PT4-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, Interim Staff Guidance (ISG), and accepted engineering practices. A more detailed evaluation of design criteria and an assessment of the compliance with those criteria is presented in Section 3 through 12 of the SER.

2.5 Reference

U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” Title 10, Part 72.

3.0 STRUCTURAL EVALUATION

This section presents the results of the structural design review of the amendment request for the addition of NUHOMS® 24PT4-DSC to the Standardized Advanced NUHOMS® Horizontal Modular Storage System (AHSM). The NUHOMS® 24PT4-DSC is designed to accommodate intact and/or damaged CE 16x16 fuel assemblies, with an assembly average burn-up of up to 60,000 MWd/MTU, a maximum enrichment level of 4.85 wt. % ^{235}U and a maximum decay heat load of 24 kW per DSC. The applicant stated that no change is required to the previously licensed AHSM or TC design to accommodate the new canister. The AHSM structural analyses presented in Chapter 3 of the FSAR for the Standardized Advanced NUHOMS® System (Ref. 1) with the 24PT1-DSC are also applicable to the system with the 24PT4-DSC because the analyses are based on a conservative DSC weight of 85,000 lb. which bounds the weight of the 24PT4-DSC. However, the stress evaluations for the support steel and the heat shield were revised based on the increased temperature for the 24PT4-DSC. On-site transfer of a loaded 24PT4-DSC is performed utilizing the NUHOMS® OS197H transfer cask (TC) described in the Standardized NUHOMS® System FSAR (Ref. 2). The FSAR analyses of the TC envelop the 24PT4-DSC configuration. Thus, no new analysis is warranted.

The review was conducted against the appropriate regulations as described in 10 CFR 72.236. The structural evaluation shows that the 24PT4-DSC design is compatible with requirements of 10 CFR 72.236 for maintaining the spent fuel in a subcritical condition, providing adequate radiation shielding and confinement, having adequate passive heat removal capability, providing a redundant sealing of the confinement system, and providing wet or dry transfer capability.

3.1 Structural Design of the Dry Storage Canister 24PT4-DSC

The 24PT4-DSC canister assembly is made of several steel structural components and the related weld filler metal, all of which are important to safety. The individual structural components are as follows: for the canister shell assembly - a cylindrical shell, the top and bottom cover plates, and shield plug assemblies; for the internal basket assembly - 24 stainless steel guidesleeve assemblies, 28 carbon steel spacer discs, and four-support rod/spacer sleeve assemblies. The 24PT4-DSC shell assembly's top and bottom ends include stainless steel forgings and stainless steel plates that encase the lead (ASTM B29) shield plugs. The cylindrical shell is fabricated from SA240, Type 316 stainless steel. ASME SA-533 Grade B Class 1 carbon steel material is used for fabrication of the 24PT4-DSC basket assembly spacer discs. This is different from the SA-537, Class 2 used for the 24PT1-DSC spacer discs in order to accommodate the higher disc temperatures. ASME Code Case N-499-1 (Ref. 3) provides the basis for limited elevated temperature service up to 1000^{E}F for the SA-533 Grade B Class 1 carbon steel and the material properties are shown in SAR Table A.3.3-2. The support rod and spacer sleeves are fabricated of SA-479, Type XM-19 stainless steel. Damaged fuel assemblies are stored in Failed Fuel Cans. Failed Fuel Cans are provided with screens at the top and bottom to contain fuel debris and allow water to fill or drain from the can. The Failed Fuel Can and the guide sleeves are all fabricated from SA-240, Type 304 stainless steel.

The 24PT4-DSC shell assembly is designed, fabricated, examined and tested in accordance with the requirements of Section III, Subsection NB of the ASME Code. The internal basket assembly, such as the spacer discs and guide sleeve assemblies, are designed to the criteria of

ASME Code, Section III, Subsection NG. The support rods and spacer sleeves are designed to the criteria of Section III, Subsection NF of the ASME Code.

3.2 Materials

The applicant provided a general description of the materials of construction in SAR Sections A.1.2, A.3.1, and A.3.3. Additional information regarding the materials, fabrication details and testing programs can be found in SAR Section A.9.1. The staff reviewed the information contained in these sections; Table A.3.1-5, ASME Code Alternatives and the information presented in the SAR drawings, to determine whether the NUHOMS 24PT4-DSC meets the requirements of 72.236(g) and (h). In particular, the following aspects were reviewed: materials selection; brittle fracture; applicable codes and standards; weld design and specifications; and chemical and galvanic corrosion.

3.2.1 Structural Materials

Most of the structural components of the 24PT4-DSC (e.g., shell, bottom plate, and top plate) are fabricated from austenitic stainless steel (i.e., Type 316). The fuel compartment tubes in the 24PT4-DSC basket are also fabricated from austenitic stainless steel. This type of steel was selected because of its high strength, ductility, resistance to corrosion and metallurgical stability. Since there is no ductile-to-brittle transition temperature in the range of temperatures expected to be encountered by this steel, its susceptibility to brittle fracture is negligible. The top shield plug is fabricated from austenitic stainless steel that encases the lead shield plug. SA-533, grade B, Class 1 carbon steel is used to fabricate the basket assembly spacer discs. An electroless nickel plating is applied to the carbon steel spacer discs. The staff concludes that the selection of these materials meets the requirements of the ASME Boiler & Pressure Vessel Code. Therefore, these materials are acceptable for use in the 24PT4-DSC.

3.2.2 Nonstructural Material

The basket assembly structure consists of welded stainless steel guidesleeve assemblies that make up the fuel compartments. Each fuel compartment accommodates neutron absorber material plates for criticality control. Boral® neutron absorber plates are used for criticality control when storing intact fuel assemblies. For storage of damaged fuel, Boral® plus boron carbide encapsulated in stainless steel tubes is used for criticality control. In accordance with Section A.9.1, appropriate acceptance testing will be used to ensure that the neutron absorbers have the minimum specified ^{10}B loading.

The staff concludes that the neutron absorbers will be adequately durable during the service life of the cask. The acceptance and qualification for the neutron absorbers are discussed in Chapter 9 of this SER.

3.2.3 Welds

The DSC cylindrical shell is assembled using full penetration longitudinal welded joints and circumferential welded joints at the junction between the inner bottom plate and the shell. These welds are performed in accordance with ASME Code, Section III, Subsection NB. The

DSC top closure welds are performed in accordance with ASME Code Case N-595-1 and the criteria in ISG-15.

The DSC materials of construction (e.g., stainless steel, carbon steel, etc.) are readily weldable using common available welding techniques. The use of an experienced fabricator will ensure that the process chosen for fabrication will yield a durable canister. The DSC welds were well-characterized on the SAR drawings, and standard welding symbols and notations in accordance with American Welding Society (AWS) Standard A2.4, “Standard Symbols for Welding, Brazing, and Nondestructive Examination” were used.

The staff concludes that the welded joints of the DSC meets the requirements of the ASME Code. Although the DSC closure welds are partial penetration welds, this configuration will perform its intended structural and confinement functions.

3.2.4 Bolting Materials

The 24PT4-DSC is an all-welded canister.

3.2.5 Coatings

No zinc, zinc compounds, or zinc-based coatings are used on the carbon steel spacer discs of the 24PT4-DSC basket assembly. The carbon steel spacer discs will be coated with an electroless nickel plating that has been analyzed for use in the 24PT1-DSC. The coating will protect the steel from excessive oxidation of the surface.

3.2.6 Mechanical Properties

Tables A.3.3-1 through A.3.3-3 of the SAR provide material property data for the major materials including: stainless and carbon steels and aluminum alloys. Most of the values were obtained from ASME Code, Section II, Part D. The staff independently verified the temperature dependent values for the yield and ultimate stresses, modulus of elasticity, and coefficient of thermal expansion. The staff concludes that these material properties are acceptable and appropriate for the expected load conditions (e.g., static or dynamic, impact loading, hot or cold temperature, wet or dry conditions) during the storage period.

3.2.7 Chemical and Galvanic Reactions

In Section A.3.4.1 of the SAR, the applicant evaluated whether chemical, galvanic or other reactions among the materials and environment would occur. The staff reviewed the design drawings and applicable sections of the SAR to evaluate the effects, if any, of intimate contact between various materials in the DSC system materials of construction during all phases of operation. In particular, the staff evaluated whether these contacts could initiate a significant chemical or galvanic reaction that could result in components corrosion or combustible gas generation. Pursuant to NRC Bulletin 96-04, a review of the DSC system, its contents and operating environments has been performed to confirm that no operation (e.g., short term loading/unloading or long-term storage) will produce adverse chemical or galvanic reactions. The 24PT4-DSC is primarily made of stainless steel. The staff concludes that in this dry, inert environment, the 24PT4-DSC components are not expected to react with one another or with the cover gas. Further, oxidation or corrosion, of the fuel (i.e., cladding) and the DSC internal components will effectively be eliminated during storage.

To ensure that the safety hazards associated with the ignition of hydrogen gas are mitigated, the procedures in SAR Section A.8.1 are employed to monitor the concentration of hydrogen gas during any welding or cutting operations. The staff concludes that these procedures are adequate to prevent ignition of any hydrogen gas that may be generated during welding operation. Further, the potential reaction of aluminum with the spent fuel pool water will not impact the ability of the neutron absorbers and its aluminum grid plates to perform their intended function because the loss of aluminum metal is negligible.

3.3 Normal and Off-Normal Conditions

3.3.1 Analysis Methods

The 24PT4-DSC shell assembly is analyzed using three finite element models and the ANSYS software package. The three finite element models are as follows: (1) an axisymmetric model of the 24PT4-DSC shell assembly, (2) three-dimensional top-end model with the top shield plug assembly, and (3) three-dimensional bottom-end model. The axisymmetric model is a complete model of the 24PT4-DSC, which includes both top and bottom shield plug assemblies, cover plates, and the cylindrical shell. The model is used to analyze axisymmetric loads such as vertical dead weight, top/bottom end drop loads, and internal/external pressure loads. The three-dimensional top and bottom end models are half-symmetric (i.e., 180^{E} representations) and are used to analyze non-axisymmetric loads such as thermal loads, side drop loads, and grapple pull/push loads.

The stress analyses of the spacer discs are performed using 3-D finite element models developed using the ANSYS computer program. A half-symmetry (180^{E}) and a full-symmetry (360^{E}) finite element model are used for analyzing in-plane loads, and a quarter-symmetry (90E) model is used for analyzing out-of-plane loads. The support rod assemblies, including the support rods, spacer sleeves and support rod to spacer sleeve mechanical connections, are analyzed using the criteria of Subsection NF and Appendix F of the ASME Code, Section III for linear component supports. Stress analyses of the guidesleeve assemblies, which consist of guidesleeve tubes, oversleeves, and shim plates, are performed using a combination of closed-form calculations and finite element analyses using an ANSYS model of the guidesleeve. Elastic analyses are used for normal and off-normal conditions, and elastic-plastic analyses are used for the postulated side drop accident load case.

3.3.2 Loading Conditions Analyzed

The normal and off-normal operating load conditions analyzed for the 24PT4-DSC shell assembly included the dead weight, internal/external pressure, thermal, operational handling loads that include normal transfer as well as off-normal transfer loads. The loading combinations are shown in Table A.3.6-1 of the SAR. These loading cases include the non-operational load cases for fabrication, and leak testing as well as operational loads such as fuel loading/unloading, draining and drying, transfer operations and storage.

The loading conditions analyzed for the basket assembly of the 24PT4-DSC are similar to the shell assembly except that for the basket assembly, there is no significant effect from the pressure loads.

3.3.3 Analysis Results

The 24PT4-DSC shell assembly has been shown to meet the appropriate material stresses allowable for the service levels defined in the ASME Code, Section III, Division 1, Subsection NB, for Class 1 Components. The calculated maximum stresses for the various components are summarized and then compared to the allowable stresses in Table A.3.6-4 of the SAR. The SAR uses the stress ratio between the calculated stress and the allowable stress to show compliance to the stress criteria of the ASME Code. Therefore, the stress ratio must be less than 1.0 for all loadings and components. The worst case (i.e., the largest stress ratio) under normal and off-normal conditions is for the outer bottom cover plate under primary membrane at a stress ratio of 0.99. The worst case for the canister shell is under primary plus secondary stresses at a stress ratio of 0.97.

For the 24PT4-DSC basket assembly components the analysis results are summarized in the SAR in Table A.3.6-7 through Table A3.6-9. The largest stress ratio for the spacer discs under normal and off-normal condition loads is only 0.45. Similarly, the maximum stress ratio for the guidesleeve is small. It is seen that the most critical component for the basket assembly is the support rods. The maximum stress ratio for the support rods is 0.90 under the 60g end drop condition. End drops are not postulated for on-site operation of the horizontal NUHOMS® System and the stress ratio is less then 1.0.

3.4 Accident Conditions

3.4.1 Analysis Methods

The same finite element models are used for evaluating the accident conditions with the only difference being the loading conditions imposed. The 24PT4-DSC shell assembly is analyzed for an accident pressure loading of 100 psig (Table A.3.1-4 of SAR). Drop loads are applied as static loads corresponding to the postulated drop decelerations for the 24PT4-DSC positioned inside the TC. A 75g side drop and a 25g corner drop (at 30^{o} from horizontal) are postulated. The 25g corner drop is not performed because it is considered to be bounded by the 75g horizontal drop and the 60g end drop condition evaluated under 10 CFR Part 71. The analysis methods include static and dynamic analyses utilizing elastic and elastic-plastic methods, as well as classical closed form solutions. Under accident conditions when stresses exceed the material elastic range, the elastic-plastic analysis method is used. The elastic-plastic analysis method permits plastic deformation that reflects more closely the structural behavior of the 24PT4-DSC shell assembly when stresses exceed the material yield stress level.

Stress analyses of the guide sleeve assemblies are performed using a combination of closed-form calculations and finite element analyses using the same ANSYS finite element model of the guidesleeve used in the normal and off-normal loading analyses. Elastic-plastic analyses are used for the postulated side drop accident load case. The spacer discs under accident conditions are analyzed using the same three ANSYS finite element models used under normal and off-normal loading cases. For the accident loading cases such as the side drop analysis, an elastic-plastic stress analysis is performed using the in-plane (e.g., half-symmetry or full-symmetry) model. The technique used a plastic modulus of 5% of the elastic modulus. These analyses are based on a spacer disc tributary weight of 2431.5 lbs. Three drop orientations are considered: 0^{E}, 18.5^{E} (e.g., directly on the cask rail), and 45^{E} from the azimuth. In addition, an eigenvalue bucking analysis is performed to demonstrate the stability of the spacer discs under in-plane loading. The analysis uses the full-symmetry (360^{E}) in-plane ANSYS model as shown in Figure A.3.6-5 of the SAR. The support rod assemblies, including the support rods, spacer sleeves and support rod to spacer sleeve mechanical connections, are analyzed using the criteria of ASME Code, Subsection NF and Appendix F for linear component supports. For loads along the axis of the 24PT4-DSC, the load distributions in the support rod assemblies are evaluated using a simple ANSYS beam model. The model includes the support rods and spacer sleeves with the moment and axial force from each spacer disc applied to the assembly.

3.4.2 Loading Cases Analyzed

The following postulated accident conditions and extreme natural phenomena loading cases have been addressed:

a. Earthquake
b. Tornado wind pressure and tornado missiles
c. Flood
d. Fire and explosion
e. Accidental drop of the 24PT4-DSC inside the Transfer Cask
f. Lightning
g. Blockage of air inlet and outlet openings
h. Accidental pressurization of the 24PT4-DSC
I. Burial
j. Inadvertent loading of a newly discharged fuel assembly

Table A.3.6-1 of the SAR provides the accident and natural phenomena loading combinations for the 24PT4-DSC canisters. These loading combinations are identified by the Service Levels C or D in accordance with the ASME Code.

3.4.3 Analysis Results

Section A.3.6 of the SAR presents the structural analyses of the 24PT4-DSC for normal, off-normal, accident and natural phenomena loading condition. It should be noted that there is no change to the structural analysis of the AHSM presented in Chapter 3 of the FSAR for the Advanced NUHOMS® System. Thus, no new structural analysis is performed for the AHSM.

Tables A.3.6-5 and A.3.6-6 of the SAR provide the summary results for the enveloping loading cases for the accident load conditions (i.e., Service Levels C and D loadings) for the 24PT4-DSC shell assembly. The calculated maximum stress intensities for the various components of the shell assembly under accident loading condition are provided in Table A.3.6-3. Tables A.3.6-5 and A.3.6-6 list the stress type, the controlling load combination, the calculated and the allowable stress intensities. The stress ratios for the controlling load combinations are provided for each component. For those loading cases governed by the Code Service Level C, the most highly stressed component has a stress ratio of 0.81 for the outer top cover plate under membrane plus bending. The highest stress ratio is 0.97 for the outer bottom cover plate for primary membrane stress. The highest stress ratio for the cylindrical shell is 0.80 for membrane plus bending stresses. For those loading cases governed by Service Level D, the most highly stressed component is the DSC shell at a stress ratio of 0.96 for primary membrane stress under postulated side drop loading condition. The inner top cover plate stress ratio is 0.98 for membrane plus bending stresses under the blockage of AHSM air inlet and outlet openings loading combination. All other stress ratios for Service Level D were less than 0.78 as shown in Table A.3.6-6.

For the 24PT4-DSC basket assembly the analysis results for accident conditions are provided in Tables A.3.6-7, A.3.6-8 and A.3.6-9 of the SAR. Allowable stress intensities for the basket components are based on ASME Code, Section III, Subsection NG or Subsection NF, as appropriate. For the spacer disc the maximum stress ratio is 0.96 for membrane plus bending stress condition for a 18.5E or a 45E azimuth drop. For vacuum drying and transfer to/from the Independent Spent Fuel Storage Installation (ISFSI) loading conditions, the primary plus secondary stress intensity exceeds the $3S_m$ stress limit criteria. Thus, the provisions of NG-3228.3 for simplified elastic-plastic analysis must be used to qualify these stresses.

The accidental load producing the high stress condition for the guidesleeve assembly is the end or the side drop. The highest stress ratio is 0.6 for end drop as shown in Table A.3.6-8 of the SAR. The support rod pretension will be overcome by the compression of sleeves and the tensile stress in the support rods becomes compression. The maximum compressive stress calculated was 2.5 ksi. This is well below the allowable stress, as rods are laterally supported by the spacer sleeves with no possibility of buckling under the 60g end drop condition.

3.5 Evaluation Findings

F3.1 The 24PT4-DSC is described in sufficient detail to enable an evaluation of its structural effectiveness and is designed to accommodate the combined loads of normal, off-normal, accident and natural phenomena events.

F3.2 The 24PT4-DSC 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 to the DSC that will prevent retrieval of the stored spent nuclear fuel.

F3.3 The 24PT4-DSC is designed and fabricated so that the spent 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 24PT4-DSC is evaluated to demonstrate that it has a redundant seal and that it will maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The SAR describes the materials that are used for structures, systems, and components (SSCs) important to safety and the suitability of those materials for their intended functions in sufficient detail to facilitate evaluation of their effectiveness.

F3.6 The design of the 24PT4-DSC and the selection of its materials adequately protects the spent fuel cladding against degradation that might otherwise lead to gross rupture.

F3.7 The 24PT4-DSC employs noncombustible materials, which will help maintain safety control functions.

F3.8 The materials that comprise the 24PT4-DSC will maintain their mechanical properties during all conditions of operation.

F3.9 The 24PT4-DSC design employs materials that are compatible with wet and dry spent fuel loading and unloading operations and facilities. These materials are not expected to degrade over time, or react with one another, during any conditions of storage.

F3.10 The staff concludes that the structural design of the 24PT4-DSC 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 addition of a 24PT4-DSC in the Standardized Advanced NUHOMS® System will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable industry codes and standards, accepted practice and confirmatory analysis.

3.6 References

2. Transnuclear, Inc, Final Safety Analysis Report (FSAR) for the Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 0, February 2003, USNRC Docket Number 72-1029.

3. Transnuclear, Inc, Final Safety Analysis Report (FSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 8, June 2004, USNRC Docket Number 72-1004.

4. American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, Division 1, Code Case N-499-1, Use of SA-533 Grade B, Class 1 Plate and SA-508 Class 3 Forgings and their Weldments, for Limited Elevated Temperature Service Section III, Div. 1.

4.0 THERMAL EVALUATION

The applicant is seeking approval for use of the 24PT4-DSC in the Standardized Advanced NUHOMS® System, in addition to the previously approved 24PT1-DSC. Both of these canisters may store up to 24 PWR spent fuel assemblies, utilize the OS-197H Transfer Cask (TC), which is used in transfer operations for the Dry Shielded Canister (DSC), and the Advanced Horizontal Storage Module (AHSM), a concrete storage module that houses the DSC in a horizontal attitude for long-term storage, and has been designed for the storage of spent fuel in areas with high seismic activity. For this review, the abbreviation “the system” will be used for the 24PT4-DSC in the AHSM or the TC.

The objective of the thermal review is to ensure that the cask/storage module components and fuel material temperatures of the system will remain within the allowable values for normal, off-normal and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding will be maintained throughout the transfer and storage periods to protect the cladding against degradation which could lead to gross rupture. This review also confirms that the thermal designs of the DSC, TC and the AHSM have been evaluated using acceptable analytical methods.

4.1 Spent Fuel

The system is designed to store light water reactor PWR fuel. The system accommodates up to 24 intact and/or reconstituted fuel assemblies, with Zircaloy or ZIRLO™ cladding and UO_2 or $(U,Er)O_2$ or $(U,Gd)O_2$ fuel pellets. The maximum heat load for the DSC is 24 kW, depending on the load configuration. There are also heat load limits per assembly.

4.1.1 Spent Fuel Cladding

The staff verified that the cladding temperatures for each fuel type proposed for storage are below the temperature limits which would preclude cladding damage that could lead to gross rupture.

The staff reviewed the discussion on material temperature limits with respect to the applicable requirements of 10 CFR Part 72, which require the spent fuel cladding to be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

For the PWR fuel assemblies, the allowable temperature limits are based on Interim Staff Guidance No. 11, Rev. 3 (ISG-11) (U.S. Nuclear Regulatory Commission, November 2003). For normal conditions (long-term) of storage and short-term fuel loading and storage operations (which includes drying, backfilling with inert gas, and transfer of the cask to the storage pad), the temperature limit of the fuel cladding is maintained below 400EC. This is done to ensure that circumferential hydrides in the cladding will not dissolve and go into solution during fuel loading operations, and that re-precipitation of radial hydrides does not occur in the cladding during storage (see ISG-11, Rev. 3 for a discussion on hydride re-orientation). The applicant established a maximum fuel cladding temperature limit of 570EC (1058EF) for off-normal thermal transients and accident conditions, and invoked the thermal cycling criteria contained in ISG-11. Fuel with burnup greater than 60 GWd/MTU is unacceptable for storage in the 24PT4-DSC system.

4.2 Cask System Thermal Design

4.2.1 Design Criteria

The design criteria for the system have been formulated by the applicant to assure that public health and safety will be protected during dry cask spent fuel storage. These design criteria cover the normal storage conditions for the 20-year approval period and postulated off-normal and accident conditions.

Section A.4.1 of the SAR defines several primary thermal design criteria for the system:

1. Pressures within the 24PT4-DSC cavity are within design values considered for structural and confinement analyses.

2. Maximum and minimum temperatures of the confinement structural components must not adversely affect the confinement function.

3. The allowable cladding temperatures that are applicable for normal, off-normal and accident conditions of storage are taken directly from ISG-11, Rev. 3.

4. Thermal stresses for the 24PT4-DSC, when combined with other loads, will be maintained at acceptable levels to ensure confinement integrity of the system.

The staff finds that the primary thermal design criteria have been sufficiently defined.

4.2.2 Design Features

To provide adequate heat removal capability, the applicant designed the system with the following features:

1. The 24PT4-DSC is cooled by buoyancy driven air flow through openings at the base of the AHSM (the module), which allows ambient air to be drawn into the module to cool the DSC. Heated air exits through vents in the top of the shield block, creating a stack effect.

2. The 24PT4-DSC contains spacer disks, support rods and guide sleeve assemblies. Heat transfer through the basket structure is achieved by conduction through the spacer disk plates and guide sleeve assemblies as well as radiation from these components and convection over the component surfaces.

3. The DSC cavity is backfilled with helium gas to aid removal of heat from the fuel assemblies and maintain an inert atmosphere.

4. A metal heat shield is placed around a major portion of the DSC to shield the AHSM concrete surfaces above and to the side of the DSC from thermal radiation effects.

The staff verified that all methods of heat transfer internal and external to the system are passive. The SAR drawings and summary of material properties provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the system.

4.3 Thermal Load Specifications

The applicant selected the bounding heat load (24 kW) and two fuel configurations as input to the thermal models. The staff has reviewed this selection and has reasonable assurance that these loads are bounding.

4.3.1 Storage Conditions

The applicant provided thermal analyses using a robust computational fluid dynamics (CFD) code to demonstrate that the system would perform satisfactorily for normal, off normal and accident conditions.

Table 4-1 provides the temperature and insolation conditions that the applicant applied in the thermal analysis.

Table 4-1
Standardized Advanced NUHOMS® System
Ambient Temperatures and Insolation Considerations

Condition	Temperature (°F)	Solar Insolation (Btu/hr-ft^2)
Normal	0 to 104	0 to 123
Off-Normal	-40 to 117	0 to 123
Accident	-40 to 117	0 to 123

4.3.2 Normal Conditions

The normal conditions of storage for the system are described in Section A.4.4 of the SAR. The normal storage conditions consider a maximum daily temperature of 104EF (40EC) and includes solar insolation of 123 Btu/hr-ft^2, which is the value recommended in 10 CFR Part 71 for solar insolation on a flat surface averaged over a 24-hour period.

The minimum normal storage condition considers a 0EF (-17.8EC) average daily temperature and assumes no solar insolation. The staff concludes that the applicant's approach of using maximum and minimum daily average temperatures and insolation for the system is acceptable because AHSM temperature response to changes in the ambient conditions will be slow due to the large thermal inertia of the AHSM. Maximum and minimum daily temperatures are included in TS Section 4.4.3 as siting parameters that must be evaluated by the storage system user.

4.3.3 Off-Normal Conditions

Off-normal conditions for the system are also described in Section A.4.4 of the SAR. Included in these conditions is a maximum temperature of 117EF (47.2EC) and a minimum temperature of -40EF (-40EC). Also included is a solar insolation of 123 BTU/hr-ft^2 which is applied to the AHSM roof surface.

4.3.4 Accident Conditions

4.3.4.1 Blocked Vents

Several accident conditions are postulated by the applicant, and are described in Section A.4.6 of the SAR. The first accident evaluated by the applicant for the system is a complete blockage of the AHSM ventilation inlet and outlet openings. The AHSM and the DSC are evaluated for the ambient temperatures and insolation values outlined in Table 4-1 above for the accident condition.

4.3.4.2 Transfer Cask Loss of Neutron Shield and Sunshade

The second accident condition postulated by the applicant is a loss of water neutron shield in the annular region of the OS-197H transfer cask, as well as a loss of the required sunshade during transfer operations at the extreme off-normal ambient temperature condition of 117EF (47.3EC). This accident is assumed to reach steady state temperature conditions. The applicant states that this accident is bounded by the blocked vent accident condition described above.

The applicant referred to a previous transfer cask accident analysis (Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 8, June 2004, NRC Docket No. 72-1004). The staff reviewed this analysis and accepted it for this application.

4.3.4.3 Fire

The third accident condition postulated by the applicant is a fire (SAR Section A.4.6.4) that occurs during transfer of the DSC to the AHSM. This analysis refers to the analysis for the original application for this system (FSAR Section 4.6.4).

A 15 minute fire with an average flame temperature of 1475EF (800EC), an average convective heat transfer coefficient of 5.21E-4 Btu/min-in^2-EF, and radiative heat transfer as recommended in 10 CFR 71.73 is hypothesized. This is postulated to be caused by the spillage and ignition of 300 gallons of combustible transporter fuel. The assumed 15-minute duration for the transient evaluation is conservative based on a calculated fire duration of 14 minutes for this amount of fuel, at a consumption rate of 0.15 in/min. In the applicant's analysis, the fuel is considered to take the form of a "pool" under the transfer cask measuring 201.5 inches in diameter with a fuel depth of 2.17 inches. The staff finds that this is a reasonable assumption.

Following the fire, the transfer cask is subjected to the prevailing maximum off-normal ambient conditions and a loss of the water neutron shield from the transfer cask is postulated. The analysis is continued to determine peak temperatures of cask components. The applicant states that the results of the fire accident analysis are bounded by the blocked vent accident condition described in Section 4.3.4.1 above. Based on review, the staff concludes that the thermal loads for the fire accident are acceptable and that the blocked vent accident does bound this accident.

4.3.4.4 Flood

The fourth accident condition analyzed by the applicant is the submersion of the AHSM in water due to a worst-case flood accident. The applicant states that this provides a favorable thermal environment for the AHSM and DSC, as heat will be removed from the DSC more efficiently by water than by air. The staff agrees with this statement. This accident is therefore bounded by all other accident conditions.

4.3.4.5 Cask Heatup During Loading

The applicant's description of the effects of loading and unloading conditions on the system is provided in Section A.4.7 of the SAR. Three bounding loading conditions were evaluated by the applicant, including heatup of the DSC prior to blowdown, an analysis of heatup of the DSC during vacuum drying, and a steady state analysis of the canister after helium backfill. The applicant also analyzed the unloading condition of a reflood of the DSC.

For heatup of water in the DSC prior to blowdown, the DSC is evaluated for an initial DSC temperature of 140EF (60EC), and fuel building ambient temperature of 120EF (49EC). DSC heat loads from 12 kW to 24 kW were analyzed. The heatup analyses assumed no axial conduction and neglected radiation within the DSC.

For heatup of the DSC during vacuum drying, the DSC has been drained of water and filled with helium or air. The air model used a three dimensional (3D) slice of the canister and the helium model used the 3D computational fluid dynamics (CFD) model developed for storage. The applicant completed both steady-state and transient analyses. An initial DSC shell temperature of 230EF (110EC), and a maximum allowable DSC heat load of 24 kW were assumed for the steady state analysis. The transient analysis sets the DSC fuel basket temperature to the saturation temperature of water as an initial condition, and is performed for a heat load of 24 kW.

The applicant also analyzed the effect of reflooding on the DSC during unloading operations. Limits placed on the flow rate of water into the DSC during this evolution will minimize thermal shock and prevent pressurization of the DSC to greater than the design pressure of 20 psig.

The staff reviewed the analyses conducted for DSC heatup, and found the performance of the DSC under the conditions described above and in Section A.4.7 of the SAR to be acceptable.

4.4 Model Specification

4.4.1 Configuration

The applicant developed thermal models of the AHSM and the 24PT4- DSC using a robust 3D CFD code (FLUENT). These models were able to capture the thermal response of all of the major components of the AHSM and DSC, including natural convection within the system. The applicant also benchmarked FLUENT against the NUHOMS-7P test data for the DSC surface and horizontal storage module, as described in SAR Section A.4.10.

The 24PT4-DSC in the transfer cask was modeled using the ANSYS finite element code for air blowdown and FLUENT for helium blowdown.

4.4.1.1 AHSM Model

The AHSM model is described in SAR section A.4.4.2. The model represents the entire AHSM and DSC shell. The analysis for the AHSM is performed for a loaded DSC located in the interior of a multiple module array with a DSC present in two adjacent AHSMs. The DSC internals are not modeled. Instead, a uniform heat flux is applied to the shell surface. The top and front surfaces of the AHSM are exposed to prevailing ambient conditions, and the side and back surfaces are modeled as adiabatic to simulate adjacent modules.

4.4.1.2 24PT4- DSC Basket Section/Fuel Assembly Model

This model is described in SAR Section A.4.4.4. A worst case, three dimensional slice of the 24PT4-DSC basket assembly and fuel cross sections is modeled in detail. Two spacer disks (truncated at the mid-plane) are included in the model to account for radial conduction through the spacer disks. Axial heat transfer is neglected by setting the ends of the model to adiabatic conditions. The outer surface of the DSC is set to a specified temperature distribution determined from the AHSM model. Each fuel region within the DSC is modeled as a solid with an effective thermal conductivity (described in SAR Section A.4.9).

4.4.1.3 24PT4-DSC in Transfer Cask Model

This model is described in SAR Section A.4.4.3. The TC model developed by the applicant to simulate the DSC within the TC is a two-dimensional axisymmetric model which includes the DSC shell assembly, and the DSC cavity modeled as a homogenous region. The model is based on an analysis conducted and submitted in the Standardized NUHOMS® System FSAR (Ref. 1). The maximum DSC shell temperature is extracted from this model and used in the DSC basket analysis. The DSC 3D slice model described above was also used for the TC/DSC analyses.

4.4.2 Material Properties

The material properties used in the thermal analysis of the storage cask system are listed in SAR Section A.4.2. The applicant provided a summary of the material compositions and thermal properties for all components used in the system. The material properties given reflect the accepted values of the thermal properties of the materials specified for the construction of the storage system. All material properties provided were within the operating temperature ranges of the storage system components. For homogenized materials such as the fuel assemblies, the applicant described the source from which the effective thermal properties were derived.

4.4.3 Boundary Conditions

Boundary conditions were applied to the models described above to analyze the behavior of the systems under normal, off-normal, and accident conditions. The applicant analyzed the model of the DSC in the transfer cask and in the AHSM to obtain maximum shell temperatures for the DSC under all conditions. The maximum shell temperatures were then used in the DSC basket/fuel assembly model to determine a maximum fuel cladding temperature for each set of conditions. Ambient temperature and insolation values were tabulated for all analyzed conditions.

4.4.3.1 Accident Conditions - Blocked Vent

For the postulated blocked vent accident conditions, the HEATING7 AHSM model described in SAR Section 4.4.2.2 for the storage condition was used, and the inlet and outlet vents were blocked. The boundary conditions include the DSC off-normal condition temperature distribution before the postulated accident, and ambient temperatures and insolation as outlined in Table 4-1 (above).

4.4.3.2 Accident Conditions- Loss of Neutron Shield and Sunshade for Transfer Cask

The applicant referred to a previous transfer cask accident analysis. The staff reviewed this analysis and accepted it for this application (see Section 4.3.4.2 above and SAR Section A.4.6.3).

4.4.3.3 Accident Conditions-Fire

The postulated fire accident conditions and the model of the DSC in the transfer cask are described in SAR Section A.4.6.4. The boundary conditions for the fire accident are described in Section 4.3.4.3 above. The boundary conditions include the DSC and transfer cask normal condition temperature distribution before the postulated fire and the maximum off-normal ambient conditions after the fire.

4.4.3.4 Cask Heatup Analysis

The cask heatup analysis is described in SAR Section A.4.7.3. The model does not credit any heat transfer in the axial direction and any radiation within the cavity. In addition, the starting time to reach boiling begins following the placement of the first fuel assembly with an assumed heat load of 24 kW for the entire duration of the heatup.

The applicant stated that to assure that known conditions exist at the start of blowdown and the initial fuel clad temperature assumed is conservative, the DSC will be filled with water, if needed, to assure that the initial cladding temperature is bounded by the 230EF annulus temperature (as stated in the procedures in SAR Section A.8).

4.5 Thermal Analysis

4.5.1 Temperature Calculations

4.5.1.1 Storage Conditions

The system has been analyzed to determine the temperature distribution under long-term storage conditions that envelop normal, off-normal, and accident conditions. The DSC basket is considered to be loaded at design-basis maximum heat loads with PWR assemblies. The AHSMs are considered to be arranged in an ISFSI array and subjected to design-basis ambient conditions with insolation. The maximum predicted and allowable temperatures of the components important to safety are discussed in Section A.4.1 of the SAR. Low temperature conditions were also considered. The calculated fuel clad temperatures for Zircaloy-clad fuel assemblies are listed in SAR Tables A. 4.1-1, A.4.1-2 and A.4.1-3, for Normal, Off-Normal, and Accident Conditions, respectively. The applicant's analysis of the fuel cladding temperatures for the maximum heat load of 24 kW showed that the fuel cladding temperatures remain below their respective acceptable temperature limits. Table 4-2 below summarizes the temperatures of key components in the cask for various environmental conditions.

4.5.1.2 Accident Conditions- Blocked Vents

Initially, the applicant postulated that if there was a complete blockage of the inlet and outlet vents of AHSM, it would be cleared by plant site personnel within 40 hours. Therefore, the analyzed event lasts 40 hours. However, the applicant revised the calculations for an event that lasts 25 hours. In addition, the Technical Specifications (TS) were modified to ensure that the module temperature monitoring frequency was acceptable.

Table 4-2
Temperatures of Key Components in the Advanced NUHOMS® Storage System[1]

Component	Normal Storage Conditions		Transfer Condition (off normal)	Normal Allowable Range (EF)	Accident Conditions	
	Maximum (EF)	Minimum[2] (EF)	Maximum (EF)		Maximum (EF)	Allowable Range (EF)
AHSM Concrete	**232**	**0**	**N/A**	**0 to 300**	**392[(3)]**	**-40 to 300**
AHSM Support Steel	**281**	**0**	**N/A**	**0 to 800**	**615**	**-40 to 800**
AHSM Heat Shield	**314**	**0**	**N/A**	**0 to 800**	**542**	**-40 to 800**
DSC Shell	**459**	**0**	**443**	**0 to 800**	**642**	**-40 to 800**
Guidesleeve or Can	**662**	**0**	**675**	**0 to 800**	**845**	**-40 to 900[4]**
DSC Oversleeve	**662**	**0**	**675**	**0 to 800**	**845**	**-40 to 900[4]**
DSC Spacer Disk	**653**	**0**	**668**	**0 to 700**	**836**	**-40 to 1000**
DSC Support Rod/Spacer Sleeve	**560**	**0**	**580**	**0 to 800**	**738**	**-40 to 800[4]**
DSC Boral® Sheet	**662**	**0**	**675**	**0 to 850**	**845**	**-40 to 1000**
Zircaloy Cladding	**697**	**0**	**712**	**0 to 752**	**880**	**-40 to 1058**

Notes:
1. Temperatures are based on 24 kW heat load
2. Assuming no credit for decay heat and a daily average ambient temperature of 0EF
3. Applicant will conduct testing on concrete samples to demonstrate acceptable concrete performance
4. See SAR Table A.3.1-6 for code alternative for maximum allowable temperatures

Extreme ambient conditions were used as boundary conditions for this analysis. The analysis included a heat source of 24 kW for qualification of the AHSM concrete and the 24PT4-DSC. Maximum DSC shell temperature and concrete temperatures were obtained in this analysis. None of the components of the DSC exceeded their temperature limits. The maximum concrete temperature reported was above the limit specified by the applicant. The applicant has committed to testing the concrete used to fabricate the AHSM at an elevated temperature to demonstrate that the concrete will perform satisfactorily. The results for this accident analysis are summarized in Table 4-2 above. Based on this analysis and the TS, the staff finds reasonable assurance that the fuel cladding integrity and the confinement boundary will not be compromised during the blocked vent transient.

4.5.1.3 Accident Conditions- Loss of Neutron Shield and Sunshade for Transfer Cask

The applicant referred to a previous transfer cask accident analysis (Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 8, June 2004, NRC Docket No. 72-1004) (Ref.1).

The applicant analyzed an accident involving the loss of water from the annular neutron shield region of the transfer cask and loss of the required sunshade during the transfer of the

24PT4-DSC to the AHSM. The scenario was run to steady-state temperature conditions. The temperatures reported by the applicant were below all material limits, and this analysis was bounded by the blocked vent transient described above. The staff reviewed this analysis and accepted it for this application.

4.5.1.4 Accident Conditions - Fire

The applicant analyzed a fire accident for the DSC in the transfer cask using the methodology presented in FSAR Section 4.6.4. The initial temperatures for the fire analysis are based on the maximum transfer conditions. The peak temperatures of the key DSC components due to a 15-minute fire with a 24 kW decay heat are enveloped by the blocked vent accident described above. All of the fire accident temperatures were below the short-term design-basis temperatures for the DSC and the transfer cask. Based on these analyses, the staff has reasonable assurance that the cladding integrity and the confinement boundary will not be compromised during the fire or post-fire transient.

4.5.1.5 Cask Heatup Analyses

The applicant utilized the DSC basket/fuel assembly HEATING7 model to determine the time (as a function of heat load) for water in the DSC cavity to boil prior to blowdown and backfilling of the DSC with helium while in the TC. The results are documented in SAR Section A.4.7.3.

ANSYS and CFD DSC models were used for the thermal analysis of the vacuum drying process (see SAR Section A.4.7.1). Calculations were completed for air and helium gas mediums.

The staff reviewed these calculations and found reasonable assurance that the temperature of the components of the DSC will remain within acceptable values.

Both the time-to-boil, and the vacuum drying time limit information presented in the SAR should be used by the end-user of the system to ensure that DSC component temperatures are not exceeded during cask loading evolutions.

4.5.2 Pressure Analysis

4.5.2.1 Storage/Off Normal/Accident Conditions

In SAR Section A.4.4.8, the applicant evaluated internal pressurization for normal conditions. The applicant assumed a fully loaded DSC. A 1% failure of fuel rods and control components is assumed. For the ruptured rods, a 100% release of the rod fill gas and a 30% release of the fission product gasses is postulated. Using the calculated temperatures for the basket and fuel cladding, the applicant used the ideal gas law to calculate the pressure. The applicant calculated a normal condition pressure of 17.5 psig, which is below the applicant's criteria of 20 psig for normal conditions.

In the same section, the applicant evaluated internal pressure of the DSC for off-normal conditions. The off-normal pressure calculation included a 10% failure of fuel rods and control components is assumed. For the ruptured rods, a 100% release of the rod fill gas and a 30%

release of the fission product gasses is postulated. The maximum off-normal pressure calculated by the applicant was 22.3 psig.

In SAR Section A.4.6.6, the applicant evaluated internal pressure of the DSC for accident conditions. The accident pressure calculation included a 100% failure of fuel rods and control components. For the ruptured rods, a 100% release of the rod fill gas and a 30% release of the fission product gasses is postulated. The maximum accident pressure calculated by the applicant was 80.7 psig. The applicant reported the results of the pressure analysis and acceptance criteria in SAR Table A.4.4-10.

The staff reviewed the applicant’s calculations and determined that the applicant’s calculations used appropriate methods and cover gas temperatures determined in SAR Section A.4. The highest predicted pressure was 80.7 psig at a cavity gas temperature of 713EF for the accident condition, which is below the DSC thermal criteria pressure of 81 psig. (The pressure used in the stress analysis was 100 psig as stated in Table A.3.1-4.)

Based on review of the applicant’s pressure analysis, the staff found reasonable assurance that the internal cask pressures remain below the cask design pressure rating under normal, off-normal, design-basis natural phenomena, and design-basis accident conditions or events.

4.5.2.2 Pressure During Unloading of Cask

Pressurization of the DSC is discussed in Section A.4.7.2 of the SAR. The DSC is vented during reflood, and therefore a rapid pressure build-up is not a concern. The procedure for reflood assures that the flow rate of water into the relatively hot DSC is controlled to avoid exceeding the 20 psig design pressure for this condition.

4.5.2.3 Pressure During Loading of Cask

The applicant discusses pressurization of the DSC during loading in Section A.3.1.2.1.3.2 of the SAR. The applicant states that the normal pressure limit during DSC blowdown is 20 psig, which is well below the maximum design pressure limits of the DSC.

4.5.3 Confirmatory Analyses

The staff, with the assistance of Pacific Northwest National Laboratorty (PNNL), performed detailed confirmatory analyses (see Sections 4.5.3.1 and 4.5.3.2, below) of the dry cask storage system.

The applicant used different modeling approaches for the thermal analyses (a robust computational fluid dynamics (CFD) program (FLUENT) to model the DSC and the AHSM temperatures and flow patterns), and validated the CFD program against experimental data (NUHOMS® 7P). The staff reviewed the analyses and found reasonable assurance that the analyses accurately depicted the system performance.

4.5.3.1 Analysis of 24PT4-DSC

A confirmatory analysis of the performance of the 24PT4-DSC for a heat load of 24 kW was conducted by PNNL using the COBRA-SFS finite difference thermal-hydraulics code. The COBRA-SFS code has been validated against data gathered from spent fuel assemblies stored at the Idaho National Engineering and Environmental Laboratory (INEEL). The COBRA-SFS code utilizes detailed fuel assembly models to accurately predict maximum fuel cladding temperatures for different heat loads.

PNNL also completed a confirmatory analysis of the 24PT4-DSC using a robust CFD program (StarCD, also validated against the NUHOMS 7P data). The results of the applicant and the staff were similar.

4.5.3.2 Analysis of AHSM

A confirmatory analysis of the performance of the AHSM for a DSC heat load of 24 kW was conducted by PNNL using the StarCD CFD program. The results of the applicant and the staff were similar.

4.5.4 Conclusion

The staff approves the applicant's request for storage of fuel as stated in Section 4.1 above, within the limits described in the Technical Specifications.

4.6 Evaluation Findings

F4.1 The staff finds that the thermal SSCs important to safety are described in sufficient detail in Sections A.1 and A.4 of the SAR to enable an evaluation of their effectiveness. Based on the applicant's analyses, there is reasonable assurance that the system is designed with a heat removal capability consistent with its importance to safety. The staff also finds that there is reasonable assurance that analyses of the systems demonstrate that the applicable design and acceptance criteria have been satisfied for the storage of the authorized fuel assemblies.

F4.2 The staff has reasonable assurance that the temperatures of the cask SSCs important to safety will remain within their operating temperature ranges and that cask pressures under normal and accident conditions were determined correctly.

F4.3 The staff has reasonable assurance that the system provides adequate heat removal capacity without active cooling systems.

F4.4 The staff has reasonable assurance that the spent fuel cladding will be protected against degradation that leads to gross ruptures by maintaining the clad temperature below maximum allowable limits and by providing an inert environment in the cask cavity.

F4.5 The staff finds that the thermal design of the system 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 system will allow safe storage of spent fuel for a certified life of 20 years. 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.

4.7 Reference

Transnuclear, Inc, Final Safety Analysis Report (FSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 8, June 2004, USNRC Docket Number 72-1004.

5.0 SHIELDING EVALUATION

The staff evaluated the capability of the Advanced NUHOMS® system with the 24PT4-DSC to provide adequate protection against direct radiation. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20 (Ref.2) and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d). Because 10 CFR Part 72 (Ref. 1) 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 Advanced NUHOMS® 24PT4-DSC fulfills the acceptance criteria listed in Section 5 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems." (Ref. 3)

The applicant performed two primary sets of shielding analysis in Chapter A.5 of the SAR. In the first set of analyses, the applicant performed shielding calculations with MCNP, using design-basis source term generated by SAS2H/ORIGEN-S. In the second set of analyses, the applicant used a dose response function methodology (i.e., source strength to dose rate conversion factors) to determine burnup, cooling time, and enrichment parameters for the fuel qualification tables. The fuel parameters calculated with the dose response methodology result in dose rates and individual heat loads that are less than, or equal to, the maximum dose and thermal limits derived from the bounding calculations in the first set of analyses. The applicant used the ANISN shielding code to calculate dose response functions and SAS2H/ORIGEN-S to generate the source terms. Each subsection of this SER section addresses both sets of analyses, as appropriate.

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The 24PT4-DSC, when used with the Advanced NUHOMS® System provides both gamma and neutron shielding during loading/unloading, transfer, and storage operations. The applicant stated that the physical design of the 24PT4-DSC is similar to the 24PT1-DSC, with the following primary differences: 10" longer DSC, use of lead shield plug assemblies instead of carbon steel shield plugs, 28 instead of 26 spacer discs with different material, and use of failed fuel cans not requiring guidesleeves.

The applicant did not identify any physical changes to the advanced horizontal storage module (AHSM) or the OS197H transfer cask (TC) in the amendment, as currently described in the Advanced NUHOMS® System FSAR.

The staff reviewed the new AHSM design configuration with respect to its shielding performance during normal conditions of operation for the new 24PT4-DSC canister specified in Appendix A of the SAR.

The staff evaluated the Advanced NUHOMS® 24PT4-DSC shielding design features and found them acceptable. The applicant's analysis provides reasonable assurance that the shielding design of the Advanced NUHOMS® 24PT4-DSC, when used with the Standardized Advanced

NUHOMS® System, meets the regulatory requirements of 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

5.1.2 Shielding and Source Term Design Criteria

The overall radiological protection design criteria are the regulatory dose requirements in 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d). The applicant analyzed the Advanced NUHOMS® 24PT4 with spent fuel as described in Section A.5.2. Appendix A of the SAR also provided methodology for determining dose rate limit criteria used to determine acceptable burnup, cooling time and enrichment parameters. These include maximum dose rate limits of 877.0 mrem/hr on the side surface of the TC and 0.079 mrem/hr on the roof surface of the AHSM. Average and peak dose rates on the front, side, top and back of the AHSM and TC are provided in Table A.5.1-2. These ANISN dose rates correlate to MCNP dose rates of 1231 mrem/hr and 0.21 mrem/hr at the same location with design-basis source term. In addition heat load limits for each configuration serve as additional source term design criteria, which in turn limit overall dose rates. Based on these design criteria, the applicant calculated maximum dose rates on the rear end of the Top Shield Block Assembly (TSBA), the back of the rear shield wall, front entrance of the bottom air inlet, the AHSM roof, and the side.

The staff reviewed the design criteria and found it to be acceptable. The shielding design criteria defined in the SAR provides reasonable assurance that the 24PT4-DSC/AHSM system can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72.

5.1.3 Loading Criteria

The applicant requested the addition of a new storage canister, the Advanced NUHOMS® 24PT4-DSC for use with the Advanced NUHOMS® Horizontal Modular Storage System which includes the AHSM and the TC. There were no changes to the AHSM, which is described in the Standardized Advanced NUHOMS® System FSAR. Therefore, the AHSM and TC are not reviewed here except as how they relate to the Advanced NUHOMS® 24PT4-DSC. The NUHOMS® 24PT4-DSC will be used to store intact and damaged PWR fuel assemblies with the specifications described in Tables A.2.1-1 - A.2.1-12. The 24PT4-DSC may store PWR fuel assemblies in one of three alternate heat zoning configurations with a maximum decay heat of 1.26 kW per assembly and a maximum heat load of 24 kW per canister, as shown in Figures A.2.1-1 through A.2.1-3. Figure A.2.1-4 shows the location for failed fuel cans inside the 24PT4-DSC.

Fuel Qualification Tables A.2.1-5 through A.2.1-8 define the minimum required cooling time based on fuel assembly burnup and initial fuel enrichment for the assembly, assuming that no reconstituted fuel assembly with stainless steel rods is present. The fuel qualification tables to be used for reconstituted assemblies with stainless steel rods present are provided in Tables A.2.1-9 through Table A.2.1-12. Table A.2.1-4 shows the six basic configurations for storing damaged fuel.

5.2 Source Specifications

The applicant calculated design basis source term to perform bounding shielding calculations with MCNP. The design basis source specifications are presented in Section A.5.2 of the SAR.

The gamma and neutron source terms were calculated with the SAS2H/ORIGEN-S modules of the SCALE 4.4 computer code using the 44-group ENDF/B-V cross section library. The design basis fuel type is the CE 16x16 assembly, as described in Section A.5.2.

The design basis source term used a heavy metal weight of 0.4555 MTU to produce the highest source strength at a given burnup, cooling time and enrichment combination. The 24PT4-DSC is limited to changes in burnup, cooling time, and minimum initial enrichment specifications. It also allows for high-burnup fuel up to 60,000 MWd/MTU. 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 0.2, 0.2, and 0.1, respectively (listed in Table A.5.2-3).

A design basis bounding source term was determined using 24 CE 16x16 fuel assemblies with a burnup of 45 GWD/MTU, an initial enrichment of 3.8 wt.% ^{235}U and 5 years cooling loaded in the DSC. The bounding gamma and neutron source term were then combined in the shielding models to calculate dose rates as discussed in Section A.5.2.3 in the amendment. Other burnup, initial enrichment, and cooling time combinations (Tables A.2.1-5 through A.2.1-8 for intact fuel and Tables A.2.1-9 through A.2.1-12 for reconstituted fuel) account for differences in the total neutron, (n, γ) reaction, and primary gamma contributions from the fuel assembly using the response functions given in Table A.5.2-7. An example calculation is provided in Table A.5.2-9. The applicant determined each cooling time in the fuel qualification tables by estimating the dose rate from each source term using the response function and ensuring that this dose rate is below the dose rate calculated for the design basis source term.

5.2.1 Gamma source

Gamma source terms are calculated for design basis fuel for various zones and are listed in Table A.5.2-5. The source term is calculated using a CE 16x16 fuel assembly with a burnup of 45 GWD/MTU, an initial enrichment of 3.8 wt. % ^{235}U and 5 years cooling loaded in the DSC. These source terms are used in the MCNP analysis by multiplying the assembly sources by the number of assemblies (24). The applicant determined this to be the bounding analysis in Section A.5.2.3 of the SAR.

5.2.2 Neutron Source

Neutron source terms are calculated for design basis fuel and are listed in Table A.5.2-6. The source term is calculated using a CE 16x16 fuel assembly with a burnup of 45 GWD/MTU, an initial enrichment of 3.8 wt. % ^{235}U and 5 years cooling loaded in the DSC. The applicant calculated the neutron source term for use in the shielding models by multiplying the individual assembly sources by the number of assemblies (24). The applicant determined this to be the bounding analysis in Section A.5.2.3 of the SAR.

5.2.3 Staff Evaluation

The staff notes that the use of an ANISN 1-D model to represent the three-dimensional 24PT4-DSC shielding system results in additional uncertainties. However, the use of ANISN in the shielding analysis is essentially limited to evaluating the relative changes in dose rates versus relative changes in source term for alternate combinations of burnup, cooling time, and enrichment. The staff finds the use of ANISN acceptable.

The staff reviewed the source term analyses in Chapter A.5 of the SAR. The staff has reasonable assurance that the design basis gamma and neutron source term for the Advanced NUHOMS® 24PT4-DSC based on 24 design basis assemblies, each with a decay heat of 1.26 kW/assembly for a total canister heat load of 30.1 kW are acceptable for the shielding analysis. The analyzed configuration used for the shielding calculations is conservative because the design basis bounding configuration would exceed the total heat load limit for the DSC. Based on the calculations provided by the applicant, the staff agrees that the 24PT4-DSC loaded with 24 design basis assemblies results in bounding dose rates over the heat load Configurations 1, 2, and 3. The staff notes that Configuration 1, 2, and 3 fuel parameters are not specifically restricted by calculated dose limits, and changes to decay heat source terms are not always directly proportional to radiation source terms. However, the staff has reasonable assurance that the respective heat load limits will result in lower doses than the design basis.

The staff performed confirmatory calculations of the source term for the specified fuel type, burnup conditions, and cooling times. The staff used the SAS2H/ORIGEN-S computer codes. The calculated source terms were in general agreement with the applicant's bounding source term used in the MCNP analysis.

5.3 Shielding Model Specifications

The Advanced NUHOMS® 24PT4-DSC system shielding and source configuration is described in Sections A.5.3 and A.5.4 of the SAR. The shielding models consist of three-dimensional representations of the AHSM and vent streaming paths, including the spent nuclear fuel source, the 24PT4-DSC canister and the OS197H transfer cask. These models are depicted in Figures A.5.4-1 through A.5.4-10 of the SAR.

As discussed in Section A.10.2 of the SAR, the applicant used MCNP to calculate off-site dose rates at large distances from one generic ISFSI array. The generic array consists of a 2x10 back-to-back array of AHSMs loaded with 24 design basis fuel assemblies in the 24PT4-DSC (see also Section 10.4 of this SER). The applicant provided a sample MCNP input file in Section A.5.5.4 of the SAR.

5.3.1 Shielding and Source Configuration

The shielding source is divided into four axial regions: bottom nozzle, in-core, plenum, and top nozzle. The lengths of these regions are specified in Table A.5.2.1. The source regions were homogenized and cross-sectional area was preserved.

5.3.2 Material Properties

The composition and material densities used in the MCNP models are specified in Tables A.5.2-1 and A.5.2-2. The composition and material densities used in the ANISN models are identical to those used in the MCNP analysis and are listed in Table A.5.3-1.

5.3.3 Staff Evaluation

The staff reviewed the shielding models and found them acceptable. The material compositions and densities used were appropriate and provide reasonable assurance that the 24PT4-DSC was adequately modeled. In addition, the methodologies used are similar to those used to support the previous Advanced NUHOMS® storage application.

5.4 Shielding Evaluation

The applicant presented dose rates for normal conditions and accident conditions in Section A.5.1 and A.11 of the SAR. The shielding analysis used ANSI/ANS Standard 6.1.1-1977 flux-to-dose rate conversion factors to calculate dose rates as shown in Table A.5.3-3. The applicant used MCNP, a three-dimensional Monte Carlo transport code, to determine dose rates at the top AHSM vent areas, surface of the AHSM and the surface, 1-ft. and 3-ft. from the surface of the TC. The applicant provided a sample MCNP input file in Section A.5.5.4 of the SAR.

5.4.1 Normal Conditions

The applicant presented design basis dose rates at various locations surrounding the AHSM and TC in Tables A.5.1-2 through A.5.1-4.

5.4.2 Occupational Exposures

The applicant determined occupational exposures in Chapter A.10 of the SAR. The exposures were based on estimations from surrounding dose rates calculated in Chapter A.5 and the operating procedures referenced in Chapter A.8. The staff's evaluation of the occupational exposures is discussed in Section 10 of the SER.

5.4.3 Off-site Dose Calculations

The applicant estimated offsite dose rates from the 2x10 back-to-back array in Section A.10.2 of the SAR. The offsite dose rates were based on 24 design basis fuel assemblies and heat load Configuration 1 analyzed in Chapter A.5 of the SAR. Tables A.10.2-1 through A.10.2-8 present the calculated offsite doses for this array at distances of 6 to 600 meters based on full-time occupancy. The off-site dose calculations are further evaluated in Section 10.4 of this SER.

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 by 10 CFR Part 20, Subpart D, by evaluations and measurements.

5.4.4 Accident Conditions

Chapter A.11 of the SAR does not identify an accident that significantly degrades the shielding of the AHSM.

5.4.5 Staff Evaluation

Section 10 of this SER evaluates the overall dose (i.e., direct radiation and hypothetical radionuclide release) from the Advanced NUHOMS® 24PT4-DSC system. The staff reviewed the dose calculations for normal operations and found them acceptable. The staff has reasonable assurance that compliance with 10 CFR Part 20 and 10 CFR 72.104(a) from direct radiation can be achieved by general licensees. The actual doses to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, and atmospheric conditions. In addition, 10 CFR 72.104(a) includes doses from other fuel cycle activities such as reactor operations. Each general licensee is responsible for verifying compliance with 10 CFR 72.104(a), in accordance with 10 CFR 72.212. 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 and workers, as required by evaluation and measurements.

The staff reviewed the accident dose analysis and found it acceptable for the specific design and contents requested in this amendment of the SAR. The staff has reasonable assurance that the direct radiation from the Advanced NUHOMS® 24PT4-DSC system satisfies 10 CFR 72.106(b) at or beyond a controlled boundary of 100-meters from the design basis accidents.

The staff notes that the off-site accident dose rate calculations may be less accurate than the dose rate calculations in the vicinity of the 24PT4-DSC/AHSM system, and that precise exposure times cannot be predicted. However, the staff notes that direct radiation is relatively easy to mitigate within a reasonable amount of time and that the exposure times are based on realistic assumptions.

The staff performed confirmatory calculations using SAS4. The staff based its evaluation on the design features and model specifications presented in the drawings shown in the SAR Appendix A. The staff's calculated dose rates were in reasonable agreement with the SAR values. Any differences are due to staff's conservative modeling of the Advanced NUHOMS® system using the 24PT4-DSC. The staff found that the SAR has adequately demonstrated that the Advanced NUHOMS® 24PT4-DSC is designed to meet the criteria of 10 CFR 72.104(a) and 72.106.

5.5 Evaluation Findings

F5.1 Section A.5 of the SAR sufficiently describes shielding design features and design criteria for the structures, systems, and components important to safety.

F5.2 Radiation shielding features of the Advanced NUHOMS® 24PT4-DSC system are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F5.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 72.106 are the responsibility of each general licensee. The Advanced NUHOMS® 24PT4-DSC shielding features (as approved by NRC) are designed to assist in meeting these requirements.

F5.4 The staff concludes that the design of the radiation protection system of the Advanced NUHOMS® 24PT4-DSC, when used with the AHSM, 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 Advanced NUHOMS® 24PT4-DSC system 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 applicants' 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, NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.

6.0 CRITICALITY EVALUATION

The staff reviewed Amendment 1 to the Standardized Advanced NUHOMS® 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 the regulatory requirements of 72.236(c), and 72.236(g) (Ref.1) are met. The staff also reviewed the SAR to determine whether the cask system was consistent with the following acceptance criteria listed in Section 6 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" (Ref. 2):

3. 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.

4. 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.

5. 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.

6. Criticality safety of the cask system should not rely on the use of the following credits:

   a. burnup of the fuel,
   b. fuel-related burnable neutron absorbers, or
   c. more than 75% for fixed neutron absorbers when subject to standard acceptance tests.

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 system under normal, off-normal, and accident conditions.

The Standardized Advanced NUHOMS® System design features relied on to prevent criticality are the fuel basket's geometry and permanent neutron-absorbing Boral® panels. Some fuel configurations additionally rely on the insertion of neutron-absorbing B_4C poison rodlets for criticality control. The Boral® panels and poison rodlets maintain subcriticality when the canister is flooded with water during loading and unloading. The dry shielded canister (DSC) design evaluated under this amendment for use with the Standardized Advanced NUHOMS® System is the 24PT4, which is a tube-and-disk design similar to the previously approved 24PT1-DSC.

The fuel assemblies are placed in baskets with square fuel cells and Boral® panels fixed to the fuel cell walls. The 24PT4-DSC may contain up to 12 damaged fuel assemblies and up to 5 poison rodlets per undamaged fuel assembly. TS 4.2.3 has been revised to describe the two alternate minimum ^{10}B areal density specifications for the 24PT4-DSC Boral® panels: a standard loading of 0.025 g/cm^2 and a high loading of 0.068 g/cm^2. TS 4.2.3 also references

Table 2-8 of the TS which shows the maximum fuel enrichment versus minimum Boral® panel areal density and number of poison rodlets required for various configurations of damaged and intact fuel assemblies. Damaged fuel assemblies contained in failed fuel cans must be loaded in the basket as shown in TS Figure 2-4. The applicant stated that 75 percent credit was taken for the minimum ^{10}B content in the Boral® panels, and 64 percent credit was taken for the minimum ^{10}B content in the B_4C poison rodlets.

Both the bottom of the basket bottom spacer disk and the bottom of the guide tube and Boral® panel start at the same cask elevation. The active fuel length of the intact fuel assembly is therefore completely covered. Damaged fuel assemblies are assumed to reconfigure such that fuel can be present above or below the Boral® panel.

The staff reviewed Sections A.1, A.2, and A.6 of the SAR and verified that the design criteria and features important to criticality safety are clearly identified and adequately described. The staff also verified that the SAR contains engineering drawings, figures, and tables that are sufficiently detailed to support an in-depth staff evaluation.

Additionally, the staff 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. Section A.3 of the SAR shows that the basket will remain intact during all normal, off-normal, and accident conditions. Based on the information provided in the SAR, the staff concludes that the Standardized Advanced NUHOMS® System design with the 24PT4-DSC meets the double contingency requirements of 10 CFR 72.124(a).

6.2 Fuel Specification

The Standardized Advanced NUHOMS® 24PT4-DSC is designed to store 24 PWR assemblies in each canister. The assembly types allowed are limited to Westinghouse-CENP (CE) 16 x 16 spent fuel assemblies. The CE 16 x 16 assemblies may contain integral fuel burnable absorber (IFBA) rods. These assemblies are discussed in SAR Section A.2.1.1 and stated in Section 2.2 of the TS. The fuel assemblies are described in detail in Section A.6.2 of the SAR. The fuel specifications that are most important to criticality safety are:

- maximum initial enrichment
- number of fuel rods
- minimum clad outer diameter
- minimum clad thickness
- fuel rod pitch
- number of guide tubes

The parameters listed above represent the limiting or bounding parameters for the fuel assemblies. In terms of criticality safety, the most important fuel specification is the fuel initial enrichment. The 24PT4-DSC may contain 24 PWR assemblies with maximum initial enrichments up to 4.85 wt% ^{235}U, depending on the number of damaged fuel assemblies present. Table 2-8 of the TS describes the allowable configurations of damaged and intact fuel assemblies.

Specifications on the condition of the fuel are also included in the SAR and TS. The 24PT4-DSC is designed to accommodate intact fuel assemblies or up to 12 damaged fuel assemblies (depending on the minimum ^{10}B areal density of the Boral® panels and the number of B_4C poison rodlets included with the intact assemblies), as defined in the TS. The damaged fuel must be placed in individual failed fuel cans, which are designed to confine gross fuel particles to a known, subcritical geometry. Up to 12 failed fuel cans may be placed in the perimeter basket locations of the 24PT4-DSC, as shown in TS Figure 2-4. Reconstituted fuel assemblies, with up to eight rods replaced with stainless steel rods or any number of rods replaced with zirconium-clad uranium rods, may be stored as intact or damaged assemblies.

In Section A.3.5 of the SAR, the applicant has shown that the fuel cladding will not fail during the cask drop accidents which bound all storage conditions. Thus the criticality analysis need only consider intact fuel pins for the undamaged fuel.

Staff verified that all fuel assembly parameters important to criticality safety have been included in the TS. The staff reviewed the fuel specifications considered in the criticality analysis and verified that they are consistent with the specifications given in Sections A.1, A.2, and A.12 of the SAR and TS.

6.3 Model Specification

6.3.1 Configuration

The Standardized Advanced NUHOMS® System evaluated in this amendment consists of the 24PT4-DSC, a TC, and an AHSM. The applicant used three-dimensional calculation models in its criticality analyses. The bounding model is based on a fully flooded 24PT4-DSC in a TC. Figures containing the details of the criticality models are provided in Section A.6.3 of the SAR. The models are based on the engineering drawings in Section A.1 of the SAR and consider the worst-case dimensional tolerance values. The design-basis off-normal events do not affect the criticality safety design features of the cask system. Under accident conditions, the neutron shield of the TC is replaced with water. Failure of the damaged fuel assemblies within the failed fuel cans was also considered.

The normal condition model combined the most reactive basket dimensions. The applicant assumed the least material condition for the guide tubes and wrappers, thereby minimizing neutron absorption in the steel and maximizing the amount of moderator present in the model. The applicant also assumed the maximum guide tube opening and spacer disk cutout size, which allows for the closest spacing of fuel assemblies in the basket. The most reactive fuel and basket dimension combinations were determined to be the following: nominal fuel pellet diameter, minimum fuel cladding thickness, minimum fuel cladding outer diameter, minimum Boral® panel thickness, and fuel assemblies shifted toward the center of the basket. For the failed fuel cases, the fuel rod pitch is allowed to vary from the nominal to the most reactive within the internal dimensions of the failed fuel can.

The calculation models also conservatively assumed the following:

- fresh fuel isotopics (i.e., no burnup credit),
- omission of spacer grids, spacers, and hardware in the fuel assembly,

- infinite axial length of the cask and active fuel,
- 75% credit for the ^{10}B content in the Boral® panels,
- 64% credit for the ^{10}B content in the B_4C poison rodlets, and
- flooding of the fuel rod gap regions with pure water whenever the cask contains water

The applicant considered various levels of internal and external moderation to determine the most reactive moderating conditions. The applicant determined that the most reactive condition is with the inside of the canister flooded with full density water. Changes in external moderation had no observable effect on the reactivity of the system. Preferential flooding of regions within the canister is not considered in the criticality analysis due to the slots which are provided at the bottom of the guide tubes and failed fuel cans to ensure uniform draining and filling of all areas of the canister. Failed fuel can mesh screens will also not interfere with draining or filling, due to their 6 x 6 mesh size and 0.047 inch wire diameter.

The model for criticality analysis under accident conditions assumed the replacement of the neutron shield with water. For damaged fuel assemblies, the applicant considered single and double ended shearing of 16 fuel rods (one assembly face), allowing for the repositioning of rods and rod fragments. The possibility of rod fragments reconfiguring above the Boral® panels was addressed by modeling a 16 x 16 array of bare, 5.25 inch fuel rods in that region of the canister. The applicant also considered variation in the pitch of damaged fuel assemblies, up to the maximum allowed by the inside dimension of the failed fuel can.

The staff reviewed the applicant's criticality models for the Standardized Advanced NUHOMS® System and agrees that they are consistent with the description of the cask and contents given in Sections A.1 and A.2 of the SAR, including the engineering drawings. Based on the information presented, the staff has reasonable assurance that the most reactive combination of cask parameters and dimensional tolerances were incorporated into the calculation models, or are bounded by the assumptions used in these models.

For its confirmatory analyses, the staff independently modeled the cask system using the engineering drawings and bills of materials presented in Section A.1.5 of the SAR. Models of the cask system and its contents created by the staff were similar to those presented by the applicant.

6.3.2 Material Properties

The composition and density of the Boral® and B_4C neutron absorber materials considered in the calculation models are provided in Table A.6.3-2 of the SAR. In Sections A.6.3.2, A.9.1.7, and A.9.1.9 of the SAR, the applicant provided a detailed description of the characteristics, historical applications, service experience, and manufacturing quality assurance of the Boral® and B_4C neutron absorber materials. The minimum required ^{10}B content is verified through the acceptance testing programs described in Section A.9.1.7 for the Boral® panels and in Section A.9.1.9 for the B_4C poison rodlets. As previously stated, the maximum ^{10}B credit taken is 75% for the Boral® panels and 64% for the B_4C poison rodlets.

The continued efficacy of the neutron absorber materials over a 20-year storage period is assured by the design of the Standardized Advanced NUHOMS® System. The applicant demonstrated that the neutron flux from the irradiated fuel results in a negligible depletion of the

^{10}B content in the neutron absorber materials. In addition, a structural analysis was performed which demonstrated that the neutron absorbers will remain in place during accident conditions. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

The staff reviewed the composition and number densities for the materials used in the Standardized Advanced NUHOMS® System and found them to be acceptable. Based on the information provided on the neutron absorber material, the staff agrees that the Standardized Advanced NUHOMS® System meets the requirements of 10 CFR 72.124(b) regarding continued efficacy of neutron absorbers.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant performed the criticality analyses for the Standardized Advanced NUHOMS® System using the CSAS25 module of the SCALE 4.4 code system (Ref. 3), with KENO V.a and the 44-group ENDF/B-V cross-section library. Staff confirmatory calculations were performed using the 4.4a version of the SCALE code. KENO V.a is a three-dimensional Monte Carlo multi-group neutron transport code used by the SCALE system to calculate k_{eff}. This code is a standard in the nuclear industry for performing criticality analyses. The staff agrees that the codes and cross-section sets used in the criticality analysis are appropriate for this particular application.

6.4.2 Multiplication Factor

Results of the applicant's criticality analyses show that the k_{eff} of the Standardized Advanced NUHOMS® System will remain below the calculated USL of 0.9411 under all conditions. Tables A.6.4-2 through A.6.4-5, and A.6.4-14 of the SAR present the results of the applicant's criticality calculations for the bounding fuel assembly configurations. Table A.6.1-1 of the SAR and the two following tables summarize the maximum k_{eff} calculated by the applicant for each fuel configuration and basket type.

Table 1: Maximum k_{eff} for Assemblies in a Type A Basket (0.025 g/cm^2 ^{10}B)

Configuration	Maximum Initial Enrichment	$K_{eff} + 2\sigma$
24 Intact Assemblies	4.10 wt. %	0.9383
4 Damaged Assemblies + 20 Intact Assemblies, no B_4C Rodlets	4.10 wt. %	0.9361
12 Damaged Assemblies + 12 Intact Assemblies, no B_4C Rodlets	3.70 wt. % (damaged) 4.10 wt. % (intact)	0.9393
12 Damaged Assemblies + 12 Intact Assemblies, 1 B_4C Rodlet per Intact Assembly	4.10 wt. %	0.9341

Table 2: Maximum k_{eff} for Assemblies in a Type B Basket (0.068 g/cm^2 ^{10}B)

Configuration	Maximum Initial Enrichment	$K_{eff} + 2\sigma$
24 Intact Assemblies	4.85 wt. %	0.9389
4 Damaged Assemblies + 20 Intact Assemblies, no B_4C Rodlets	4.85 wt. %	0.9382
12 Damaged Assemblies + 12 Intact Assemblies, no B_4C Rodlets	4.10 wt. % (damaged) 4.85 wt. % (intact)	0.9391
12 Damaged Assemblies + 12 Intact Assemblies, 5 B_4C Rodlets per Intact Assembly	4.85 wt. %	0.9378

The staff reviewed the applicant's calculated k_{eff} values and agrees that they have been appropriately adjusted to include all biases and uncertainties at a 95% confidence level or better.

The staff performed independent criticality calculations for the Standardized Advanced NUHOMS® System with the 24PT4-DSC. The modeling assumptions used by the staff were similar to those used by the applicant. The staff's model considered the most reactive conditions in modeling each of the intact and damaged spent fuel configurations described in the above two tables. The results of the staff's confirmatory analyses were in close agreement with the applicant's results.

Based on the applicant's criticality evaluation, as confirmed by the staff, the staff has reasonable assurance that the Standardized Advanced NUHOMS® System with the 24PT4-DSC will remain subcritical under all credible normal, off-normal, and accident conditions.

6.4.3 Benchmark Comparisons

The applicant performed benchmark calculations on critical experiments selected, as much as possible, to bound the range of variables in the Standardized Advanced NUHOMS® System design. The parameters in the 121 benchmark experiments selected bounded the parameters in the analysis with respect to fuel enrichment, fuel pin pitch, ^{10}B concentration in the control elements, water to fuel volume ratio, assembly separation, and average energy group. USL Method 1: Confidence Band with Administrative Margin, from NUREG/CR-6361, *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages* (Ref. 4), was used to determine the USL. The applicant stated that the benchmark calculations were performed with the same computer codes and cross-section data and on the same computer hardware used in the criticality calculations for the Standardized Advanced NUHOMS® System.

The USL resulting from the applicant's benchmark analysis is 0.9411. This USL incorporates the biases and uncertainties of the model and computer code into a value that has a 95%

confidence level such that any k_{eff} less than the USL is less than 0.95, which is the design criterion.

The staff reviewed the benchmark comparisons in the SAR and agrees that the computer code used for the analysis was adequately benchmarked using representative critical experiments. The staff also reviewed the applicant's method for determining the USL and found it to be acceptable and conservative. Additionally, the staff verified that only biases that increase k_{eff} have been applied.

6.5 Supplemental Information

The spent fuel assemblies that can be loaded into the Standardized Advanced NUHOMS® System with the 24PT4-DSC without compromising criticality safety requirements are listed in the TS. All supportive information has been provided in the SAR, primarily in Sections A.1, A.2, and A.6.

6.6 Evaluation Findings

Based on the information provided in Revision 1 of the SAR and the staff's own confirmatory analyses, the staff concludes that the Standardized Advanced NUHOMS® System with the 24PT4-DSC meets the acceptance criteria specified in NUREG-1536. In addition, the staff finds the following:

F6.1 SSCs important to criticality safety are described in sufficient detail in Sections A.1, A.2, and A.6 of the SAR to enable an evaluation of their effectiveness.

F6.2 The Standardized Advanced NUHOMS® System with the 24PT4-DSC is designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry and fixed neutron absorbers. An appraisal of the fixed neutron absorbers has shown that they will remain effective for the 20-year storage period. In addition, there is no credible way to lose the fixed neutron absorbers; therefore, requirements of 10 CFR 72.124(b) have been met.

F6.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will provide for the safe storage of spent fuel for 20 years with an adequate margin of safety.

F6.5 The staff concludes that the criticality design features for the Standardized Advanced NUHOMS® System with the 24PT4-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 Standardized Advanced NUHOMS® System with the 24PT4-DSC will allow safe storage of spent nuclear fuel. This finding 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. SCALE4.4, “A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” Oak Ridge National Laboratory, March 1997.

4. U.S. Nuclear Regulatory Commission, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” NUREG/CR-6361, March 1997.

7.0 CONFINEMENT EVALUATION

The staff reviewed the NUHOMS® 24PT4-DSC confinement features and capabilities to ensure a) that any radiological releases to the environment will be within the limits established by the regulation (Ref. 1), 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® 24PT4-DSC fulfills the acceptance criteria listed in Section 7 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems (Ref. 2), and applicable Interim Staff Guidance documents (ISGs). The staff's conclusions are based on information provided in the NUHOMS® 24PT4-DSC SAR.

7.1 Confinement Design Characteristics

The confinement boundary of the NUHOMS® 24PT4-DSC is described as follows: The cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies, the vent and siphon block, the vent and siphon cover plates, and the associated welds. An outer top cover plate and associated welds function as a redundant welded barrier to radioactive material release, meeting the requirement of 10 CFR 72.236(e). All penetrations in the DSC confinement boundary are welded closed.

All welds involving the confinement boundary are either partial or full penetration welds, with a minimum of two passes, and inspected in accordance with the requirements of Section III, Subsection NB, of the ASME Code (e.g., radiographic or ultrasonic and liquid penetrant). Additionally, the circumferential and longitudinal seam welds and the inner bottom cover plate weld are pressure tested in accordance with Section III, Article NB-6300, of the ASME Code.

7.2 Confinement Monitoring Capability

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

In lieu of performing leak testing of the closure welds for the top inner shield plug assembly cover plate, the vent and siphon cover plates, and the outer top cover plate, the applicant has demonstrated that these welds and applicable non-destructive examinations meet the applicable requirements demonstrating DSC integrity as set forth in NRC Interim Staff Guidance Memoranda (ISG) Nos. 5, Revision 1, "Confinement Evaluation," dated May 21, 1999; ISG-15, "Materials Evaluation," dated January 10, 2001; and ISG-18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," dated May 2, 2003. Hence, there is no contribution to the radiological consequences due to a potential release of canister contents.

7.4 Confinement Analysis

TN has demonstrated that the welds and applicable non-destructive examinations meet the applicable requirements demonstrating DSC integrity as set forth in ISG-18, as follows:

1. The DSC is fabricated from austenitic stainless steel;

2. The DSC closure welds meet the requirements of ISG-15, Section X.5.2.3, "Weld Design and Specifications," or an approved alternative [e.g., multi-pass welding, stress reduction factor of 0.8 (TN conservatively assumes 0.7)];

3. The DSC maintains its integrity during normal operating conditions, anticipated off-normal conditions, and credible accidents, including natural phenomena, as required by 10 CFR Part 72;

4. Records documenting the fabrication and closure welding of DSCs meet the requirements of 10 CFR 72.174, "Quality Assurance Records," ANSI N45.2.9, "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants," and ISG-15; and

5. Activities related to the inspection, documentation, and welding of DSCs are performed in accordance with an NRC-approved quality assurance program, as required by 10 CFR 72, Subpart G, "Quality Assurance."

The confinement boundary is shown to maintain confinement during all normal, off-normal, and accident conditions. Also, the temperature and pressure of the canister are within design-basis limits. Therefore, no discernable leakage is credible. As discussed in Sections 5 and 10 of this SER, the staff finds that the NUHOMS® 24PT4-DSC meets the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

7.5 Supporting Information

Supporting information or documentation includes drawings of the NUHOMS® 24PT4-DSC confinement boundary and applicable pages from referenced documents.

7.6 Evaluation Findings

F7.1 Section A.7 of the SAR describes confinement SSCs important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the NUHOMS® 24PT4-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® 24PT4-DSC provides redundant sealing of the confinement system closure joints using dual welds on the canister lid and closure.

F7.4 The NUHOMS® 24PT4-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 NUHOMS® 24PT4-DSC uses an entirely welded redundant closure system, no direct monitoring of the closure is required.

F7.5 The confinement system will reasonably maintain confinement of radioactive material. Section 10 of the SER shows that the direct dose from the NUHOMS® 24PT4-DSC satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F7.6 The staff concludes that the design of the confinement system of the NUHOMS® 24PT4-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® 24PT4-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

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® 24PT4-DSC, as described in Section A.8 of the SAR are very similar to those previously approved by the staff for the Standardized Advanced NUHOMS® System (Ref. 1).

8.1 Cask Loading

Detailed loading procedures must be developed by each user.

The loading procedures described in Section A.8 of 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® 24PT4-DSC to identify any damage that may have occurred since receipt inspection.

8.1.1 Fuel Specifications

The procedures described in Section A.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® 24PT4-DSC. It outlines appropriate administrative controls to preclude a cask misloading. Failed fuel cans, required for loading damaged fuel assemblies, must replace the DSC guidesleeves at the locations specified for the specific basket configurations within the NUHOMS® 24PT4-DSC basket.

8.1.2 ALARA

The ALARA practices utilized during operations are discussed in Section 10 of this SER and are found to be acceptable.

8.1.3 Draining, Drying, Filling and Pressurization

Section A.8 of the SAR clearly describes draining, drying, filling and pressurization procedures for the NUHOMS® 24PT4-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 Advanced NUHOMS® System.

8.1.4 Welding and Sealing

Welding and sealing operations of the NUHOMS® 24PT4-DSC are similar to that previously approved by the staff for the other DSC used with the Standardized Advanced NUHOMS® System. The procedures include monitoring for hydrogen during welding operations. As described in Section 7 of this SER, DSC integrity is ensured by meeting the requirements of ISG-5 and ISG-18 for leaktightness.

8.2 Cask Handling and Storage Operations

All handling and transportation events applicable to moving the NUHOMS® 24PT4-DSC to the storage location are similar to those previously reviewed by the staff for the Standardized Advanced NUHOMS® System are bounded by Section A.11 of the SAR. Monitoring operations include surveillance of the AHSM air inlets and outlets in accordance with TS 5.2.5, and temperature performance in accordance with TS 5.2.5. Occupational and public exposure estimates are evaluated in Section A.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 A.8 provides unloading procedures similar to those previously approved by the staff for use with the Standardized Advanced NUHOMS® System. The procedures provide a caution on reflooding the DSC to ensure that the cask vent pressure does not exceed 20 psig to prevent damage to the cask.

Section A.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® 24PT4-DSC is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Section A.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 DSC 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.

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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® 24PT4-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.

8.5 Reference

Transnuclear, Inc, Final Safety Analysis Report (FSAR) for the Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 0, February 2003, USNRC Docket Number 72-1029.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAMS

9.1 Acceptance Tests

All materials and components will be procured with certification and supporting documentation to assure compliance with procurement specifications and receipt inspected for visual and dimensional traceability .

9.1.1 Visual and Nondestructive Examination Inspections

The DSC confinement boundary is fabricated and inspected in accordance with ASME Code Section III, Subsection NB. Alternatives to the ASME Code are identified in Chapter A.3 of the SAR. The staff reviewed these alternatives, and the corresponding justifications, and found them to be acceptable.

The nondestructive examination (NDE) of weldments is well characterized in the drawings and discussed in Sections of the SAR. Standard NDE symbols and/or notations are used in accordance with AWS 2.4, “Standard Symbols for Welding, Brazing, and Nondestructive Examination.” Fabrication inspection include visual (VT), liquid penetrant (PT), ultrasonic (UT), and radiographic (RT) examinations, as applicable.

9.1.2 Leakage Testing

The NUHOMS® 24PT4-DSC is designed and tested at fabrication to be leaktight, as specified in “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials,” ANSI N14.5-1997 (Ref. 1). During fabrication, leak tests of the 24PT4-DSC shell assembly and the bottom closure welds are performed in accordance with ANSI N14.5-1997 to demonstrate that the shell is leaktight.

9.1.3 Neutron Absorber Tests

There are two types of neutron absorbers (also called poisons) used in the 24PT4 DSC basket. They are Boral® and boron carbide (B_4C) pellets or powder encapsulated in stainless steel tubes.

The Boral® neutron absorber has an minimum total ^{10}B area density of 0.025 gm/cm^2, for Type A basket and 0.068 gm/cm^2 for Type B basket. The acceptance program for the Boral® neutron absorber remains the same as in the original approved SAR. The acceptance program supports crediting 75% of the Boron loading specified for fabrication in the criticality analysis. Visual inspection of all Boral® plates is performed to ensure that they are free of cracks, porosity, blisters, or foreign inclusions. Dimensional inspections of all plates is also performed.

The boron carbide (B_4C) encapsulated in stainless steel tubes have a linear density of 0.70 gm/cm. The acceptance program supports crediting 64% of the boron loading specified for fabrication in the criticality analysis. Additional tests have been added for the B_4C encapsulated in stainless steel tubes (i.e., ASTM C751). The closure welds for the B_4C stainless steel enclosure tubes are to be liquid dye-penetrant inspected per ASME Code, Section V. Inspection criteria is to be ASME Code, Section III, Subsection NB 5350. Dimensional inspections of all rods is also performed.

9.2 Evaluation Findings

F9.1 Sections A.9.1.7 and A.9.1.9 of the SAR describes the applicant's proposed program for pre-operational testing and initial operations of the neutron absorber and B_4C encapsulated in stainless steel tubes in the 24PT4-DSC.

F9.2 The applicant will examine and/or test the 24PT4-DSC to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Sections A.9.1.3 and A.13 of the SAR describes this inspection and testing.

F9.3 The applicant will mark the cask with a data plate indicating its model number, unique identification number, and empty weight. Drawing ANUH-01-4001, sheet 6 of 7, note 47 in SAR Section A.1 illustrates and/or describes this data plate.

F9.4 The staff concludes that the acceptance tests and maintenance program for the 24PT4-DSC are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. 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 Reference

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

10. RADIATION PROTECTION EVALUATION

The staff evaluated the radiation protection design features, design criteria, and the operating procedures of the NUHOMS® 24PT4-DSC which will be used with the Advanced NUHOMS® Horizontal Storage Module (AHSM) 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), and 10 CFR 72.236(d). (Ref. 1)

10.1 Radiation Protection Design Criteria and Design Features

The radiological protection design criteria are the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. 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 radiation protection design features are referenced in Section A.10 of the SAR. The radiation protection design features of the NUHOMS® 24PT4-DSC/AHSM system are similar to the previously approved NUHOMS® 24PT1-DSC system.

The staff reviewed the design criteria and found it acceptable. The staff reviewed the integrated shielding ability of the NUHOMS® 24PT4-DSC/AHSM and found it acceptable. Chapters 5, 7, and 8 of this SER discuss specific staff evaluations of the design criteria and features for the shielding system, confinement systems, and operating procedures, as appropriate. Chapter 11 of this SER discusses staff evaluations of the capability of the shielding and confinement features during off-normal and accident conditions, as appropriate.

10.2 ALARA

The ALARA objectives, procedures, practices, and policies are referenced in Section A.10 of the SAR and the previously approved FSAR. The ALARA objectives, procedures, practices and policies of the NUHOMS® 24PT4-DSC/AHSM system are the same as the previously approved NUHOMS® 24PT1-DSC/AHSM system. Each site licensee will apply its additional site-specific ALARA objectives, policies, procedures, and practices for members of the public and personnel.

The staff evaluated the previously approved ALARA assessment for the NUHOMS® 24PT4-DCS/AHSM system and found it acceptable. Section 8 of this SER discusses the staff's evaluation of the operating procedures with respect to ALARA principles and practices are the responsibility of the site licensee as required by 10 CFR Part 20 and 10 CFR 72.104(b).

10.3 Occupational Exposures

The applicant determined occupational exposures in Section A.10 of the SAR. The exposures were based on estimates from surrounding direct radiation dose rates calculated in Chapter A.5 and the operating procedures referenced in Chapter A.8. The dose rates were based on a design basis bounding source term that was determined using 24 CE 16x16 fuel assembly with a burnup of 45 GWD/MTU, an initial enrichment of 3.8 wt. % 235U and 5 years cooling loaded in the DSC. The operating procedures are generic procedures that general licensees will use for fuel loading, canister/TC operations, canister transfer into the AHSM, and fuel unloading. Table A.10.3-1 of the amendment shows the estimated number of workers, completion time and estimated dose rate, for each task involved in the loading of an NUHOMS® 24PT4-DSC system. The dose estimates indicate that the total occupational dose is approximately 3970 person-mrem for a single loading. The applicant indicated that the general licensee may choose to modify the sequence of operations, and will also use ALARA practices to mitigate occupational exposure.

The staff reviewed the overall occupational dose estimates and found them acceptable. The occupational dose exposure estimates provide reasonable assurance that the occupational limits in 10 CFR Part 20 Subpart C can be achieved. The staff expects actual operating times and personnel exposure rates will vary for each system depending on site-specific operating conditions, including detailed procedures and special measures taken to maintain exposures ALARA. Each licensee will have an established radiation protection program, as required in 10 CFR Part 20, Subpart B. In addition, each licensee will demonstrate compliance with occupational dose limits in 10 CFR Part 20, Subpart C and other site-specific 10 CFR Part 50 licensee requirements with evaluations and measurements. Staff evaluation of the operating procedures is discussed in Section 8 of this SER.

10.4 Public Exposures From Normal and Off-Normal Conditions

Section A.10.2 of the amendment presents the calculated direct radiation doses at distances from 6 to 600 meters from a single AHSM and a generic cask array configuration loaded with design basis fuel. The applicant used MCNP to calculate off-site dose rates at large distances from a single AHSM and one generic ISFSI array. The generic array consists of a 2x10 back-to-back array of AHSMs loaded with 24 design basis fuel assemblies in the NUHOMS® 24PT4-DSC. Figures A.10.2-1 and A.10.2-2 depict estimated dose rate versus distance curves for a single AHSM and 2x10 back-to-back arrays, respectively. An array of 20 NUHOMS® 24PT4-DSC systems without site-specific shielding loaded with design basis fuel is below the regulatory limit of 25 mrem/yr at approximately 150 meters from the front and back of the AHSMs. 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 sky shine). The confinement function is not affected by normal or off-normal conditions. Therefore, no content leakage is credible. A discussion of the staff's evaluation and confirmatory analysis of the shielding calculations and confinement analysis are presented in Sections 5 and 7 of this SER, respectively.

The staff reviewed the shielding models and found them acceptable. In addition, the methodologies used in this analysis are similar to those used to support the previous Advanced NUHOMS® storage application.

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® 24PT4-DSC system 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 individuals 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. 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 general licensee.

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 by 10 CFR Part 20, Subpart D by evaluations and measurements.

10.5 Public Exposures From Design-Basis Accidents and Natural Phenomena Events

Section A.11 of the SAR 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. Section A.11 of the SAR 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 is presented in Section 11 of this 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).

10.6 Evaluation Findings

F10.1 The SAR sufficiently describes the radiation protection design bases and design criteria for the SSCs 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® 24PT4-DSC is designed to facilitate decontamination to the extent practicable.

F10.4 The SAR adequately evaluates the NUHOMS® 24PT4-DSC and its systems important to safety that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F10.5 The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

F10.6 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® 24PT4-DSC is designed to assist in meeting these requirements.

F10.7 The staff concludes that the design of the radiation protection system of the NUHOMS® 24PT4-DSC when used with the AHSM, 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® 24PT4-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 System, NUREG-1536, January 1997.

3. U.S. Nuclear Regulatory Commission, Information Relevant to Ensuring that Occupational Radiation Exposures Will Be As Low As 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 ANALYSES

The objective of the accident analysis evaluation is to ensure that the applicant has identified and analyzed potential hazards for both off-normal and accident or design basis events for this amendment request. Thus, the evaluation will concentrate on those analyses specific to the use of the 24PT4-DSC for storage in the Advanced Horizontal Storage Module (AHSM) and transfer in OS197H Transfer Cask (TC).

The 24PT4-DSC has been designed to accommodate Westinghouse-CENP 16x16 (CE16x16) intact and /or damaged PWR fuel assemblies. The configuration of the 24PT4-DSC is very similar to that of the 24PT1-DSC. As a result, when description or analysis presented in the FSAR for the Advanced NUHOMS® system with 24PT1-DSC is applicable, the descriptions and analyses are not repeated in this amendment.

The Accident Dose Calculations Section reports the expected dose resulting from the postulated event in terms of whole body dose only. The all-welded leaktight canister design and maintenance of confinement boundary integrity under all credible off-normal and accident scenarios ensured no radiation leakage from the 24PT4-DSC, thus, dose consequences have been limited to direct and scattered radiation doses without any associated inhalation or ingestion doses.

11.1 Off-Normal Operations

The application has identified two off-normal events which will bound the range of off-normal conditions as follows:

1. A “jammed” 24PT4-DSC during loading or unloading from the AHSM.

2. The extreme ambient temperatures of -40EF and +117EF.

The discussion in Section 11.1.1, “Off-Normal Transfer Loads,” is also applicable to the jammed 24PT4-DSC during loading or unloading. The applicant performed new thermal analysis of the Advanced NUHOMS® System with the 24PT4-DSC and CE16x16 fuel for the extreme ambient temperatures. The analysis results are presented in Chapter A.4. The Technical Specifications require that a transfer cask solar shield must be installed on the OS197H transfer cask when the ambient temperature is greater than 100 EF. There is no radiological impact resulting from the off-normal operations.

11.2 Accident -Level Events and Conditions

The application states that the discussion in Section 11.2 of the FSAR for the 24PT1-DSC also applies to 24PT4-DSC. The earthquake stress evaluations for the AHSM presented in Sections 3.6 and 11.2.1 for the 24PT1-DSC were based on a bounding weight of 85,000 lbs. Thus, the analysis results presented in the FSAR are applicable to the AHSM loaded with a 24PT4-DSC. For tornado wind, tornado missile, and flooding loading conditions, the calculated overturning, sliding, and missile impact analyses are bounding for the AHSM loaded with a 24PT4-DSC because the stabilizing moment against overturning and the force required to slide the AHSM were based on the slightly lower weight of the 24PT1-DSC. The applicant performed a new specific evaluation of the hypothetical fire event for the 24PT4-DSC and the analysis results are presented in Chapter A.4 of the SAR. Drop accidents are postulated for the 24PT4-DSC only when it is positioned inside the transfer cask, since drop accidents cannot occur once the 24PT4-DSC has been transferred into the AHSM. The structural integrity of the 24PT4-DSC shell and internal basket assemblies are evaluated for the drop accident conditions. The stress analyses of the 24PT4-DSC resulting from the postulated drop scenarios are presented in Section A.3.6.

The transfer cask neutron shield may be damaged or lost in an accidental transfer cask drop event. Based on the MCNP transfer cask model described in Chapter A.5, with the neutron shield eliminated, the average dose rate at the transfer cask surface is 1796 mrem/hr (i.e., 862 mrem/hr gamma and 934 mrem/hr neutron). It is assumed that 8 hours are required to either recover the neutron shield or add temporary shielding while arranging recovery operations. As a result, it is estimated that on-site workers at an average distance of 15 feet would receive an additional dose of 1.6 rem and off-site individuals at a distance of 200 feet would receive an additional dose of 0.18 mrem. Based on these additional doses, the applicant estimated that the accident dose from a damaged cask would be approximately 69 mrem at 100 meters. The dose increase is well within the limits of 10 CFR 72.106 for accident condition (i.e., 5000 mrem at the nearest boundary and 100 meters from an ISFSI). The lead shielding provided by the top and bottom shield plug may be reduced during a cask side drop condition. During a postulated side drop, lead slump filling all potential voids in the shielding plug would hypothetically result in a small area of the shield plug to be without lead shielding. The application estimated that the effect of this streaming path is to increase the end dose by approximately 50%. The end dose increase is insignificant when comparing it to the side dose increase in an accidental side drop event.

11.3 Evaluation Findings

F11.1 The applicant has evaluated the 24PT4-DSC for storage in the NUHOMS® Advanced Horizontal Storage Module (AHSM) and for transfer in OS197H Transfer Cask to demonstrate that it will reasonably maintain confinement of radioactive material under applicable off-normal and design basis accident or a natural phenomenon events.

F11.2 The applicant has evaluated off-normal and design basis accident conditions to demonstrate with reasonable assurance that the Advanced NUHOMS® Horizontal Storage Module System loaded with a 24PT4-DSC has adequate radiation shielding capability to meet the requirements in 10 CFR 72.104(a).

F11.3 The spent fuel will be maintained in a subcritical condition under accident conditions.

F11.4 The staff concludes that the accident design criteria for the NUHOMS® AHSM and the 24PT4-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 system 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.

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® 24PT4-DSC, in conjunction with the Standardized Advanced NUHOMS® Storage System, are clearly defined in the CoC and TS.

The staff made a revision to the CoC by removing the reference to the Technical Specification Bases. The removal of this reference is consistent with guidance specified in NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance."

12.2 Technical Specifications

Based on the addition of the NUHOMS® 24PT4-DSC to the Standardized Advanced NUHOMS® Storage System, the TS have been revised to accommodate the new DSC and the fuel types to be stored in the DSC. These changes have been identified in the TS, which are referenced as Appendix A to the CoC.

Limiting Condition For Operation (LCO) 3.1.4, "24PT1-DSC Helium Leak Rate of Inner Top Cover Plate Weld and Vent/Siphon Port Cover Welds" has been removed from the TS. The removal of this LCO is consistent with guidance specified in ISG-18. The basis for this revison is discussed in Section 7 of this SER.

Table 12-1 lists the TS for use of the NUHOMS® 24PT4-DSC system, in concert with the Standardized Advanced NUHOMS® Storage System.

12.3 Evaluation Findings

F12.1 Table 12-1 of this SER lists the TS for the NUHOMS® 24PT4-DSC, in concert with the Standardized Advanced NUHOMS® Storage System. These TS are identified as Appendix A of the CoC.

F12.2 The staff concludes that the conditions for use of the NUHOMS® 24PT4-DSC, in concert with the Standardized Advanced NUHOMS® Storage system, identify necessary TS to satisfy 10 CFR Part 72 and that the applicant acceptance criteria have been satisfied. The TS 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 Advanced NUHOMS® Horizontal Modular Storage System
Technical Specifications for use with the NUHOMS® 24PT4-DSC

Operating Controls and Limits

- 1.0 Use and Application
  - 1.1 Definitions
  - 1.2 Logical Connectors
  - 1.3 Completion Times
  - 1.4 Frequency
- 2.0 Functional and Operational Limits
  - 2.1 Fuel to be Stored in the NUHOMS® 24PT1-DSC
  - 2.2 Fuel to be Stored in the NUHOMS® 24PT4-DSC
  - 2.3 Functional and Operating Limits Violations
- 3.0 Limiting Condition for Operation (LCO) and Surveillance Requirement (SR) applicability
  - 3.1 DSC Integrity
    - 3.1.1.a NUHOMS® 24PT1-DSC Vacuum Drying time (Duration) and Integrity
    - 3.1.1.b NUHOMS® 24PT4-DSC Vacuum Drying time (Duration) and Integrity
    - 3.1.2.a NUHOMS® 24PT1-DSC Helium Backfill Pressure
    - 3.1.2.b NUHOMS® 24PT4-DSC Helium Backfill Pressure
- 4.0 Design Features
  - 4.1 Site
    - 4.1.1 Site Location
  - 4.2 Storage System Features
    - 4.2.1 Storage Capacity
    - 4.2.2 Storage Pad
    - 4.2.3 Canister Neutron Absorber
    - 4.2.4 Canister Flux Trap Configuration
    - 4.2.5 Fuel Spacers
  - 4.3 Codes and Standards
    - 4.3.1 Advanced Horizontal Storage Module (ASHM)
    - 4.3.2 Dry Shielded Canister, 24PT1-DSC or 24PT4-DSC
    - 4.3.3 Transfer Cask
    - 4.3.4 Alternatives to Codes and Standards
  - 4.4 Storage Location Design Features
    - 4.4.1 Storage Configuration
    - 4.4.2 Concrete Storage Pad Properties to Limit DSC Gravitational Loadings Due to Postulated Drop
    - 4.4.3 Storage Location Design Features
- 5.0 Administrative Controls
  - 5.1 Procedures
  - 5.2 Programs
    - 5.2.1. Safety Review Program

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether TN has a quality assurance program that complies with the requirements of 10 CFR Part 72, Subpart G. The staff has previously reviewed and accepted the TN quality assurance program in the Standardized Advanced NUHOMS® Horizontal Modular Storage System FSAR.

14.0 DECOMMISSIONING

The decommissioning evaluation was previously reviewed and approved in the Standardized Advanced NUHOMS® Horizontal Modular Storage System FSAR. There were no changes proposed by the addition of the NUHOMS® 24PT4-DSC.

CONCLUSIONS

The staff performed a detailed safety evaluation of the proposed CoC amendment request and found that the addition of the NUHOMS® 24PT4-DSC does not reduce the safety margin for the Standardized Advanced NUHOMS® Horizontal Modular Storage System. Based on the statements and representations contained in the applicant's SAR, and the conditions of the CoC, the staff concludes that the addition of the NUHOMS® 24PT4-DSC to the approved contents of the Standardized Advanced NUHOMS® Horizontal Modular Storage System meets the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1029, Amendment 1,
on ___May 31, 2005________.