TRANSNUCLEAR
STANDARDIZED ADVANCED
NUHOMS®
HORIZONTAL MODULAR
STORAGE SYSTEM
FOR IRRADIATED NUCLEAR FUEL

SAFETY EVALUATION REPORT

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DOCKET NO. 72-1029
MODEL NO. STANDARDIZED ADVANCED NUHOMS®-24PT1
TRANSNUCLEAR, INC.
CERTIFICATE OF COMPLIANCE NO. 1029

SUMMARY

By letter dated September 29, 2000, Transnuclear West, Inc. (TN West) submitted an application to obtain a Certificate of Compliance for the Standardized Advanced NUHOMS® System. The staff performed a detailed safety evaluation of the application which is documented in this safety evaluation report (SER). The staff's evaluation and conclusions regarding the acceptability of the Standardized Advanced NUHOMS® System are based on information submitted by TN West on September 29, 2000, as supplemented. The staff determined that the Standardized Advanced NUHOMS® System meets the requirements of 10 CFR Part 72.

In addition, the staff determined that all analytical methods used by the applicant in the design of the Standardized Advanced NUHOMS System, as described in the SAR, are acceptable with the exceptions involving the thermal, shielding, and criticality analyses as summarized in the conclusion of this SER.

1.0 GENERAL DESCRIPTION

The objective of the review of the general description of the Standardized Advanced NUHOMS® System is to ensure that TN West has provided a non-proprietary description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system.

1.1 General Description and Operations Features

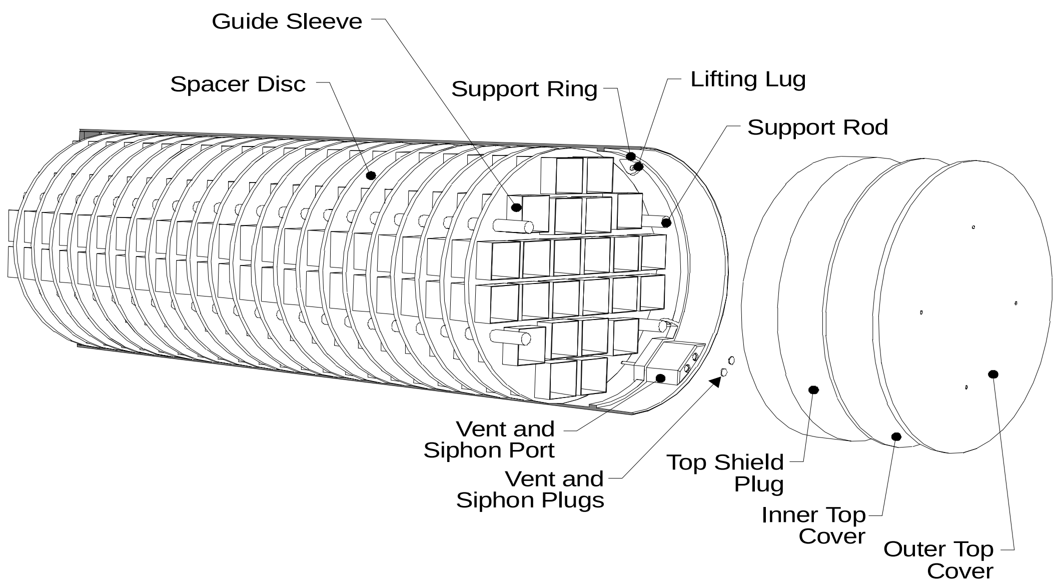
The Standardized Advanced NUHOMS® System is based on the Standardized NUHOMS® System described in Certificate of Compliance (CoC) No. 1004. The 24PT1 dry shielded canister (DSC) included in this system is nearly identical to that previously approved for transportation in the NUHOMS® MP-187 transportation package, CoC 71-9255, and for use at the Rancho Seco Nuclear Plant in accordance with special nuclear material license SNM-2510.

The 24PT1 DSC will be transferred during loading operations using the OS-197 transfer cask (TC). The OS-197 was previously approved for use in CoC 1004 and was reviewed in this application to ensure that the previous analysis demonstrated the ability of the OS-197 to be used with the Standardized Advanced NUHOMS® System. The 24PT1 DSC will be stored in the advanced horizontal storage module (AHSM).

1.1.1 Dry Shielded Canister (24PT1 DSC)

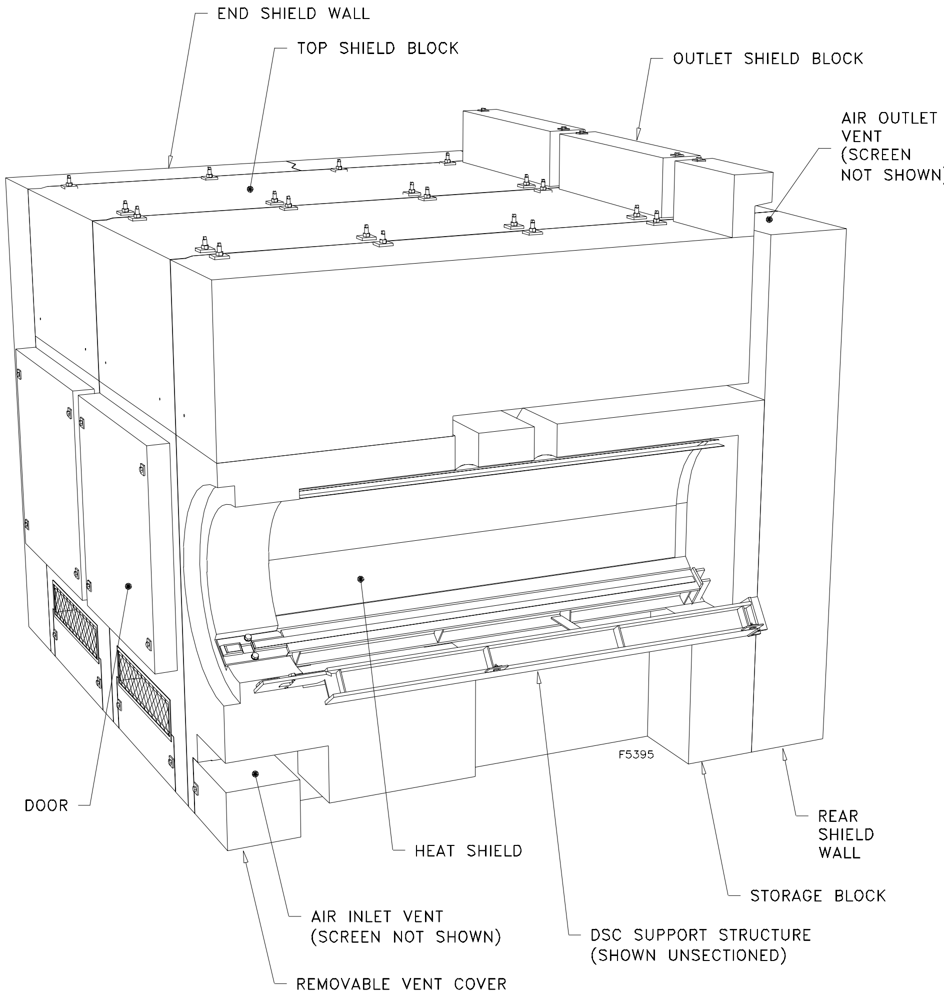
The 24PT1 DSC is designed to store 24 intact pressurized water reactor (PWR) Westinghouse 14X14 (WE 14X14) fuel assemblies with or without rod cluster control assemblies (RCCAs), neutron source assemblies (NSAs), or thimble plug assemblies (TPAs). Westinghouse 14X14 fuel assemblies mixed oxide fuel may also be stored with or without inserts. The 24PT1-DSC is also designed to store up to 20 intact and 4 damaged fuel assemblies in failed fuel cans. In addition, two slots in the 24PT1-DSC may be filled with stainless steel model assemblies which do not contain fuel. These model (or dummy) assemblies would be used in the event that the cask user requires only 22 spent fuel assemblies to be stored in the cask. The 24PT1-DSC is designed for a maximum heat load of 14kW.

The 24PT1-DSC consists of a cylindrical shell with welded top and bottom cover plates which form the confinement boundary. Shield plugs are installed inside of the confinement boundary, at the top and bottom, to provide radiological shielding. Inside of the 24PT1-DSC is a basket assembly that consists of 24 guide sleeves which contain fixed borated neutron absorbing material for criticality control during loading operations. The structural support for the PWR fuel and basket guide sleeves is provided by circular spacer disc plates.



1.1.2 Advanced Horizontal Storage Module (AHSM)

The AHSM is constructed of reinforced concrete and structural steel. The key design parameters of the AHSM are provided in Table 1.2-1 of the SAR. The design is similar to the previously approved Standardized NUHOMS® System HSM, however, the AHSM contains improved shielding and resistance to high seismic events. The function of the AHSM is to ensure that under normal conditions and postulated accidents, including natural phenomena, do not impair the ability of the 24PT1-DSC to safely store the fuel. The AHSM is designed to passively remove heat from the 24PT1-DSC by natural circulation of airflow through the cask.



The AHSMs are located on a reinforced concrete pad and fastened to adjacent AHSMs. For design basis seismic events a minimum of three AHSMs must be fastened together.

1.1.3 Transfer System

The OS-197 TC, used with the Standardized Advanced NUHOMS® System, provides shielding protection from the 24PT1-DSC during loading and unloading operations. The OS-197 was previously approved for use by Certificate of Compliance 1004 and was only evaluated in this SER to the determine whether it could be used with the 24PT1-DSC.

The transfer trailer is not important to safety because the Standardized Advanced NUHOMS® System Technical Specifications (TS) limit the lifting height of the 24PT1-DSC to eighty inches which is within the design basis drop for the DSC. Therefore, the transfer trailer was not evaluated in this SER.

1.2 Drawings

Section 1 of the SAR contains the non-proprietary drawings for the Standardized Advanced NUHOMS® System, including drawings of the structures, systems, and components (SSC) important to safety. The staff determined that the drawings contain sufficient detail on dimensions, materials, and specifications to allow for a thorough evaluation of the Standardized Advanced NUHOMS® System. Specific SSC are evaluated in Sections 3 through 12 of this SER.

1.3 24PT1-DSC Contents

The NUHOMS® 24PT1-DSC may store up to 24 intact PWR WE 14X14 stainless steel clad (SC) fuel assemblies or 24 WE 14X14 mixed oxide (MOX) fuel assemblies with or without reactor cluster control assemblies (RCCAs), thimble plug assemblies (TPAs), and neutron source assemblies (NSAs). In addition, the 24PT1-DSC may be used to store up to 20 intact fuel assemblies plus 4 damaged fuel assemblies (as described in Section 2 of the SAR) in failed fuel cans. Additional fuel characteristics are discussed in Sections 2 and 6 of the SAR.

1.4 Technical Qualifications of Applicant

Section 1.3 of the SAR contains details of the applicant's qualifications and experience regarding its ability to design and fabricate the Standardized Advanced NUHOMS® System in accordance with its approved 10 CFR Part 72 quality assurance program.

1.5 Evaluation Findings

F1.1 A general description of the Standardized Advanced NUHOMS® System is presented in Section 1 of the SAR with special attention to design and operating characteristics, unusual or novel design features and principal safety considerations.

F1.2 Drawings for SSC important to safety are presented in Section 1 of the SAR. Specific SSC are evaluated in Sections 3 through 12 of this SER.

F1.3 Specifications for the spent fuel to be stored in the Standardized Advanced NUHOMS® System are stated in SAR Sections 1, 2, and 6.

F1.4 The technical qualifications of the applicant to engage in the proposed activities are identified in Section 1.3 of the SAR.

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

F1.6 The staff concluded that the information presented in this section of the SAR satisfied the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, Regulatory Guide 3.61, and accepted practices.

2.0 PRINCIPAL DESIGN CRITERIA

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

2.1 Structures, Systems, and Components Important to Safety

The SSCs important to safety are discussed in Section 2.5 of the SAR and summarized in Table 2.5-1 of the SAR. In this table, each component is assigned a safety classification. The SSCs important to safety include the 24PT1-DSC, the AHSM, and the OS-197 TC. The staff agrees with the determinations stated in Section 2.5 of the SAR.

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

2.2.1 Spent Fuel Specifications

Intact and/or damaged WE 14X14 SC and WE 14X14 MOX fuel may be stored in the 24PT1-DSC. This includes fuel assemblies utilizing boron coated fuel pellets, integral fuel burnable absober (IFBA) assemblies. In addition, RCCAs, TPAs, and NSAs may be stored with the fuel assemblies. Up to four damaged fuel assemblies may be stored in failed fuel cans in specific DSC locations.

The fuel to be stored in the 24PT1-DSC is limited to fuel with a maximum initial enrichment of 4.05 weight percent U-235. The maximum allowable burnup is given as a function of intial fuel enrichment but does not exceed 45 GWd/MTU. The minimum cooling time is ten years for SC fuel assemblies and twenty years for MOX fuel assemblies.

2.2.2 External Conditions

The Standardized Advanced NUHOMS® System SAR Section 2.2 includes a summary range, environmental conditions, natural phenomena, and manmade situations that the system has been designed to withstand. These include:

- tornado and wind loadings
- flooding
- seismic events
- snow and ice loadings
- tsunamis
- lighting
- fire and explosion
- cask drop

The staff determined that the descriptions provided sufficient detail to provide an overview of which conditions, phenomena, and situations required consideration during the evaluation of this SER. Further evaluation of these and other normal, off-normal, and accident conditions are discussed in Sections 3 through 11 of this SER.

2.3 Design Criteria for Safety Protection Systems

The safety protection systems, a summary of design criteria for the Standardized Advanced NUHOMS® System, are described in Section 2.3 of the SAR.

2.3.1 General

The Standardized Advanced NUHOMS® System was designed to provide spent fuel storage for at least 40 years. The internal pressure of the NUHOMS®-24PT1 DSC is always above atmospheric pressure during the storage period to protect against in-leakage of air that could damage the fuel. The welded confinement boundary is verified to be leak tight after loading to ensure the gas cannot escape.

2.3.2 Structural

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

2.3.3 Thermal

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

2.3.4 Shielding/Confinement/Radiation Protection

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

2.3.5 Criticality

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

2.3.6 Operating Procedures

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

2.3.7 Acceptance Tests and Maintenance

The acceptance test and maintenance program for the Standardized Advanced NUHOMS® System are described in Section 9 of the SAR, including the commitments, industry standards, and regulatory requirements used to establish the acceptance, maintenance, and periodic surveillance tests.

2.3.8 Decommissioning

Decommissioning considerations are for the Standardized Advanced NUHOMS® System evaluated Section 14 of the SAR.

2.4 Evaluation Findings

The staff concluded that the principal design criteria for the Standardized Advanced NUHOMS® System are acceptable with regard to meeting the regulatory requirements of 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices. A more detailed evaluation of design criteria and an assessment of compliance with those criteria is presented in Sections 3 through 14 of the SER.

3.0 STRUCTURAL EVALUATION

This section presents the results of the structural design review of the amendment request for the Advanced Standardized NUHOMS® System to the CoC and the SAR submitted under 10 CFR Part 72, Subpart L[1]. The review was conducted to assess the safety analysis of the structural design features, the structural design criteria, and the structural analysis methodology to evaluate the expected structural performance capabilities under normal operations, off-normal operations, accident conditions and natural phenomena events for those structures, systems and components (SSCs) important to safety. The on-site transfer system used with the Advanced Standardized NUHOMS® System OS 197 transfer cask (TC), which was previously reviewed and approved by the staff with the NUHOMS®-24P DSC CoC No. 1004. The evaluation only considers the transfer cask system to the extent that it has not been previously evaluated and approved for use with the Advanced Standardized NUHOMS® System. The compatibility of the Advanced Standardized NUHOMS® System for use with the transfer casks is included in the evaluation. Included in the Standardized Advanced NUHOMS® System is the Advanced Horizontal Storage Module (AHSM) and the contained dry shielded canister, 24PT1-DSC.

The Standardized Advanced NUHOMS® System was designed to accommodate the use of the NUHOMS® System in locations where there are high seismic requirements and increased needs for shielding. The Standardized Advanced NUHOMS® System, while intended for use in high seismic regions, allows free-standing casks that are allowed to slide and tip some distance on the concrete pad/basemat in response to a high g-level seismic event. Section 11.2.2 provides additional information relative to the evaluation of this high seismic characteristic of the Standardized Advanced NUHOMS® System.

The review was conducted against the appropriate regulations as described in 10 CFR 72.236 that identify the specific requirements for spent fuel storage cask approval and fabrication. The unique characteristics of the spent fuel to be stored are identified as required by 10 CFR 72.236(a) so that the design basis and design criteria that must be provided for the SSCs important to safety can be assessed under the requirements of 10 CFR 72.236(b). The structural evaluation of the structures, systems and components important to safety must also consider and be compatible with the other specific applicable requirements of 10 CFR 72.236 for maintaining the spent fuel in a subcritical condition, providing adequate radiation shielding and confinement, providing redundant sealing of the confinement system, providing adequate passive heat removal, providing wet or dry transfer capabilities, providing for ease of decontamination, providing for a minimum design life of 40 years, providing for testing and other appropriate means to demonstrate acceptable performance under the design conditions. The structural systems are also evaluated to determine if the DSC is compatible, to the extent possible, for handling and retrievability of the stored spent fuel. The evaluation also addresses whether or not the design, fabrication and testing are conducted under a quality assurance program meeting 10 CFR Part 72, Subpart G, as required by 10 CFR 72.234.

3.1 Structural Design of the Standardized Advanced NUHOMS® System

The Standardized Advanced NUHOMS® System is made up of the welded, dry shielded canister, 24PT1-DSC, that is placed inside an AHSM. The dry shielded canister used in the Standardized Advanced NUHOMS® system is a modification of the FO-DSC (for Fuel Only) associated with the Ranch Seco independent spent fuel storage installation (SNM 2510, Docket 72-11), which allows storage of intact and damaged fuel assemblies, along with control components in a single DSC. The canister is fabricated as a high-integrity stainless steel , all-welded, cylindrical vessel that provides for the dry storage of spent fuel assemblies in an inert atmosphere of helium. This welded vessel provides the system's confinement barrier for the spent fuel assemblies and other permitted contents. The canister assembly also includes the internal structures that provide support to the canister's contents and provides a portion of the shielding. The 24PT1-DSC is designed to remain intact under all operating conditions and defined accident conditions identified in Chapter 11 of the SAR without losing its function to provide confinement of the spent fuel assemblies.

The AHSM is primarily a massive reinforced concrete enclosure structure referred to as a storage module that provides environmental protection and the major shielding protection for the 24PT1-DSC and its contents. The internal structural steel components provide support for the canister assembly. Only a single canister is placed in a single storage module so that for a typical installation several modules are utilized and are grouped together in various arrays to suit the specific site configuration and operations scheme regarding spent fuel assemblies and other permitted contents. The arrays may be arranged in single or double rows and are placed so as to be in direct contact with one another. To meet the seismic design conditions for the Standardized Advanced NUHOMS® system, it is necessary to utilize a minimum of three AHSMs that are structurally linked together to form a continuous structural array that will remain stable under the postulated design earthquakes.

The significant loads for structural considerations are the seismic loads and the cask drop loads. Seismic loads are important because this system is specifically designed to include high seismic requirements; the cask drop loads are significant since they control the component design in most cases.

3.1.1 Structural Design Features

3.1.1.1 Dry Storage Canister: 24PT1-DSC

The 24PT1-DSC canister assembly is made up of several steel structural components and the related weld filler metal, all of which are considered to be important to safety. The individual structural components are as follows: for the canister shell - shell cylinder, shield plugs, shell end cover plates; for the canister internals or basket assembly - support rods, spacer discs and the guide sleeves.

The 24PT1-DSC stainless steel pressure and confinement boundary consists of the cylindrical shell, the inner bottom cover plate, the inner and outer top cover plates, and the associated welds. It is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Code[2]. All pressure boundary components are constructed of Type 316 stainless steel. The circumferential and longitudinal shell plate welds are full penetration butt welds which are radiographed and inspected according to NB-5000 of the ASME Boiler and Pressure Vessel Code. Specific exceptions to the ASME Code have been identified and documented in SAR Table 3.1-14. In addition, the welded joints between the top inner and outer cover plates and the cylindrical shell are being designed and fabricated in accordance with ASME Code Case N-595-1 by having the root and final passes of the partial penetration welds examined by PT. The non-confining components of the 24PT1-DSC include the shield plugs of A-36 steel, the outer bottom cover plate, the grapple ring assembly, the support ring, and the lifting lugs. The grapple ring assembly is welded to the outer bottom cover plate to serve as the connection location for the insertion/extraction operations associated with the AHSM. The support ring is welded to the cylindrical shell and supports the top shield plug. Four lifting lugs are welded to the inside of the cylindrical shell and the support ring and are utilized to lift the unloaded DSC into the transfer cask prior to the loading of spent fuel. All non-pressure boundary components welded to the pressure boundary are also stainless steel

The internal basket assembly provides structural support for and geometric separation of the spent fuel assemblies. The assembly consists of 24 stainless steel guide sleeve assemblies, 26 carbon steel spacer discs and four support rod spacer sleeve assemblies. In the vertical position, the basket assembly components do not carry the fuel weight since the fuel weight is transferred to the inner bottom cover plate. In this position the spacer discs are held in place by the support rods and spacer sleeves. The support rods extend over the full axial dimension of the canister cavity and are provided with allowances for the axial thermal growth of the rods. In the horizontal position, the guide sleeves provide full support for the fuel and transfer the load to the spacer discs, which then distribute the fuel weight to the 24PT1-DSC shell.

3.1.1.2 Advanced Horizontal Storage Module (AHSM)

Each AHSM provides a self-contained modular reinforced concrete structure for physical protection and significant shielding for each DSC during storage and provides the internal structural steel support system for the 24PT1-DSC. Both of these components are important to safety. The thick concrete roof with the top shield block and walls provide significant shielding and there are provisions for the attachment of supplemental shield wall sections (at the back and at row ends) that can be used depending upon the configuration of the spent fuel storage facility and the need to minimize doses during loading and retrieval operations. The top shield block is tied to the base AHSM module by eight steel rods in the vertical direction and interlocking concrete keys in the horizontal directions. Adjacent AHSMs are connected to each other with module-to-module ties located at the top and bottom of the AHSMs. The top ties are integrated into the roof portion of the AHSMs and consist of reinforced concrete tie beams with reinforcing steel between adjacent modules mechanically connected. The bottom ties consist of steel rods connecting the bases of adjacent modules. A system of horizontal and vertical keys between adjacent modules restrain relative horizontal and vertical movement between individual AHSMs. The internal structural steel support system for the canister consists of a pair of stainless steel support rails for the 24PT1-DSC, rail stiffener plates, extension plates and cross-members. The AHSM is designed to provide the removal of spent fuel decay heat by a combination of radiation, conduction and convection cooling. This is accomplished by a ventilation inlet opening through the lower portion of the front wall and an exit ventilation opening through the top portion of the AHSM.

The AHSM is designed in accordance with ACI-349[3] and built to ACI-318[4] code. The ultimate strength method of analysis is utilized in the design and evaluation. The load combinations specified in ANSI 57.9-1984[5] and ASCE 7-95[6] are used for combining normal, off-normal and accident loads for the AHSM.

The AHSMs are supported on a load-bearing foundation system that will consist of a reinforced concrete basemat (3-foot minimum thickness) that is supported on a subgrade of adequate capacity to support all loading conditions. This is known as the ISFSI basemat and is considered as not important to safety. The design basis accelerations for the AHSM are 1.5g in two horizontal directions and 1.0g in the vertical direction acting simultaneously at the surface of the ISFSI basemat. When subjected to a design basis earthquake, the connected (3 minimum) AHSMs are free to slide on the reinforced concrete basemat.

The air inlet vent extends through the front block of the base AHSM and the block also provides inlet shielding. The AHSM concrete is thermally protected by a thin stainless steel heat shield that is inside the base storage module as a liner.

The AHSM shield door is located in front of the opening into the base module and is steel-backed concrete to provide missile protection and radiation shielding.

3.1.2 Structural Design Criteria

The structural design criteria for the Standardized Advanced NUHOMS® System components that are important to safety are derived to assure that the structural aspects of these structures, systems and components provide the following conditions for the functioning of the system: maintain conditions required to store spent fuel safely, prevent damage to the spent fuel container during handling and storage, and provide reasonable assurance that spent fuel can be received, handled, packaged, stored and retrieved without undue risk to the health and safety of the public.

3.1.2.1 Individual Loads

From the various conditions that are imposed on the SSCs important to safety arising under normal operations, off-normal operations and accident circumstances including natural phenomena events, there are loads or other environmental parameters that influence the performance of the SSCs important to safety. These individual loads, or the loads that arise from the phenomenon or condition form the basis for the structural design, analysis and component sizing including the detailed design. The Standardized Advanced NUHOMS® System includes the following elements in the criteria for the design of the system.

3.1.2.1.1 Dead Loads

The dead load is the weight of the structure and attachments, including permanently installed equipment.   The dead weight of the 24PT1-DSC includes the self-weight of the loaded DSC, including the basket assembly components, cover plates, control components and stored fuel. The weight of the fully loaded (dry condition) NUHOMS® 24PT1-DSC is 79,400 pounds (with failed fuel cans) with the design value taken as 82,000 pounds. These loads are considered for the design of the system in all of its possible orientations.

The weight of the AHSM (not including the empty or loaded 24PT1-DSC) is 318,300 pounds with the design value taken as 320,000 pounds. The loaded weight of the AHSM is 400,300 pounds.

3.1.2.1.2 Live Loads

The live loads considered for the design of the Advanced Standardized NUHOMS® System are the normal handling loads associated with lifting the cask, placing the cask in the transfer cask (TC), downending the cask in the TC to a horizontal orientation, moving the cask in the TC with the transport trailer, removal from the transport system, and hydraulic insertion into the AHSM or extraction from the AHSM. The transfer loads include the following loads: axial load of +/- 1.0g, transverse load of +/- 1.0g, vertical load of +/- 1.0g, and under a combined condition of all loads of +/- 0.5g in each of the three directions. The normal design insertion load into the AHSM acting axially on the 24PT1-DSC is 60,000 pounds and the extraction load is 60,000 pounds. The off-normal design loads for insertion is 80,000 pounds and extraction is 60,000 pounds acting axially on the 24PT1-DSC. The accident design load for unloading a 24PT1-DSC is 80,000 pounds. The incidental live load for the AHSM is taken at 200 psf.

3.1.2.1.3 Pressure Loads

The design internal pressure for normal conditions as presented in Table 3.1-6 of the SAR is 10 psig and for the off-normal conditions is 20 psig. The internal test pressure of 12 psig is applied without the NUHOMS®-24PT1-DSC top cover plate in place. The accident internal pressure is 60 psig. Table 3.1-6 of the SAR provides the maximum internal pressures during normal, off-normal, and accident conditions that were used in the design of the NUHOMS®-24PT1-DSC.

3.1.2.1.4 Thermal Loads

The thermal loading is based on the NUHOMS®-24PT1-DSC containing spent fuel rejecting 14 kW decay heat with the ambient air temperature range of -40°F to 120°F. In some cases the analysis was performed for heat loads up to 24 kW. The thermal evaluation of normal conditions, off-normal conditions and accident conditions are provided in Section 4 of the SAR with Tables 4.1-3, through 4.1-5 providing the calculated temperatures under these various loading conditions. These temperature extremes are expected to occur only for short periods of time, on the order of hours.

3.1.2.1.5 Flood Loads

Flood loading is addressed in Section 3.1.2.1.3.7 of thes SAR. The Advanced Standardized NUHOMS® System is designed for flood water to a depth of 50 feet with a water velocity of 15 feet per second. The 50 foot static head which equates to an external load of 22 psi, is applied uniformly to the exterior of the DSC. Since the AHSM rapidly fills with water through the ASHM vents, no evaluation of static head on the ASHM is needed, but the AHSM is evaluated for the impinging flow of 15 feet per second on the sides of a submerged AHSM.

3.1.2.1.6 Wind, Tornado and Tornado Missiles

Extreme wind load effects are much less severe than the design basis tornado wind forces, so that the tornado wind conditions will govern and are used in the design process. The Advanced Standardized NUHOMS® System is designed for the same tornado wind loads and tornado missiles as the Standardized NUHOMS® System. The Advanced Standardized NUHOMS® System is evaluated for a design basis tornado wind velocity of 360 mph, meaning a translational velocity of 70 mph, a rotational velocity of 290 mph and a pressure drop of 3 psig at the rate of 2 psi per second as discussed in Section 3.6.2.2.4 of the SAR. A spectrum of four tornado missiles are also listed in that Section of the SAR that were used in the design of the Standardized Advanced NUHOMS® System

3.1.2.1.7 Seismic

The design earthquake for the Advanced Standardized NUHOMS® System is based on an earthquake that produces accelerations in two horizontal directions of 1.5g and a vertical acceleration of 1.0g acting simultaneously. The location of these accelerations is taken at the top of the concrete pad/basemat that supports the AHSM. A specific site utilizing the Standardized Advanced NUHOMS® System will have to demonstrate that the design seismic conditions at the facility do not produce seismic effects greater than these at the top of the Standardized Advanced NUHOMS® System concrete pad/basmat. Since to provide for the high seismic capability of the Standardized Advanced NUHOMS® System it is necessary to link a minimum of three AHSMs together, the ties and or keys that are used for such a purpose must be analyzed. For this condition the AHSM top shield block response was analyzed for a 2.25g acceleration to allow for the effect of ungrouted shear keys and the subsequent amplification in response that could occur. Section 11.2.9 provides additional information on these ties and keys.

3.1.2.1.8 Snow and Ice

Snow and ice loads for the AHSM are derived from ASCE 7-95[6] and consist of the maximum 100 year roof snow load for an unheated structure in the continental United States of America of 110 psf. There are no credible snow and ice loads for the 24PT1-DSC or the TC.

3.1.2.1.9 Lightning

Lightning striking the AHSM, which encapsulates and protects the 24PT1-DSC, and causing off-normal conditions (events causing conditions more than approximately one-time per year) is not considered a credible event. Lightning is considered a credible accident event for the AHSM system as shown in Table 11.2-1 of the SAR; however, only the AHSM itself can be potentially affected since the 24PT1-DSC is contained within the concrete AHSM. Consequently, the 24PT1-DSC is essentially electrically isolated by the high impedance concrete of the AHSM and there are no electrical components required for the safety functions of the AHSM system that could be affected by induced lightning surges. The DSC is also essentially isolated from thermal and mechanical effects that can arise from exterior direct lightning strikes on the AHSM. Lightning protection system requirements are site specific and depend on the frequency of thunderstorms at an ISFSI site as well as the location and characteristics of the surrounding grounded structures and the protection they provide or the system they have in place. If a specific site is to utilize simple lightning protection equipment, such grounded equipment is considered miscellaneous attachments and can be added to the exterior of the AHSM.

3.1.2.1.10 Fire and Explosion

The Advanced Standardized NUHOMS® System contains no flammable material and the concrete and steel used for the system fabrication can withstand any credible fire hazard. No explosive materials are present in the fission products or cover gases. Externally initiated explosions are considered to be bounded by the design basis tornado generated loads and are evaluated in Section 11.2.2 of the SAR. In order to utilize the Advanced Standardized NUHOMS® System, licensees are required by 10 CFR Part 72, Subpart K, to confirm that no conditions exist near the ISFSI that would result in pressures due to off-site explosions which would exceed those postulated for tornado wind or missile effects.

3.1.2.1.11 Tsunami

For use at a specific site that can be influenced by tsunami, verification that the tsunami effects will not exceed any of the bounding load conditions will have to be established, or a site specific analysis will have to be performed.

3.1.2.1.12 Cask Drop

The cask drop conditions that are considered can arise when the 24PT1-DSC is positioned inside the transfer cask. The side drop is postulated at 75g and the corner drop at 30-degrees from the horizontal is taken at 25g.

3.1.2.2 Loading Combinations

The loading combinations for the Advanced Standardized NUHOMS® System are provided in Tables 3.1-5, 3.1-10, 3.1-11, 3.1-13, 3.6-1, 3.6-12, and 3.6-13 of the SAR. The loading combinations reflect the various operational conditions and events that may occur during the lifetime of the utilization of the Advanced Standardized NUHOMS® System and the design calculations reflect consideration for these loading combinations. The loading combinations include the following cases:

- Non-operational events including natural phenomena
- Fuel loading
- Draining/Drying
- Transfer Trailer Loading
- Transfer to/from ISFSI
- AHSM Loading
- AHSM Storage
- AHSM Unloading
- DSC Unloading/Reflooding

Tables 3.1-5 and 3.6-1 of the SAR provide load combinations for the 24PT1-DSC canister assembly and applicable ASME Service Levels for normal, off-normal, and postulated accident conditions, the normal operating loads are designed for the safety-related/important to safety components.

Tables 3.1-10 and 3.6-12 provide load combinations for the concrete structures that make up the AHSM

Tables 3.1-11 and 3.6-13 provide load combinations for the structural steel components of the AHSM.

Table 3.1-13 provides the load combinations that are considered when analyzing for sliding and overturning of the Standardized Advanced NUHOMS® System from the position on the ISFSI concrete pad/basemat.

3.1.3 Materials

The SAR provides information on the usage of materials and methods of fabrication for components of the Advanced NUHOMS® system. The components include the AHMS, the DSC and the transfer cask. In general, the materials and fabrication methods selected for the Advanced NUHOMS® system have already been approved for the system in use in the Rancho Seco plant's site-specific license under 10 CFR Part 72 (SNM-2510; Docket No. 72-11) for on-site storage of spent fuel.

The main shielding features of the two systems (the Rancho Seco licensed system and Advanced NUHOMS® system) are provided by a concrete horizontal storage module, a transfer cask, and the 24PT1-DSC. The shielding materials, lead and NS-3 materials, of the MP-187 Transfer Cask (previously approved for use at Rancho Seco) are the same as those in the OS-197 Transfer Cask of the Advanced NUHOMS® system. Shielding in both systems is provided by the concrete HMS, the DSC and the lead and NS-3 materials in the transfer cask, with the Advanced NUHOMS® system using the OS-197 Transfer Cask and the Rancho Seco using the MP-187 Transfer Cask. The Bisco NS-3 of the neutron shield material is used in shields for both systems. Lead is used in the gamma shield for both systems. Concrete provides shielding for the AHSMs and is used as the basemat for the ISFSI. The advanced module functions like and uses concrete materials that have increased strength when compared with the earlier approved design and this supports higher seismic loads.

Components of the DSC, which is a modification of the model FO-DSC approved previously, are fabricated of materials and in a manner that is similar to that previously approved for use at Rancho Seco. The welding processes, welders and welding materials used for the welding of the 24PT1-DSC meet the requirements of the appropriate ASME Section III subsections and Section IX. Non-Code welds meet the provisions of Section IX of the ASME Code or AWS D1.1 [3.38] or D1.6 [3.39]. Weld metal material properties meet the requirements of Section II of the ASME Code or associated AWS requirements (See further discussions in SER Section 7.1). The stainless steels in the 24PT1-DSC (modified from the FO-DSC) were chosen for increased corrosion resistance in marine environments. In addition, the current DSC has additional provisions for allowing the storage of selected assemblies and control components. Thus, the components of this 24PT1-DSC system use materials in ways previously approved.

The thermal neutron absorber material used in the DSC in these systems is sheet material fabricated from Boral,® a product containing a blend of aluminum 1100 alloy powders and coarser particles (average diameter of 85 micrometers) of B_4C. The blend is placed into a box that is formed from sheet material of aluminum 1100 alloy. This box is then formed into plate or sheet material. The fixed neutron absorbers serve as a neutron absorber for criticality control and as a heat conduction path. The DSC's safety analysis does not rely upon the mechanical strength of the absorber sheet material, as the basket structural components surround the plates on all sides. The Boral® sheet material borders the guide sleeve of the fuel assembly and is held in place by a stainless steel oversleeve wrapper. As was done in the previous criticality evaluation, for the Advanced NUHOMS® system, credit is taken for only 76 percent of the boron shown to be present in the neutron absorber panels. Thus, this usage of the Boral® as a thermal neutron absorber in the DSC is the same as or consistent with previously approved conditions in the earlier design.

3.2 Normal and Off-Normal Conditions

3.2.1 Analysis Methods

The 24PT1- DSC assembly was analyzed using the finite element method of the ANSYS software package. The three-dimensional models were developed by creating four separate models, two each of the top and bottom half-length of the canister shell, and utilizing the symmetry of the shell so that only a half-section or quadrant-section was idealized for the models. This is an acceptable modeling technique because of the known stress conditions for general shell membrane structures under some loadings such as internal pressure based on classical unique numerical solutions. With this information it is possible to verify the acceptability of the model in representing the shell sections by comparing the results with classical theory. Once the model has been verified by such comparisons, the computer model can be used in more difficult regions of a structure to analyze the stresses. Boundary conditions can be imposed to duplicate the prototype behavior. Figures 3.6-1 and 3.6-2 of the Standardized Advanced NUHOMS® System SAR present the two quadrant models used for the analyses. The models were three-dimensional models that included the details of the cylindrical shell, the shield plug, the closure plates, grapple ring for the bottom model, and support ring for the top model. It should be noted that the A-36 shield plugs are not specifically analyzed since they are free to expand thermally and serve only as a mass for shielding or carry very minor compressive stresses.

The 24PT1- DSC basket assembly was broken down into individual components for analysis. The separate components included the guide sleeve assemblies, the spacer discs, and the support rod/spacer sleeve assemblies. The guide sleeve assemblies while made up of several individual elements, the guide sleeve tubes are the only element of the assembly relied on for structural capacity. The analyses were performed using a combination of closed-form calculations and finite element analyses using an ANSYS model. Elastic analyses were used for loading combinations for normal and off-normal conditions. The guide sleeves are unrestrained radially and axially so no significant thermal stresses are produced. The analyses of the spacer discs were performed using three-dimensional finite element models with the ANSYS software package. In order to adequately represent the physical conditions under the various types of loading, three separate finite element models were developed for analysis using ANSYS. A model was developed for a quadrant, a semicircle and the full disc. The full disc three-dimensional model also included the transfer cask rails and the inner liner of the transfer cask since the in-plane loads from the spacer disc are transferred to those parts of the Standardized Advanced NUHOMS® System. The models used for these analyses are provided in Figures 3.6-4 through 3.6-6 of the SAR. The analyses of the support rod/spacer sleeve assemblies were completed using a combination of ANSYS finite element analyses and closed-form calculations considering them as linear component supports. The support rods were evaluated using a simple beam model with the ANSYS software with the model including the support rods as well as the spacer sleeves. This allowed the transfer of axial force and moment from each spacer disc to the assembly.

The AHSM was analyzed using a linear elastic three-dimensional finite element model with the ANSYS software package for the analyses. For conditions of physical loading, the model included the interaction of a three-dimensional finite element model of the 24PT1-DSC, the support rails and the cross-members that were analyzed by separate linear elastic finite element analyses for various loadings. Since the AHSM is made up of several different concrete components, the ANSYS coupling capability was utilized to model the connectivity. For thermal loading conditions, the AHSM finite element included only the concrete components since the structural steel components and concrete elements such as the door to the AHSM are free to grow thermally and therefore do not induce thermal stresses into the concrete structure. Models used in the analyses of the AHSM are provided in SAR Figures 3.6-7 through 3.6-13.

3.2.2 Loading Cases Analyzed

The normal and off-normal operating load conditions analyzed for the 24PT1-DSC canister shell included the dead weight loads, design basis live loads, design internal pressure, design external pressure, design basis thermal loads, loading operation pressure load, operational handling loads that include normal transfer as well as off-normal transfer loads. A total of 39 loading combinations for normal and off-normal conditions were analyzed. These are listed in Tables 3.1-5 and 3.6-1 of the SAR. These included the non-operational load cases for testing the pressure vessel during the fabrication process, the load cases for spent fuel loading into the canister during operations, the draining and drying load cases during operations, the load cases arising from the transfer of the loaded canister to the transport trailer, loading cases arising from the transfer to and from the ISFSI site, the loading of the 24PT1-DSC into the AHSM, the loading cases arising from the storage in the ASHM, and the cases arising from the unloading of the 24PT1-DSC from the AHSM.

The normal and off-normal loading conditions analyzed for the basket assembly of the 24PT1-DSC that included the individual basket components consisted of the same matrix of loading combinations used for the 24PT1-DSC canister shell for normal and off-normal conditions. Except for the basket assembly, there is no significant loading resulting from the pressure loads so that the calculations for the 39 loading combinations were somewhat simplified.

The normal and off-normal loading conditions analyzed for the AHSM included not only the loads identified as being included in the loading combinations for the 24PT1-DSC canister, but also those environmental loads that impact the exterior of the AHSM such as wind loads. The individual loads for the AHSM are listed in the SAR in Table 3.6-10 and the loading combinations are provided in Tables 3.1-10, 3.1-11, 3.6-12, and 3.6-13.

3.2.3 Analysis Results

For the 24PT1-DSC canister shell the analysis results have been shown to meet the appropriate material stress allowables under the various service levels identified by Subsection NB-Class I Components, of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Rules for Construction of Nuclear Facility Components. This means that the confinement/pressure boundary of the 24PT1-DSC canister for spent fuel storage meets essentially the same pressure vessel criteria used in the design and construction of the reactor pressure vessel of an actively operating nuclear power plant. Within the context of the Code regarding compliance with respect to the analyses, one type of evaluation parameter is the value of material stress created under the various design conditions that is predicted on the basis of the analytical calculations. For the 24PT1-DSC canister the relevant computed stresses have been summarized in the SAR in tabular form along with the Code allowable values. The Code allowable values are Service Levels A and B for the normal and off-normal load conditions. As is the standard Code procedures, these stresses are reported and compared against allowable values for the temperatures under which the material will be performing as a structural material. The allowed stress value is prescribed by the Code depending upon several variables such as the type of stress (primary membrane, membrane plus bending or primary plus secondary), the loading combination, and the service level that has been specified which has taken into account the margin to material failure. It should be noted that the SAR summary for the 24PT1-DSC uses the terminology “stress ratio” to quantify the closeness of the calculated stress to the allowable and that the stress allowable may be based on the Code value for either stress intensity, the yield strength, or the ultimate strength. Table 3.6-4 of the SAR provides a summary of calculated stresses and stress ratios for the components of 24PT1-DSC canister for the various stress types. The table also identifies the controlling loading combination for each component and each stress type. The worst case under normal and off-normal load conditions (the stress ratio closest to or greater than 1.0) is for the outer bottom cover plate under membrane plus bending stress that occurs under the conditions of off-normal unloading in high outdoor temperatures. The stress ratio is 0.97 with the material still below the material yield point. It should be noted that this component is not part of the 24PT1-DSC confinement/ pressure boundary. The next worst case is for the canister shell also under primary membrane stress and under the same loading conditions. The stress ratio is 0.92. The descending stress ratios are 0.91, 0.87, 0.86, and then 0.80 or less.

For the 24PT1-DSC basket assembly components the analysis results are also summarized in the SAR in a manner similar to the 24PT1-DSC shell. The design criteria for the all safety significant components of the basket assembly are based on the appropriate material stress allowables under the various service levels identified by Subsection NG-Core Support Structures of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Rules for the Construction of Nuclear Facility Components, except the design criteria for the support rods and spacer sleeves are based on Subsection NF-Supports of the same ASME Code. The guide sleeves assembly analyses resulted in a maximum stress ratio of 0.09 under the normal and off-normal loading conditions, occurring under handling conditions for the primary membrane plus bending stress. The values are summarized in Table 3.6-8 of the SAR. The summary for the spacer discs maximum stress ratios is provided as Table 3.6-7 in the SAR. The maximum value was 0.45 for the primary membrane and bending stress arising from the transfer loading operation to or from the ISFSI concrete pad/basemat. The summary of the stress ratios for the support rods are provided in Table 3.6-9 of the SAR with the maximum value being 0.90, indicating calculated stresses less than the allowable. The stress checks for the spacer sleeves were performed using the interaction equations of the ASME Code in NF-3322.1(e)(1) and found to meet the allowables.

The results of analyses for the AHSM reinforced concrete components while reported in the form of a ratio, the ratio is not a stress ratio as used for those components designed under the ASME Code comparing a computed stress to an allowable stress which may or may not alone identify the margin to failure, but is a ratio of the computed internal forces and moments using the ultimate strength design method, with the appropriate load factors and capacity reduction factors to the computed strength capacity of the concrete component. However, in both ways of presenting the analysis results, the design criteria are met only if the ratio is 1.0 or less. The strength capacities of the various reinforced concrete components were computed in accordance with the requirements of ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures"[3]. Table 3.6-15 provides the summary of the analysis results for the reinforced concrete components of the AHSM identifying the various load combinations and directional maximum shears and moments along with the computed capacities and ratios. Under the normal and off-normal conditions the maximum value of the ratio was 0.97 for shear in the front slab of the base section. The next highest value for the ratio is 0.87, again for shear forces, but in the slab of the storage section. All normal and off-normal load combination results demonstrate that no reinforced concrete section capacities are exceeded for the design. The structural steel within the AHSM was designed and evaluated using the allowable stress design method of the AISC Manual[7]. Tables 3.6-19 and 3.6-20 of the SAR provide the summary results of the analyses for these structural steel components as stress ratios that represent the ratio of actual computed stress to the allowable stress. The highest values of the ratio for the normal and off-normal conditions for the rail was the shear stress at 0.74 and the rail cross member compression stress at 0.79

3.3 Accident Conditions

Included in the consideration of accident conditions that are man-made are the effects of the various natural phenomena events that create accident conditions that were discussed under Section 3.1.2.1, Individual Loads.

3.3.1 Analysis Methods

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

The finite element analysis methods described herein in Section 3.2.1 were also utilized in the analysis of accident conditions, in fact many of the same idealized models were used for evaluating the accident conditions, with the only difference being the loading conditions imposed. In some instances a different type of structural analysis was performed on the element or component than was used under the normal and off-normal conditions. For example, where stress allowables are permitted to exceed the range of elastic behavior, it is incorrect to use a linear elastic method of analysis. For such a situation under accident conditions the analysis method that would be more appropriate would be an elastic-plastic analysis method that reflects the structural behavior under a higher stress level with generally increased deformations. All methods of structural analysis used for evaluation of the Standardized Advanced NUHOMS® System are accepted methods and have been previously used for similar analyses.

For loadings under the accidental condition, the 24PT1-DSC cask drop load corresponded to g-loads being applied as static loads. For the frequency evaluation of the steel canister shell, that portion of the seismic analysis was completed using a closed-form calculation based on varying boundary assumptions. Using the response spectra for the vertical and horizontal directions, the resulting internal forces were determined and the steel canister was analyzed using linear elastic analysis. The seismic stability analysis for the 24PT1-DSC was performed as a non-linear stability analysis as discussed herein in Section 11.0.

For accident conditions the basket assembly components were analyzed as discussed below. The guide sleeve assemblies were analyzed using the ANSYS finite element model used in the normal and off-normal loading analyses as well as closed-form calculations. The spacer discs under accident conditions were analyzed using the same three ANSYS finite element models used under the normal and off-normal loading cases. For the accident loading cases such as the side drop analysis, an elastic-plastic stress analysis was performed. The technique utilized a plastic modulus of 5% of the elastic modulus. Additionally, a stability analysis was performed using an eigenvalue buckling analysis. The support rod/spacer sleeve assemblies were analyzed under accident loading conditions as described in Section 3.2.1 for the normal and off-normal conditions.

The AHSM under accident conditions was analyzed by the same method as described in Section 3.2.1 for the normal and off-normal conditions.

3.3.2 Loading Cases Analyzed

Section 11.2 addresses the accident conditions which also, in this document, encompasses the loading conditions resulting from natural phenomena. The following loading cases have been addressed.

a. Blockage of AHSM air inlet and outlet openings
b. Inadvertent loading of newly discharged fuel assembly
c. Accidental transfer cask drop
d. Pressurization due to fuel cladding failure within the DSC
e. Design basis flood
f. Tornado winds and tornado generated missiles
g. Design basis seismic event
h. Fire and explosion
i. Lightning
j. Burial

Table 3.1-5 of the SAR identifies the loading combinations for the accident conditions with the specific analysis load cases identified and related to Table 3.6-1 that identifies all of the load components that constitute each of the loading combinations for the 24PT1-DSC.

Table 3.1-10 of the SAR provides the accident loading combinations for the reinforced concrete AHSM components and Table 3.1-11 provides the accident loading combinations for the structural steel components of the AHSM.

The loading combinations used to analyze the Standardized Advanced NUHOMS® System for sliding and overturning under the tornado and flood accident conditions are provided in SAR Table 3.6-13.

3.3.3 Analysis Results

Tables 3.6-3, 3.6-5, and 3.6-6 of the SAR provide the summary results for the enveloping loading cases for the accident load conditions for all accident conditions and Service Levels C and D respectively for the 24PT1-DSC shell. The calculated stresses for the various shell components for each accident loading combination are provided in Table 3.6-3. Tables 3.6-5 and 3.6-6 list the stress type, the controlling (resulting in highest calculated stress) load combination and the calculated and allowable stresses. The allowable stresses are those defined in Subsection NB of the ASME Code as discussed in Section 3.2.3 herein. The "stress ratio" for these controlling load combinations are provided for each component and stress type in the same manner as described in Section 3.2.3. For those loading cases governed by the Code Service Level C, the most highly stressed component had a stress ratio of 0.97 for the shell's primary membrane stress and a stress ratio of 0.92 for the membrane plus bending stress. The next most highly stressed shell component is the outer bottom cover plate with stress ratios of 0.95 for membrane plus bending and 0.93 for the primary membrane stress. The only other stress ratio at 0.90 or greater was found for the weld between the outer bottom cover plate to the shell that was 0.90. For those loading cases governed by Service Level D, the most highly stressed component had a stress ratio of 0.81. This was for the outer top cover plate membrane plus bending stress. The next most highly stressed component had a stress ratio of 0.80 and was the weld between the inner top cover plate and the shell. All other stress ratios for Service Level D were less than 0.80.

For the 24PT1-DSC basket assembly the analysis results for the accident conditions are provided in Tables 3.6-7, 3.6-8 and 3.6-9 of the SAR. Stress allowables for the basket components are based on Subsection NG or Subsection NF of the ASME Code, as appropriate. For the spacer disc the maximum stress ratios occur as 1.00, the permitted limiting value, and represent the primary membrane plus bending stress condition for the 18.5-degree drop and the 45-degree drop. These are shown in SAR Table 3.6-7. The accidental loading producing the mostly highly stressed condition for the guide sleeve assembly is the end drop that produced a stress factor of 0.54 for the axial loading as shown in Table 3.6-8 of the SAR. Under the accident conditions the support rod assemblies had a maximum interaction ratio as defined in the ASME Code, Subsection NF 3322.1(e)(1) of 0.58 compared to the permitted value of 1.0.

Under accident conditions the analyses of the reinforced concrete portions of the AHSM produced results that were identified in the same manner as was done for the normal and off-normal loading conditions. The highest ratio of computed load to the actual capacity was 0.96 and occurred in the sidewalls of the AHSM storage unit for the shear loads. This condition occurs during the extremes of the thermal conditions and is identified in Table 3.6-15 of the SAR. The analysis results for the structural steel portions of the AHSM under the accident conditions are provided in Tables 3.6-19 and 3.6-20. For the rails the maximum value of the stress ratio was 0.98 for shear stress. For the extension plates and cross members the highest values were 0.99 for the plates and 0.99 for the cross members in compression.

The discussion of the results of the analyses of the AHSM keys and ties that must function during the accident conditions involving a seismic event are provided in Section 11.2.2.

3.4 Evaluation Findings

F3.1 The SSCs important to safety are described for the Advanced Standardized NUHOMS® System in sufficient detail to enable an evaluation of their structural effectiveness and are designed to accommodate the combined loads of normal, off-normal, accident and natural phenomena events.

F3.2 The Advanced Standardized NUHOMS® System is designed to allow handling and retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage of the system that will prevent retrieval of the stored spent nuclear fuel.

F3.3 The cask is designed and fabricated so that the spent nuclear fuel is maintained in a subcritical condition under credible conditions. The configuration of the stored spent fuel is unchanged. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.4 The cask and its systems important to safety are evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F3.5 The staff concludes that the structural design of the Advanced Standardized NUHOMS® System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the Advanced Standardized NUHOMS® System will enable safe storage of spent nuclear fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable industry codes and standards, accepted practices and confirmatory analysis.

3.5 References

1. U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,” Title 10, Part 72; Subpart L, “Approval of Spent Fuel Storage Casks.”

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, 1992 including the 1994 Addenda.

3. American Concrete Institute, ACI 349-97, “Code Requirements for Nuclear Safety Related Concrete Structures.”

4. American Concrete Institute, ACI 318-89(92), ”Structural Concrete Building Code.”

5 American National Standards Institute/ American Nuclear Society, ANSI/ANS 57.9-1984, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).”

6. American Society of Civil Engineers, ASCE 7-95. “Minimum Design Loads for Buildings and Other Structures.”

7. American Institute of Steel Construction, AISC Manual of Steel Construction, 9th Edition

4.0 THERMAL EVALUATION

The Standardized Advanced NUHOMS® System for spent nuclear fuel includes the 24PT1-DSC, in which 24 PWR spent fuel assemblies may be stored, the OS-197 TC, which is used in transfer operations for the DSC, and the 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 at nuclear power plants in areas with high seismic activity. The objective of the thermal review is to ensure that the cask/storage module component and fuel material temperatures of the 24PT1-DSC and AHSM will remain within the allowable values or criteria for normal, off-normal, and accident conditions. This objective includes confirmation that the temperatures of the fuel cladding (fission product barrier) 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 design of the DSC, TC, and storage module has been evaluated using acceptable analytical methods.

4.1 Spent Fuel Cladding

The Standardized Advanced NUHOMS® System is designed to store 24 PWR Westinghouse (WE) 14x14 fuel assemblies, either mixed-oxide (MOX) fuel with Zircaloy cladding (WE 14x14 MOX Zirc), or UO_2 fuel with stainless steel cladding (WE 14x14 SS304). The applicant has analyzed these two fuel types for storage in the DSC. The maximum heat load for the DSC loaded with only WE 14x14 SS304 is 14 kW or 0.583 kW per assembly. The maximum heat load for the DSC loaded with a combination of the two fuel assembly types is 13.706 kW or 0.583 kW per assembly for stainless steel clad fuel and 0.294 kW per assembly for Zircaloy clad fuel. The 13.706 kW limit is based on the maximum heat from 23 WE 14x14 SS304 assemblies and one MOX assembly. While the applicant has analyzed several components of the Standardized Advanced NUHOMS® System for higher heat loads in the SAR, the staff's thermal review as stated in this SER, does not approve the use of any of the components of the Standardized Advanced NUHOMS® System for heat loads higher than 13.706 kW when loaded with MOX fuel and 14 kW when loaded with WE 14x14 SS304 fuel. Should the applicant request approval of heat loads higher than these for the storage of spent fuel in a NUHOMS® 24PT1-DSC, the staff will conduct an independent review of a separate submittal at that time.

The staff verified that the analyzed cladding temperatures for each fuel type proposed for storage are below temperatures which could cause cladding damage that would lead to gross rupture. For normal conditions of storage, the applicant calculated a limiting fuel cladding temperature of 604°F (318°C) for both MOX Zircaloy and SS304 clad fuel. For normal conditions the transfer mode maximum cladding temperature calculated for both fuel types is 658°F (348°C). This is based on a 14 kW heat load in conjunction with a DSC shell temperature based on a 24 kW transfer cask analysis. For the accident condition, the maximum calculated cladding temperature for both fuel types is 749°F (398°C). This was calculated for a 16 kW heat load blocked vent condition and 117°F (47°C) ambient. The maximum fuel clad temperature reported by the applicant in the SAR was 751°F (399°C) for the vacuum drying transient. This is based on a 14 kW heat load in the DSC. The limits given by the applicant for the MOX Zircaloy clad fuel are 1058°F (570°C) for transfer (short term) and 618°F (326°C) for storage (long term). The limits given by the applicant for SS304 clad fuel are 806°F (430°C) for transfer and 690°F (366°C) for storage.

The fuel cladding temperature limits used by the applicant are described in Section 3.5 of the SAR. The applicant determined the long term storage temperature limit for the MOX Zircaloy clad fuel using the methodology described in PNL-6189, "Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircaloy Clad Fuel Rods in Inert Gas,"[1] for Zircaloy clad fuel. The limit was determined graphically by plotting the midwall hoop stress equation in PNL-6189 on the Commercial Spent Fuel Management (CSFM) Program generic limit curves. The limit derived is based on a 50 year design life. This methodology is acceptable to the staff. For the short-term accident and loading/unloading operations, the applicant used the temperature limit from PNL-4835. The applicant states that the MOX fuel assemblies to be stored in the DSC will have less burnup than the rods tested in PNL-4835, "Technical Basis for Storage of Zircalloy-Clad Spent Fuel in Inert Gases,"[2] and therefore the short term limit of 1058°F (570°C) can be conservatively applied to the MOX assemblies. The staff agrees with this assertion and finds that the limit is acceptable for short-term conditions.

For stainless steel clad fuel, the applicant utilized EPRI TR-106440 "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage," to determine the long term fuel cladding temperature limit to prevent creep failure, which was determined to be the bounding failure mechanism. This limit was determined using a steady state shear strain rate for irradiated fuel rods from curves for shear stress vs. temperature for a 50 year time to failure from the EPRI report. For the short term temperature limit, the applicant based their temperature on data for the sensitization of stainless steel at elevated temperatures in a non-inerted environment provided in the EPRI report. The applicant states that the temperature limit derived from this data will be conservative, as the DSC is inerted with helium for storage and transfer. The staff agrees with this assertion and finds that the limit is acceptable for short-term conditions. A summary of the fuel cladding temperature limits and the cladding temperatures calculated by the applicant are presented in Table 4-1 below.

Table 4-1
Standardized Advanced NUHOMS® System
Calculated Fuel Cladding Temperatures and Limits

Fuel Type	Normal Condition Maximum Storage Temperature (°F)	Normal Condition Maximum Transfer Temperature (°F)	Accident Condition Maximum Temperature (°F)	Maximum Temperature for All Conditions (°F)	Temperature Limit (°F)
WE 14x14 MOX Zirc	604[1]	658	749	751[2]	618 1058 (*short-term*)
WE 14X14 SS304	604[1]	658	749	751[2]	690 806 (*short-term*)
1. Maximum temperature reported for 16kW heat load and 70°F long term ambient condition 2. Maximum reported cladding temperature for vacuum drying transient					

4.2 Cask System Thermal Design

4.2.1 Design Criteria

The design criteria for the Standardized Advanced NUHOMS® 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 both the normal storage conditions for the 20-year approval period and postulated accidents that last a short time, such as a fire.

Section 4.1 of the SAR defines several primary thermal design criteria for the Advanced NUHOMS® storage system:

1. Pressures within the 24PT1-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 short-term allowable cladding temperatures that are applicable to off-normal and accident conditions of storage are based on PNL-4835, and EPRI TR-106440.

4. The allowable fuel cladding temperatures to prevent cladding degradation during long-term dry storage conditions are provided in Section 3.5 of the SAR.

5. Thermal stresses for the ASHM, 24PT1-DSC, and transfer cask, when combined with other loads, will be maintained at acceptable levels to ensure confinement integrity of the Standardized Advanced NUHOMS® System.

The staff concludes 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 Standardized Advanced NUHOMS® System with the following features:

1. The AHSM cools the 24PT1-DSC by buoyancy driven air flow through an opening at the base of the module, which allows ambient air to be drawn into the AHSM to cool the DSC. Heated air exits through a vent in the top of the shield block, creating a stack effect, which will ensure a cooling air flow across the DSC.

2. The 24PT1-DSC contains spacer disks, support rods and guide sleeve assemblies. Heat transfer through the basket structure in the radial direction is achieved by conduction through the spacer disk plates and guide sleeve assemblies as well as radiation from these components.

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

4. A metal heat shield is placed around the upper half 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 Standardized Advanced NUHOMS® System are passive. Drawings in Section 1.5 of the SAR along with the summary of material properties in SAR Section 4.2, provide sufficient detail for the staff to perform an in-depth evaluation of the thermal performance of the entire package as required by 10 CFR 72.24(c)(3)[3].

4.3 Thermal Load Specifications

The design-basis fuel for the Standardized Advanced NUHOMS® System is WE 14x14 PWR fuel and is described in Section 2.1 and Tables 2.1-1, 2.1-2, and 2.1-3 of the SAR. Each 24PT1-DSC can accommodate a maximum of four damaged WE 14x14 stainless steel clad assemblies, or one damaged MOX assembly, with all remaining assemblies being intact stainless steel assemblies.

The Advanced NUHOMS® Storage System is designed to dissipate 14 kW, which includes a bounding decay heat for the fuel of 0.581 kW/assembly and a bounding rod cluster control assembly (RCCA) decay heat of 0.002 kW/assembly. This makes a total decay heat limit of 0.583 kW/assembly. Due to temperature concerns for the zircalloy clad MOX assemblies, these assemblies are limited to a decay heat of 0.294 kW/assembly, which includes the RCCA decay heat. The axial profiles for the design-basis fuels are in SAR Section 4.4.1. The applicant used a peaking factor of 1.08 times the average power for the WE 14x14 assemblies. Maximum fuel assembly heat load is given in TS 12.2.1.c. By review and confirmation using independent analysis, the staff has reasonable assurance design-basis decay heats were determined properly.

4.3.1 Storage Conditions

The applicant provided a thermal analysis of the Advanced NUHOMS® Storage System to demonstrate that the system will perform satisfactorily for normal storage, off normal, and design-basis natural phenomena conditions. Table 4-2 provides the temperature and insolation conditions that the applicant applied in the thermal analysis.

Table 4-2
Standardized Advanced NUHOMS® System
Ambient Temperatures and Insolation Considerations

Condition	Temperature (°F)	Solar Insolation (Btu/hr-ft^2)
Normal	0 to 104	0 to 72.6
Long Term Average	70	72.6
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 Advanced NUHOMS® storage system are described in Section 4.4 of the SAR. The normal storage condition considers a maximum average daily temperature of 104°F (40°C) and includes solar insolation of 72.6 Btu/hr-ft^2, as specified by the applicant. The staff had concerns that this value for solar insolation was incorrect, due to the fact that it was well below the value recommended in NUREG-1536 for insolation upon a flat surface. The staff determined that the value used by the applicant was derived from a misapplication of the information provided in the ASHRAE 1981 Fundamentals Handbook. The applicant provided a revised analysis, using the value 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 revised analysis showed a small rise in the maximum concrete temperature for the ASHM. While the results of the applicant's analysis show very little change in concrete temperature, the applicant's use of ASHRAE values is incorrect and will not be accepted by the staff in future submittals.

The minimum normal storage condition considers a 0°F (-17.8°C) 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 Advanced NUHOMS® Storage 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 average daily temperatures are included in TS Section 12.4.4.3 as siting parameters that must be evaluated by the storage system user.

4.3.3 Off-Normal Conditions

Off-normal conditions for the Advanced NUHOMS® Storage System are described in Section 4.5 of the SAR. Included in these conditions is a maximum temperature of 117°F (47.2°C) and a minimum temperature of -40°F (-40°C). Also included is a solar insolation of 123 BTU/hr-ft^2 which is applied to the AHSM roof surface. This value is based on the solar insolation values suggested in 10 CFR Part 71, averaged over a 24 hour day.

4.3.4 Accident Conditions - Blocked Vents

Four accident conditions are postulated by the applicant, and are described in Section 4.6 of the SAR. The first accident evaluated by the applicant for the Standardized Advanced NUHOMS® System is a complete blockage of the AHSM ventilation inlet and outlet openings for a 40 hour period. The AHSM and the DSC are evaluated for the ambient temperatures and insolation values outlined in Table 4-1 above for the accident condition.

4.3.5 Accident Conditions - 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-197 transfer cask, as well as a loss of the required sunshade during transfer operations at the extreme off-normal ambient temperature condition of 117°F (47.3°C). 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 in Section 4.3.4.

4.3.6 Accident Conditions - Fire

The third accident condition postulated by the applicant is a fire that occurs during transfer of the DSC to the AHSM. A 15 minute fire with an average flame temperature of 1475°F(800°C), an average convective heat transfer coefficient of 5.21E-4 Btu/min-in^2-°F, 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 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. Based on review, the staff concludes that the thermal loads for the fire accident are acceptable. The applicant once again states that the results of the fire accident analysis are bounded by the blocked vent transient described in Section 4.3.4.

4.3.7 Accident Conditions - Flood

The final 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 the other accident conditions.

4.3.8 Cask Heatup During Loading

The applicant's description of the effects of loading and unloading conditions on the Advanced NUHOMS® storage system is provided in Section 4.7 of the SAR. Two loading conditions were evaluated by the applicant including heat up of the DSC after removal from the spent fuel pool and prior to blowdown and backfill, and an analysis of heatup of the DSC during vacuum drying (before helium backfill is completed). The analysis of vacuum drying included both a steady state and a transient analysis of the vacuum drying process. The applicant also analyzed the unloading condition of a reflood of the DSC.

For heatup of water in the DSC prior to blowdown and vacuum drying, the DSC is evaluated for two cases: initial DSC temperatures of 140°F (60°C) and 100°F (38°C), and fuel building ambient temperatures of 120°F (49°C) and 100°F (38°C), respectively. A maximum allowable DSC heat load of 14 kW was assumed. The heatup analysis 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 air. The applicant completed both a steady-state analysis as well as a transient analysis. An initial DSC shell temperature of 230°F (110°C), and a maximum allowable DSC heat load of 14 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 heat loads from 12 to 14 kW.

Finally, the applicant 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 4.7 of the SAR to be adequate.

4.4 Model Specification

4.4.1 Configuration

The applicant developed thermal models of the AHSM, the 24PT1- DSC fuel basket and fuel assemblies, and the 24PT1-DSC in the transfer cask using the finite difference computer code HEATING7. Descriptions of these models are provided in the SAR, Sections 4.4.2.2, 4.4.4, and 4.4.3, respectively.

The staff reviewed the applicant's use of the HEATING7 computer code and the associated inputs, assumptions, material properties, boundary conditions, and initial conditions. The staff has reasonable assurance that the temperatures of the cask components and the cask pressures under normal and accident conditions were determined correctly. Due to the confirmatory analysis performed by the staff, a concern was raised over the applicant's approach to modeling the fuel assemblies in the 24PT1-DSC. A discussion of the staff's findings is presented in Section 4.5.4 of this SER. Details of the applicant's modeling assumptions and approach follow.

4.4.1.1 AHSM Model

This model is described in SAR section 4.4.2.2. The model represented the symmetric right half of the AHSM and 24PT1-DSC cross section. The geometry of the DSC and associated heat shield within the AHSM are approximated, due to limits with the HEATING7 computer code. 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 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.

An additional model was constructed by the applicant to calculate maximum concrete temperature gradients. This model considered a free standing AHSM with all sides exposed to prevailing ambient conditions.

The DSC is approximated as a rectangle within the AHSM. The surface area of the rectangle is made to match the surface area of the actual DSC. Heat generation (for 14, 16, and 24 kWs) is distributed over the entire internal cavity volume for the DSC. The applicant does not utilize an axial peaking factor for the DSC model in the AHSM, and states that test data presented in reference 4.21 of the SAR, indicates that heat flux is nearly uniform over the surface of a cylindrical cask, and therefore the temperature profile over the cask surface is relatively flat.

4.4.1.2 24PT1- DSC Basket Section/Fuel Assembly Model

This model is described in SAR Section 4.4.4. A worst case, three dimensional slice of the 24PT1-DSC basket assembly and fuel cross sections is modeled in detail. Two spacer disks 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 constant temperature that was determined from the AHSM model. The fuel region within the DSC is modeled as a heat source 1.08 times the maximum allowed decay heat of 0.583 kW/assembly for an overall heat load of 15.1 kW.

This model includes details of the fuel assembly regions, to aid in determining the maximum fuel cladding temperatures for various conditions. Gaps within the model were maximized in order to present an accurate representation of an as-manufactured fuel basket. Fuel effective conductivity is based on test data, as discussed in SAR Section 4.2.k, and, therefore, includes the effects of radiation, conduction and convection, as present in the test configuration.

4.4.1.3 24PT1-DSC in Transfer Cask Model

This model is described in SAR Section 4.4.3. The model developed by the applicant to simulate the DSC within the OS-197 transfer cask 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 a previous SAR. The maximum DSC shell temperature is extracted from this model and used in the DSC basket analysis.

4.4.2 Material Properties

The material properties used in the thermal analysis of the storage cask system are listed in SAR Section 4.2, sub-sections (a) to (l). The applicant provided a summary of the material compositions and thermal properties for all components used in the cask model. 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 in SER Section 4.4.1 to analyze the behavior of the Advanced NUHOMS® storage system 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 temperature was 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 for all analyzed conditions are summarized in SER Table 4-2.

4.4.3.1 Storage/Transfer Conditions

For storage and transfer, normal and off-normal conditions, the applicant included boundary conditions for ambient temperatures and insolation as described in SER Sections 4.3.2 and

4.3.3, respectively. The ambient conditions were applied to the models described above to obtain normal/transfer and off-normal condition maximum temperatures.

4.4.3.2 Accident Conditions - Blocked Vent

For the postulated fire 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 SER Table 4-2.

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

For the postulated loss of the liquid neutron shield and sunshade for the transfer cask, during the transfer operation, the model described in SAR Section 4.4.3 was used with a heat load of 16 kW to determine the maximum DSC shell temperature. This temperature was then used with the 24PT1-DSC basket model described in SAR Section 4.4.4 to determine the basket component temperatures for a heat load of 14 kW. Extreme off-normal summer ambient conditions were utilized for this analysis.

4.4.3.4 Accident Conditions-Fire

For the postulated fire accident conditions, the model of the DSC in the transfer cask described in SAR Section 4.4.3 is used, and the boundary conditions for the fire accident described in SER Section 4.3.6, was applied to the outer surface of the transfer cask. 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.

4.4.3.5 Cask Heatup Analysis

For the cask heatup analysis, the DSC basket/fuel assembly model described in SAR Section 4.4.4 was used. In this case, all gaseous heat conduction within the DSC cavity is through air instead of helium. Radiation and conduction heat transfer between the basket and the DSC wall was included. Convection was neglected. The boundary conditions include the initial DSC shell temperature of 230°F (110°C) for the steady state vacuum drying analysis. For the transient vacuum drying analysis, the temperature of the basket was set to the saturation temperature of water. The DSC in the transfer cask model was used for the heatup of water in the DSC. The temperatures of the components in the DSC are considered to be at the maximum spent fuel pool temperature at the beginning of the analysis.

4.5 Thermal Analysis

4.5.1 Computer Programs

The thermal analysis was performed using the HEATING7 finite difference modeling package. HEATING 7 is capable of general 3-D steady-state and transient calculations. The models developed were reproduced in SAR Figures 4.4-2 thru 4.4-8, and discussed in SAR Section 4.4.

4.5.2 Temperature Calculations

4.5.2.1 Storage Conditions

The Standardized Advanced NUHOMS® System has been analyzed to determine the temperature distribution under long-term storage conditions that envelop normal, off-normal, and design basis natural phenomena (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 allowable temperatures of the components important to safety are discussed in Section 4.1 of the SAR. Low temperature conditions were also considered.

The calculated fuel clad temperatures for both the stainless steel and Zircalloy-clad fuel assemblies are listed in SAR Table 4.1-3, 4.1-4, and 4.1-5, for Normal, Off-Normal, and Accident Conditions, respectively. Temperature criteria for the spent fuel cladding are discussed in Section 4.1 of the SER. The applicant's analysis of the fuel cladding temperatures for the maximum heat load of 13.706 kW for MOX fuel and 14 kW for WE 14x14 SS304 fuel showed that the fuel remain below their respective acceptable temperatures limits. (See SER Table 4-1).

SER Table 4-3, below, summarizes the temperatures of key components in the cask for various environmental conditions.

Table 4-3
Temperatures of Key Components in the Advanced NUHOMS® Storage System[1]

Component	Normal Storage Conditions		Transfer Condition	Normal Allowable Range (°F)	Accident Conditions	
	Maximum (°F)	Minimum[2] (°F)	Maximum (°F)		Maximum (°F)	Allowable Range (°F)
AHSM Concrete	219	0	N/A	0 to 300	392[3]	-40 to 350
AHSM Support Steel	351	0	N/A	0 to 2,600	615	-40 to 2,600
AHSM Heat Shield	258	0	N/A	0 to 2,600	542	-40 to 2,600
DSC Shell	399	0	439	0 to 800	646	-40 to 800
DSC Top Inner Cover Plate	296	0	337	0 to 800	424	-40 to 800
DSC Top Shield Plug	316	0	345	0 to 700	444	-40 to 700
DSC Bottom Inner Cover Plate	315	0	402	0 to 800	450	-40 to 800
DSC Spacer Disk	617	0	658	0 to 700	695	-40 to 700
DSC Support Rod/Spacer Sleeve	479	0	522	0 to 650	588	-40 to 650
DSC Boral Sheet	618	0	658	0 to 850	696	-40 to 1000

Notes:
1. Temperatures are based on heat loads as listed in SAR Table 4.1-6
2. Assuming no credit for decay heat and a daily average ambient temperature of 0°F
3. Applicant will conduct testing on concrete samples to demonstrate acceptable concrete performance

4.5.2.2 Accident Conditions- Blocked Vents

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. The staff agrees with this assumption. Extreme ambient conditions as described in SAR Table 4.1-1 were used as boundary conditions for this analysis. The analysis included a heat source of 24 kW for qualification of the AHSM concrete, and 14 kW for qualification of the 24PT1-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 SER Table 4-3. Based on this analysis, the staff has reasonable assurance that the fuel cladding integrity and the confinement boundary will not be compromised during the blocked vent transient.

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

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 24PT1-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.

4.5.2.4 Accident Conditions - Fire

The applicant analyzed a fire accident on the Advanced NUHOMS® Storage System, specifically the DSC in the transfer cask model, using the conditions specified earlier in Section 4.3.6 of the SER. 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 15.1 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.

4.5.2.5 Cask Heatup Analysis

The applicant utilized the DSC basket/fuel assembly model to determine maximum temperatures during vacuum drying of the DSC (before helium backfill) and for the heatup of water in the DSC prior to blowdown.

For the steady state vacuum drying analysis, temperatures of DSC components and the fuel cladding were calculated for the steady state vacuum condition with air as the fill gas. The DSC shell temperature was set to 230°F (110°C) at the beginning of the analysis. The results indicated that the Spacer Disk within the DSC exceeded its ASME code allowable temperature of 700°F (371°C). In addition, the highest fuel cladding temperature reported in the applicant's analysis, 751°F (400°C), was reported in the steady state vacuum drying analysis.

The vacuum drying transient analysis was conducted using the DSC basket/fuel assembly model. The fuel basket was set to the saturation temperature of water at the start of the analysis. Table 4.7-2 in the SAR presents the results of the analysis in the form of time limits for the vacuum drying process. These limits must be followed to avoid exceeding the service temperature limits of any of the DSC components.

The analysis of heatup of water in the DSC was completed using the DSC in the transfer cask model, with modifications made as described in SAR Section 4.7.3, to remove credit for axial heat transfer. The 24PT1-DSC is given homogenized effective thermal properties, based on weight, volume, and material of the DSC components. All DSC components are assumed to be at the maximum spent fuel pool temperature at the beginning of the analysis. The applicant provided the results of the analysis in Table 4.7-3 in the SAR. This table provides time-to-boil values for heat loads between 10 and 14 kW.

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 Standardized Advanced NUHOMS® System to ensure that DSC component temperatures are not exceeded during cask loading evolutions.

4.5.3 Pressure Analysis

4.5.3.1 Storage/Off Normal/Accident Conditions

In SAR Section 4.4.8, the applicant evaluated internal pressurization for normal conditions. The applicant assumed a fully loaded DSC (24 assemblies) of WE 14x14 SS304 clad fuel, as well as four neutron source assemblies and 24 RCCAs. Four damaged fuel assemblies are also included in the analysis. A 1% failure of fuel rods and control components is assumed. For the ruptured rods, a 100 percent release of the rod fill gas and a 30 percent 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 9.8 psig, which is below the applicant's criteria of 10 psig for normal conditions.

In SAR Section 4.5.3, 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 12.2 psig.

In SAR Section 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 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 accident pressure calculated by the applicant was 42.7 psig.

The applicant reported the results of the pressure analysis and acceptance criteria in SAR Table 4.4-11.

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 4. The highest predicted pressure was 42.7 psig at a cavity gas temperature of 595°F (300°C) for the accident condition, which is below the DSC design pressure of 60 psig. Based on review of the applicant's pressure analysis, the staff concluded that 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.

4.5.3.2 Pressure during Unloading of Cask

Pressurization of the DSC is discussed in Section 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.

4.5.3.3 Pressure during Loading of Cask

The applicant discusses pressurization of the DSC during loading in Section 4.7.4 of the SAR. The applicant states that the pressure limit during DSC blowdown is 20 psig, which is well below the maximum design pressure limits of the DSC. This is discussed further in Section 3.1.2.1.3.2 of the SAR.

4.5.4 Confirmatory Analyses

The confirmatory analyses of the Advanced NUHOMS® Storage System SAR can be divided into six categories: (1) review of models used in the analyses, (2) review of material properties used in the analyses, (3) review of boundary conditions and assumptions, (4) perform independent, confirmatory analyses, (5) compare the results of the analyses with the applicant's design criteria, and (6) assure that the applicant's design criteria will satisfy the regulatory acceptance criteria and regulatory requirements.

The staff reviewed the approaches used by the applicant in the thermal analyses. The staff performed a confirmatory analysis of the thermal performance of the cask systems, structures, and components identified as important to safety. A detailed model of the Advanced NUHOMS® Storage System, and the 24PT1-DSC in particular, was developed using the COBRA-SFS computer code to evaluate the SAR results.

The staff conducted a confirmatory analysis of airflow in the AHSM by creating a two dimensional finite element model of the AHSM in the ANSYS FLOTRAN computational fluid dynamics (CFD) program. The staff confirmed the flow of air in the AHSM under normal conditions, for an ambient temperature of 100°F (38°C).

4.5.4.1 Analysis of 24PT1-DSC

A confirmatory analysis of the performance of the 24PT1-DSC, for heat loads of 13.706 kW, 14 kW and 16 kW was conducted by Pacific Northwest National Labs (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), and therefore, is a best-estimate code. The COBRA-SFS code utilizes detailed fuel assembly models to accurately predict maximum fuel cladding temperatures for different heat loads.

4.5.4.2 Description of COBRA-SFS Model of NUHOMS® 24PT1-DSC

The COBRA-SFS model of the NUHOMS® 24PT1-DSC is a detailed representation of the fuel assemblies, solid internal structures, and top and bottom end-caps of the DSC. The model represents a half-section of symmetry, with the line of symmetry passing vertically through the center of the horizontal DSC. The effect of gravity was neglected, in that it was assumed that the gaps between the spacer discs and the DSC shell, and between the assembly guide tubes and the spacer discs, remain uniform on all sides. This is a conservative assumption, in that it gives the greatest possible resistance to heat transfer across the gap. In reality, the gaps between these elements would be closed on the lower edge, forming an additional heat transfer path due to contact conductance.

The boundary condition for the calculations is specified by imposing a surface temperature on the outer shell of the DSC and end-caps. The internal fill gas was assumed to be helium. Radiation heat transfer within the fuel assemblies, between the fuel assemblies and guide sleeve walls, and within the 'basket' was included in the model, using 2-dimensional grey-body view factors calculated for the geometry of the fuel array in the guide sleeve and for the basket region.

The detailed geometry model of the spent fuel cask for COBRA-SFS consists of two main elements: (1) the fluid channels in the fuel assemblies (within the guide sleeves) and in the basket region, and (2) the solid structure nodes modeling the guide sleeves, spacer discs, DSC shell, and end-cap structures. The dimensions of all structures and their relative locations were obtained from information and drawings provided in the SAR, and in all cases, 'as designed' dimensions from the drawing were used to determine the physical distances, gaps, and locations of the nodes in the SFS model. (Note: additional calculations were also performed with adjusted gaps determined by considering thermal expansion of the components of the cask at operating temperature.) The following paragraphs provide a condensed description of the model.

The fluid channels containing the interior fill gas are represented in the model using the subchannel approach characteristic of the COBRA series of codes. Within the fuel assembly, the subchannel is the flow area defined by the grid of the rod array, with a single unit subchannel defined by any four adjacent rods. Sensitivity studies of the degree of detail in the subchannel model required to obtain an accurate resolution of the radial temperature profile across the fuel assembly have shown that it is not necessary to model each unit subchannel (and consequently each individual rod) in the assembly. Adequate resolution can be obtained by 'lumping' subchannels and rods in 'rings' that subdivide the radial gradient in the assembly.

These 'rings' of lumped channels and rods are further subdivided azimuthally, to capture the effect of non-uniform temperatures on the wall of the guide sleeve seen by the different faces of the square fuel array. Figure 4.1 illustrates the channel and rod noding used to represent the fuel assemblies included in COBRA-SFS model. This representation uses 56 channels and 70 rods to model each fuel assembly.

The subchannel modeling approach was also used in the basket region, to represent the flow area around and between the guide sleeves containing the assemblies. Figure 4.2 (a) shows a diagram of the channel layout for the basket region. In this region, the channels are defined by the flow paths around the guide sleeves and between the guide sleeves and the cask wall. The effect of the spacer discs, which effectively shut off axial flow in these channels, is modeled in COBRA-SFS using very large axial loss coefficients at the locations of the spacer discs. These large losses, plus the horizontal orientation of the cask, effectively shut off axial convection, and there is no significant axial flow in any of the channels. The connections between channels allow heat transfer by conduction through the fluid, but this model did not include the option to consider natural convection in the lateral direction.

The solid structure nodes in COBRA-SFS, called "slab nodes," were used to model the guide sleeves, spacer discs, and cask shell. The guide sleeves were modeled with a separate node on each face of the rectangular structure, with connections for convective and radiative heat transfer with the gas in the channels facing a node, and with the solid structure nodes "seen" by a given node (i.e. fuel rods on the inside face and basket structures on the outside face).

The boral sheets were modeled with separate nodes on each face of a guide sleeve, as were the cover plates over the boral sheets. It was assumed that these structures were in perfect contact, with no gaps between the sheets. The thermal connections between the guide sleeve nodes or oversleeve nodes and the nodes representing the spacer discs include radiative heat transfer across the gap between the sleeve and the opening in the spacer discs through which the guide sleeve passes.

The spacer discs are represented by a network of connected slab nodes following a pattern similar to that of the channel layout for the basket region (as shown in Figure 4.2(b)). This was done in order to capture the radial temperature distribution in the spacer discs. The discontinuous nature of the discs in the axial direction is modeled in the code by specifying dimensions that give the appropriate heat transfer area in the radial direction for each axial node. (The axial noding is specified so that each axial node contains only one spacer disc.) In addition, axial conduction is neglected in the slab nodes modeling the spacer discs. This approach allows the COBRA-SFS calculation to consider radial conduction through the slab nodes, and also to determine radial heat transfer through the gas in the gaps between the spacer discs. The heat transfer connections between the spacer discs and the inner surface of the cask shell include the effect of radiative heat transfer across the gap between the two structures.

The optional plenum model is used to represent the narrow open regions between the cask internals and the top and bottom of the cask. This model contains nodes representing the complete structure of each end cap, including shield plugs, inner cover plates, and outer cover plates; however, the four support rods and the associated hardware holding the spacer discs in position were neglected. Leaving these items out is conservative, as it omits a path for axial heat transfer between the spacer discs. This is not an important heat transfer path, for two main reasons. First, the total area comprised is small, relative to the cask cross-sectional area. Second, the tie rods are isolated from the cask ends, and do not provide an easy heat transfer path to the cask edge.

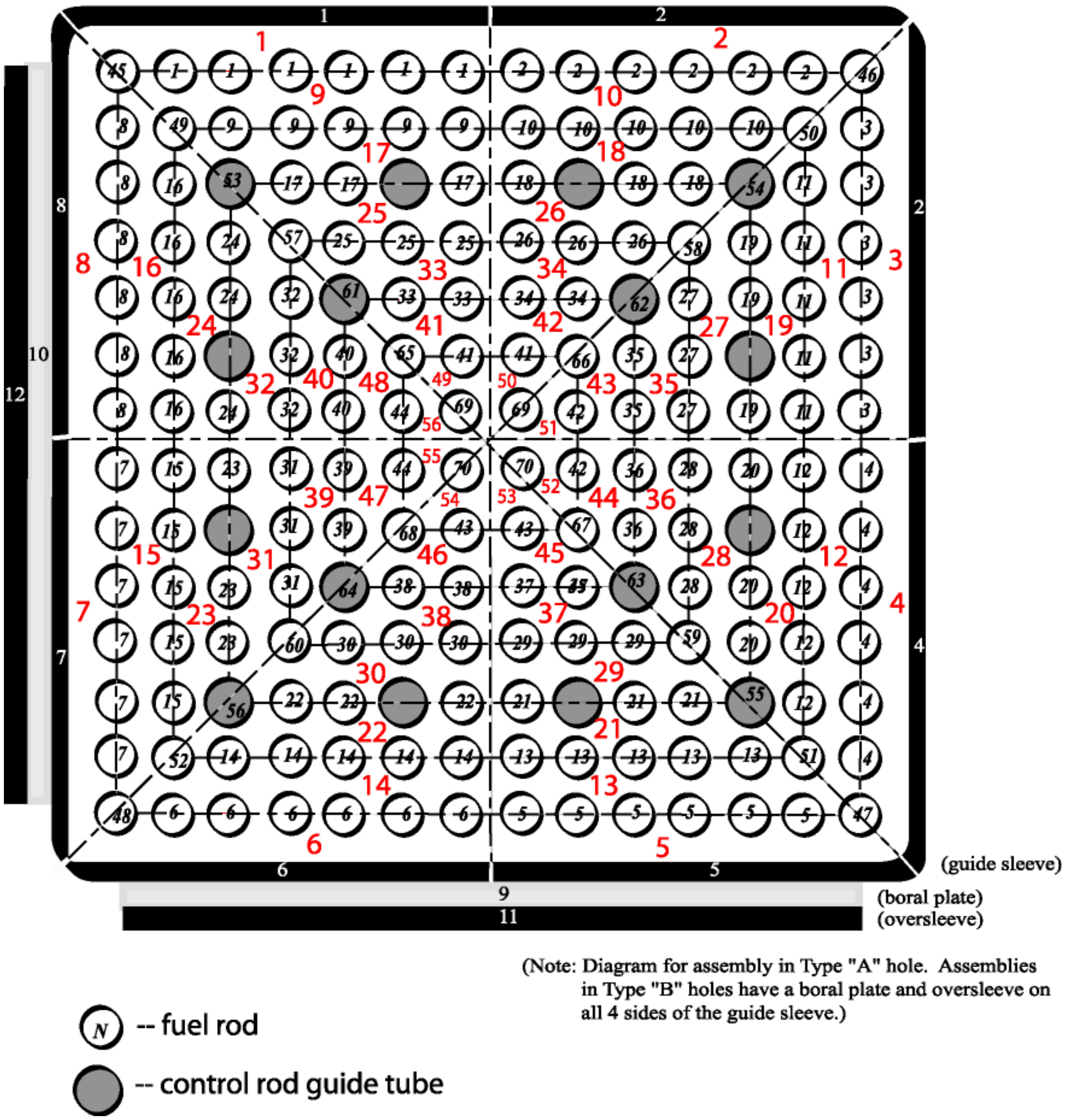


Figure 4.1 Example of COBRA-SFS noding (showing channels, rods, and slab nodes) for all fuel assemblies and guide sleeve structures in model of NUHOMS 24PT1-DSC

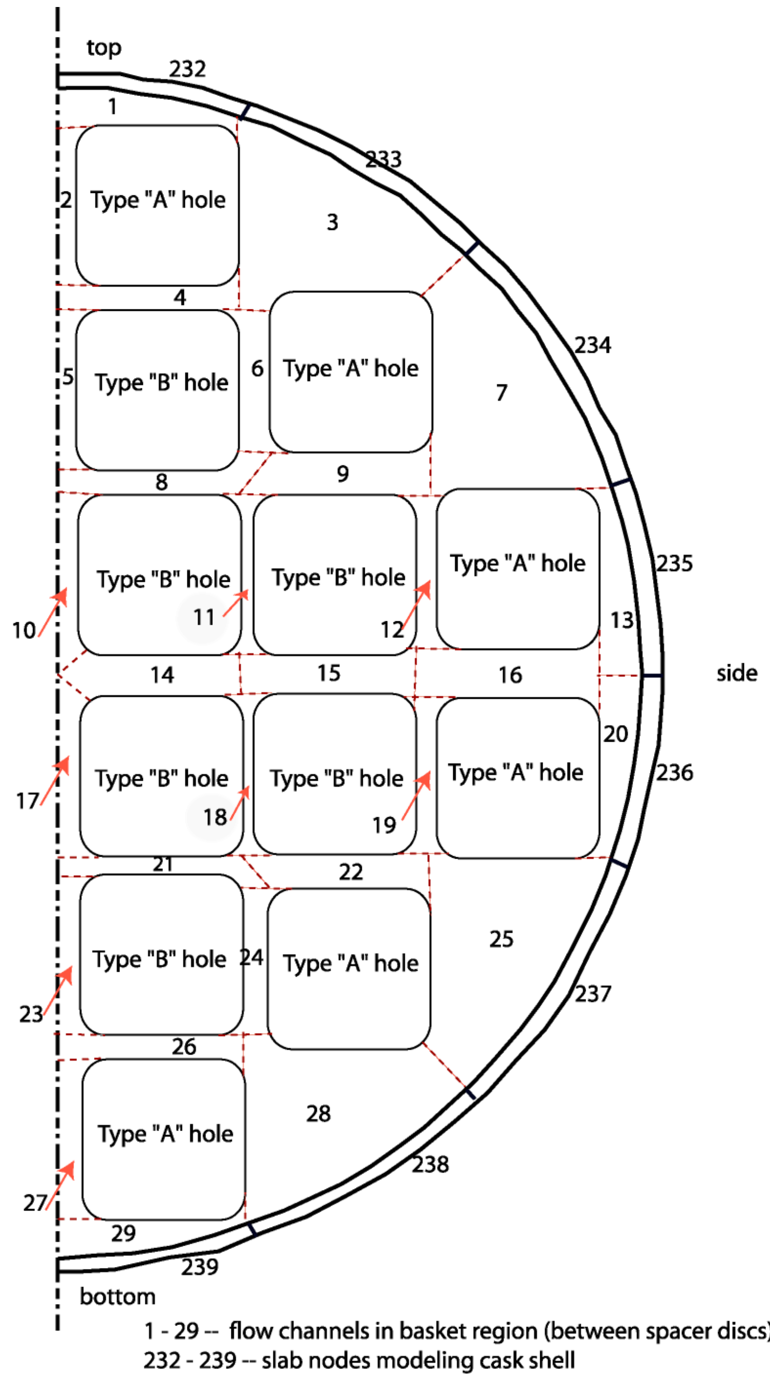


Figure 4.2(a) Diagram of COBRA-SFS model of flow channels in basket region between spacer discs in the NUHOMS 24PT1-DSC

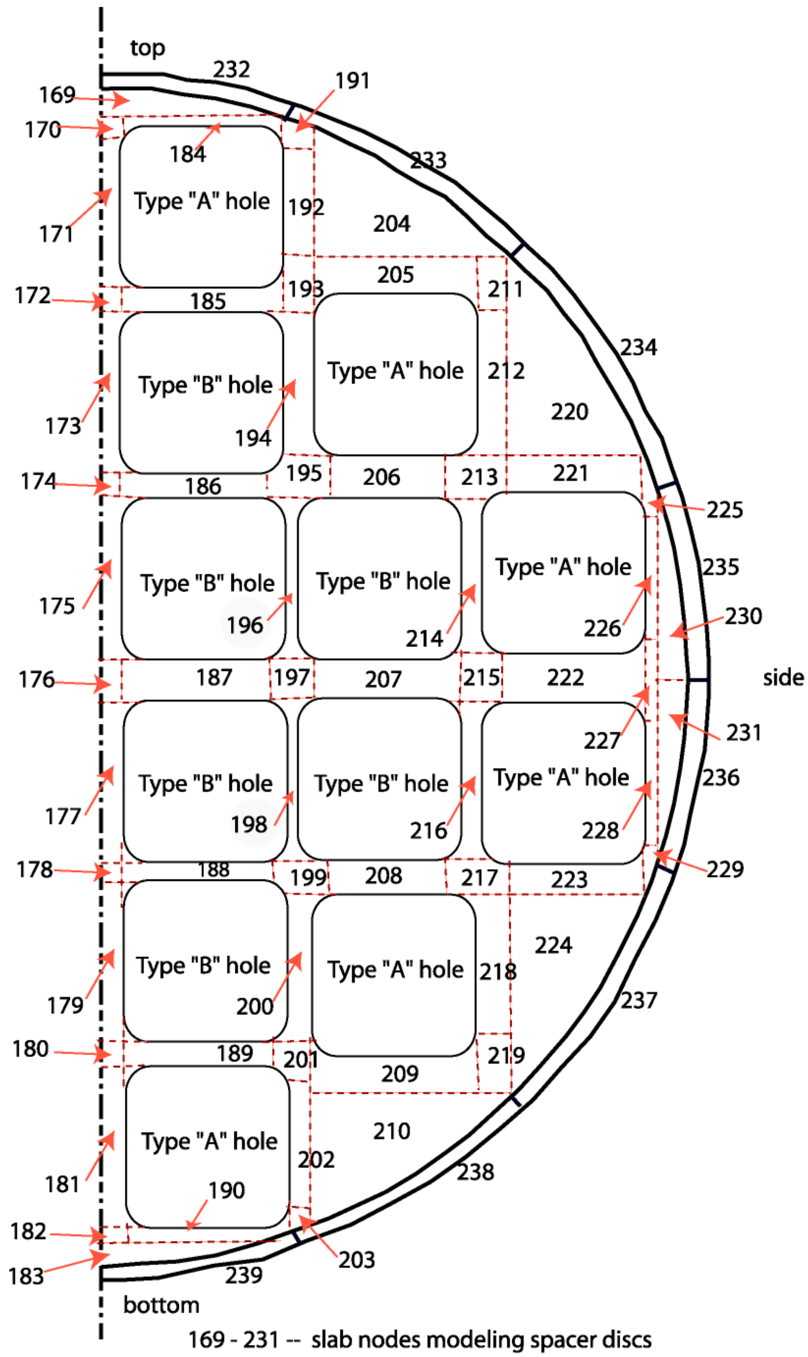


Figure 4.2(b) Diagram of COBRA-SFS slab nodes modeling spacer discs in the NUHOMS 24PT1-DSC

4.5.4.3 Results of COBRA-SFS Analysis of NUHOMS® 24PT1-DSC

The initial analysis conducted by PNNL showed maximum fuel cladding temperatures for MOX fuel assemblies higher for normal operation than those predicted by the applicant and reported in the SAR. Based on staff concerns, the applicant re-evaluated the 24PT1-DSC with 23 WE 14 x 14 SC fuel assemblies and one WE 14 x 14 Zircaloy clad MOX fuel assembly at a reduced heat load of 0.294 kW, in the hottest position. The reduced heat load was based on the characteristics of the actual Zircaloy MOX fuel assemblies that are currently in existence. A limiting total heat load of 13.706 kW was modeled by the applicant for the 24PT1-DSC. The applicant's analysis was based on a long term ambient temperature of 70°F and a varying DSC shell temperature based on a 16 kW heat load. The peak cladding temperature reported for this configuration by the applicant was 558°F (292°C), which is below the calculated limit of 618°F (325°C).

The analysis conducted by PNNL utilized the same 13.706 kW heat load and used DSC shell temperatures identical to the applicant's. The PNNL analysis showed a peak clad temperature of 593.6°F (312°C), which is below the temperature limit of 618°F (325°C) specified by the applicant for long term storage of Zircaloy clad MOX fuel; however, this temperature is significantly higher than the 558°F (292°C) maximum fuel clad temperature predicted by the applicant using the HEATING7 code. In addition, the staff concluded that, while its confirmatory analysis identified higher cladding temperatures for the WE 14 x 14 SC fuel, it was still within design limits for storage of spent fuel that did not exceed 14 kW per cask or 0.583 kW per assembly. The staff's analysis confirmed that loading a 24PT1-DSC with WE 14 x 14 SC fuel having a heat load of 16 kW met the temperature limits for long term storage.

4.5.4.4 Comparison of COBRA-SFS Results with SAR for NUHOMS® 24PT1-DSC Analysis

Upon comparison of the radial temperature plots of the hottest cross section of the 24PT1-DSC provided by the applicant in response to Request for Additional Information No. 2, and by PNNL, the staff noted a discrepancy in the temperature of the fuel assembly regions between the two models. While the analysis performed by PNNL indicates that the fuel cladding temperature is within the calculated fuel temperature limits, and is therefore, acceptable, the staff believes that the applicant's methodology for determining maximum fuel cladding temperatures, as presented in the SAR, is non-conservative for several reasons. First, the applicant's model uses a homogenized region for the fuel assembly with an effective thermal conductivity or "smeared" property approach in their thermal model. This method provides an average temperature for the fuel assembly and does not provide a precise peak cladding temperature for the hottest fuel rod within the assembly. The result reported by the applicant in the SAR is actually a maximum average temperature for the fuel assembly region. Second, the applicant provided a limited number of nodes per fuel assembly to record the fuel assembly temperature, which does not accurately capture the temperature gradient that exists across the fuel assembly, nor the temperature of individual fuel rods. This is in contrast to the PNNL model which accurately models individual fuel rod temperatures within the assembly and the PNNL model indicates the maximum temperature to be a specific rod within the hottest fuel assembly. Finally, the applicant's thermal code has not been validated against actual fuel temperature data, and therefore cannot be considered reliable for predicting fuel cladding temperatures given the current fuel parameters.

When approaching the temperature limits of fuel cladding, analysis tool must be validated against published data or benchmarked against another validated analysis code to ensure that the temperature predictions for the fuel clad are accurate. The staff has confidence in the analysis conducted by PNNL due to the fact that the COBRA-SFS code is a validated thermal code, and has been shown through comparison with available fuel temperature data to be a best-estimate code.

4.5.5 Conclusions

The staff approves the applicant's request for storage of Zircaloy clad MOX fuel, within the limits described in the Technical Specifications, as well as SS304 clad fuel in the Advanced NUHOMS® Storage System described in the SAR; however, the staff finds the current methodology used by the applicant for determining fuel cladding temperatures to be non-conservative. In future reviews, the applicant will be asked to validate the thermal code used for the thermal analysis against the data provided by the INEEL and PNNL studies of fuel temperatures in storage casks. This will ensure that the temperatures obtained in the analysis, especially related to fuel cladding, are accurate.

The staff has determined that the thermal SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness. Based on the applicant's analyses, there is reasonable assurance that the Advanced NUHOMS® Storage System is designed with a heat removal capability having testability and reliability consistent with its importance to safety. The staff further concludes, based on review and confirmatory analysis, that there is reasonable assurance that analysis of the Advanced NUHOMS® Storage System demonstrates that the applicable design and acceptance criteria have been satisfied for the storage of stainless steel and zircalloy clad fuel assemblies.

4.6 Evaluation Findings

F4.1 SSCs important to safety are described in sufficient detail in Sections 1 and 4 of the SAR to enable an evaluation of their effectiveness.

F4.2 The staff has reasonable assurance that the decay heat loads were determined appropriately and accurately reflect the burnup, cooling times, and initial enrichments specified.

F4.3 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.

F4.4 The staff has reasonable assurance that the Standardized Advanced NUHOMS® System is designed with a heat removal capability having testability and reliability consistent with its importance to safety.

F4.5 The staff has reasonable assurance that the Standardized Advanced NUHOMS® System provides adequate heat removal capacity without active cooling systems.

F4.6 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.

F4.7 The staff concludes that the thermal design of the Standardized Advanced NUHOMS® 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 Standardized Advanced NUHOMS® 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 References

1. Levy, I. S., et al., Pacific Northwest Laboratories, “Recommended Temperature Limits for Dry Storage of Spent Light-Water Zircaloy Clad Fuel Rods in Inert Gas,” PNL-6189, May 1987.

2. Johnson, A. B. and E.R. Gilbert, Pacific Northwest Laboratories, “Technical Basis for Storage of Zircalloy-Clad Spent Fuel in Inert Gases,” PNL-4835, September 1983.

3. U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear and High-level Radioactive Waste,” Title 10, Part 72.

4. U.S. Code of Federal Regulations, “Packaging and Transportation of Radioactive Material,” Title 10 Part 71.

6. Kane, W. F., "Spent Fuel Project Office Interim Staff Guidance 7," ISG-7. U. S. Nuclear Regulatory Commission, October 1998.

8. Michener, T. E., et. al., Pacific Northwest Laboratories, COBRA-SFS: “A Thermal-Hydraulic Analysis Code for Spent Fuel Storage and Transportation Casks,” PNL-10782, September 1995.

5.0 SHIELDING EVALUATION

The staff reviewed the capability of the Standardized Advanced NUHOMS® System to provide adequate protection against direct radiation and material release from the canister contents. The major components which provide shielding include: the AHSM, the OS-197 TC, and the 24PT1-DSC. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), and 10 CFR 72.236(d).[1,2]

This application was also reviewed to determine whether the Standardized Advanced NUHOMS® System components fulfill the acceptance criteria listed in Section 5 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."[3]

5.1 Shielding Design Features and Design Criteria

The Standardized Advanced NUHOMS® System is based on the previously licensed Standardized NUHOMS® System which includes a Horizontal Storage Module and the OS 197 TC. Significant design enhancements include thicker concrete, and increased resistance to high seismic and marine environments. The Standardized Advanced NUHOMS® System is designed to store up to 24 intact PWR Westinghouse 14x14 (WE 14x14) fuel assemblies or 20 intact fuel assemblies plus four damaged fuel assemblies in specially designed failed fuel cans.

5.1.1 Shielding Design Features

The 24PT1-DSC, when used with the Standardized Advanced NUHOMS® System provides both gamma and neutron shielding during loading/unloading, transfer, and storage operations. The 24PT1-DSC consists of a 0.625-inch thick steel canister with a 8.75-inch thick steel bottom shield plug, and a 10.25-inch thick steel top shield plug.

The OS 197 TC is the same as that used for the Standardized NUHOMS® System and is depicted in drawing NUH-03-8000-SAR, of the Standardized NUHOMS® System (Docket Number 72-1004). The TC consists of a 0.5-inch inner steel shell, a 3.56-inch lead shield, a 1.5-inch steel structural shell, a 3-inch water jacket neutron shield, and a 0.19-inch steel skin. The TC also has a top cover plate consisting of 2 inches of a neutron absorbing, NS3, and 3.25 inches of steel; and a bottom cover plate consisting of 2.25 inches of NS3 and 2.75 inches of steel.

The AHSM is constructed of thick concrete walls and a shielded access door. The side walls are 12 inches thick with a 36-inch side shield wall. The roof is 60 inches thick. The rear wall is 12 inches thick with a 36-inch rear shield wall, and the front door and front wall are 24 inches thick. The AHSM air inlet and outlet paths are designed to preclude radiation streaming.

The staff evaluated the Advanced NUHOMS® System shielding design features and found them to be acceptable. The applicant's analysis provides reasonable assurance that the shielding design of the 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 Design Criteria

The overall radiological protection design criteria are the regulatory dose requirements in 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), and maintaining occupational exposures as-low-as-reasonably-achievable (ALARA). The applicant analyzed the Standardized Advanced NUHOMS® System loaded with spent fuel as described in Section 2.1, 2.1.1, 5.2, and 10 of the SAR.

The SAR analyses provide reasonable assurance that the Advanced NUHOMS® System can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). While no specific dose rates limits are incorporated into the technical specifications; a radiation protection program is described in Section 12.5.2.4, which limits personnel doses to ALARA and which shows compliance with 10 CFR 72.104(a).

5.2 Contents and Source Specification

5.2.1 Contents

A detailed description of the payload for the Standardized Advanced NUHOMS® System can be found in Section 2.1 and 2.1.1 of the SAR. The payload consists of intact and/or damaged WE 14x14 assemblies including stainless steel or Zircaloy cladding, UO_2 or MOX fuel pellets, and with or without integral control components. These integral control components include: assemblies utilizing boron coated fuel pellets, Integral Fuel Burnable Absorber (IFBA) assemblies; and Non-Fuel Assembly Hardware (NFAH) such as Rod Cluster Control Assemblies (RCCAs), Neutron Source Assemblies (NSAs), and Thimble Plug Assemblies (TPAs).

Each 24PT1-DSC can accommodate a maximum of four damaged WE 14x14 stainless steel clad (SC) fuel assemblies with the balance intact WE 14x14 SC assemblies. One damaged MOX assembly may also be accommodated, with the remaining assemblies (up to 23) intact WE 14x14 SC assemblies. The damaged fuel may include assemblies with known or suspected cladding defects greater than hairline cracks or pinhole leaks, up to and including broken rods, portions of broken rods, and rods with missing sections. Damaged fuel assemblies shall be encapsulated in individual failed fuel cans that confine any loose material and gross fuel particles to a known subcritical volume during normal, off-normal, and accident conditions and facilitate handling and retrievability.

Tables 2.1-1 and 2.1-2 describe the characteristics, enrichment, burnup, and cooling times of the WE 14x14 stainless steel clad UO_2 and Zircaloy clad MOX fuel assemblies. The maximum allowable burnup and minimum cooling time for each type of assembly is presented below and was summarized from SAR Tables 2.1-2.

Parameter	**Stainless Steel Clad UO_2**	**Zircaloy Clad MOX**
Burnup (MWd/MTU)	45,000	25,000
Cooling Time (years)	10 (min)	20 (min)

Authorization for the storage of 24 Westinghouse 14x14 (WE 14x14) PWR fuel assemblies, as described in Sections 2.1 and 2.1.1 of the SAR, only is granted.

5.2.2 Source Specification

The source specification is presented in Section 5.2 of the SAR. The gamma and neutron source terms were calculated with the SAS2H (ORIGEN-S) module of SCALE 4.4 computer code. The SAS2H calculation results were modified to output the sources in the CASK-81 22 neutron, 18 gamma energy group coupled cross section library. As discussed in Section 5.2.2.2 of this SER, the fuel assemblies are divided into five regions to account differences in neutron flux that are present in the reactor.

Following the individual bounding gamma and neutron source term determinations, the source terms were combined in the shielding models to calculate the dose rates.

Although modeling analyses in the SAR were performed only for the design basis fuel and source terms, any other PWR fuel assembly which falls within the geometric, thermal, and nuclear limits established for the design basis fuel in Sections 2.1 and 5.2 of the SAR can be stored in the 24PT1-DSC.

5.2.2.1 Gamma Source

Tables 5.2-8 through 5.2-14 list the SAS2H calculated gamma source terms for the SC and MOX fuel assemblies, and for the different types of non-fuel assembly hardware (NFAH). The hardware activation analysis considered the cobalt impurities in the assembly hardware. The amounts of impurities considered in the analysis are presented in Table 5.2-18. Although cobalt impurities can vary, the applicant’s assumed values are reasonable and acceptable. The assemblies were irradiated to their maximum burnups, minimum cool times, and minimum initial enrichments present, as discussed in SAR Section 5.2. Based on these results, the WE 14x14 SC assembly irradiated to a burnup of 45,000 Mwd/MTU and cooled 10 years, together with the thimble plug assembly (TPA) type of NFAH, provide the design bases gamma source term. Table 5.2-15 shows this gamma source for 24 assemblies.

5.2.2.2 Neutron Source

The SAS2H calculated neutron source terms for the two fuel assemblies are provided in Table 5.2-16. Based on these results, the WE 14x14 SC assembly irradiated to a burnup of 45,000 MWd/MTU and cooled 10 years is again the design basis source term. Table 5.2-17 shows the design basis spectrum normalized to the Cm-244 spectrum. 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, top end fitting, and outside the top end fitting were multiplied by scaling factors of 0.2, 0.2, 0.1, and 0.01 respectively. These are the scaling factors recommended in Reference 4 of the SAR and are considered to provide bounding values.

5.2.2.3 Confirmatory Analyses

The staff reviewed the proposed contents and the assumed hardware cobalt impurities listed in Table 5.2-18 of the SAR. The staff has reasonable assurance that the design basis gamma and neutron source terms are acceptable for the Advanced NUHOMS® System shielding analysis. The staff also reviewed the flux scaling factors for the hardware source terms and found them to be appropriate.

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

The exterior dose rates for both the OS 197 TC and the AHSM are adequately controlled by the limits in the Certificate of Compliance for fuel specifications, maximum burnup, and minimum cooling time.

5.3 Shielding Model Specifications

The shielding analysis was performed with DORT, a 2-D discrete ordinates code used to calculate the dose rates on and around the AHSM and OS-197 TC. The staff considers this 2-D code acceptable for this application under the following limitations: the utilization of already proven technology in the Standardized NUHOMS® System, the thickness improvements in the concrete, the relatively low design basis source term, and the relatively long fuel cooling times of 10 and 20 years. Normally, the staff finds a 3-D shielding analysis needed, particularly for conditions where higher source terms may exist.

To determine the total off-site dose, the MCNP computer code was used. The off-site dose models include 1) a single AHSM, and 2) a 2x10 array of AHSMs loaded with design basis fuel in Standardized Advanced NUHOMS® System 24PT1-DSCs. The assumptions used in this calculation are listed in Section 10.2 of the SAR. The staff agrees that these assumptions are reasonable. Tables 10.2-1 through 10.2-5 list the resultant dose rates for the two models at varying distances from 6.1 meters to 500 meters.

5.3.1 Shielding and Source Configuration

The shielding source is divided into 20 axial regions as summarized in Table 5.4-1. The axial distribution of the gamma and neutron sources is assumed to follow the relative burnup profile depicted in Reference 5.7 of the SAR. A number of other simplifications and bounding assumptions are discussed in Section 5.4 of the SAR.

The analysis includes streaming paths through the AHSM air vents. The overall design minimizes potential streaming paths. Thus, although evaluation of streaming from narrow and long holes is difficult for DORT, streaming is not suspected to be a major problem for the Advanced NUHOMS® System. While DORT is subject to ray effects, this tends to over-predict radiation streaming.

5.3.2 Material Properties

The composition and densities of the materials used in the shielding analysis are presented in Tables 5.3-1 through 5.3-3. The homogenized fuel assembly region accounts for the uranium dioxide; stainless steel cladding and spacers; and steel and other materials present in the in-core region of the assembly and associated hardware.

The materials used in modeling the 24PT1 DSC, OS197 TC, and AHSM were reviewed and accepted by the staff. The material compositions and densities used were appropriate and provide reasonable assurance that the materials densities were adequately modeled for the 24PT1-DSC, OS-197 TC, and AHSM.

5.4 Shielding Analyses Results

5.4.1 Computer Programs

The applicant's shielding analysis was performed with DORT and is presented in Section 5.4 of the SAR. The cross section data used are based on the CASK-81 22 neutron, 18 gamma energy group coupled cross section library. The staff considers this 2-D code acceptable for this application under the following limitations: the utilization of already proven technology in the Standardized NUHOMS® System, the thickness improvements in the concrete, the relatively low design basis source term, and the relatively long fuel cooling times of 10 and 20 years. Normally, the staff finds a 3-D shielding analysis needed, particularly for conditions where higher source terms may exist.

5.4.2 Flux-to-Dose-Rate Conversion

The SAR uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose rate conversion factors to calculate dose rates. The values listed in this standard are summarized in Table 5.3-4.

5.4.3 Normal Conditions

Tables 5.1-2 through 5.1-5 of the SAR present calculated, normal-condition dose rates for the AHSM and Transfer Cask. Based on the conservative assumptions used during the analyses, the relatively low source term and long cooling time of the contents, and the administrative programs established in Sections 12.5.2.3 and 12.5.2.4 of the SAR; the staff has reasonable assurance that the user will be able to maintain normal-condition doses ALARA and meet the requirements of 10 CFR Part 72.

The expected dose rates for the AHSM are shown in Table 5.1-2 and are dominated by the gamma component. This is expected due to the thick concrete walls of the AHSM.

The expected dose rates for the transfer cask are shown in Tables 5.1-3 through 5.1-5. While the gamma component dominates the dose rates, there is still a significant contribution from neutron radiation. Table 5.1-4 also gives the surface peak dose rate at the top of the TC as approximately 697 mrem/hr. Exposure from localized peak dose rates may be mitigated by the actual locations of personnel and use of temporary shielding during loading/unloading operations. The doses for the expected occupational exposures are discussed in Section 10.3 of the SAR. Tables 5.1-3 through 5.1-5 of the SAR also present dose rates for the transfer cask at various distances, which show that the dose rates significantly decrease from peak locations to the edges of the top, bottom, and sides of the cask.

5.4.4 Off-Normal

Chapter 11 of the SAR does not identify any off-normal event that significantly degrades the components of the Standardized Advanced NUHOMS® System. For loading and unloading operations, the stresses on the 24PT1-DSC shell assembly components are demonstrated in Section 3 of the SAR to be within ASME Code Service Level B stress limits. Therefore, there is no permanent deformation of the shell. Thus, there is no potential for breach of the confinement pressure boundary or release of radioactive material. Similarly, the stress levels for the 24PT1-DSC during the extreme ambient conditions are demonstrated in Chapter 3 of the SAR to be within the ASME Code Service Level B stress limits. The AHSM also considers stresses due to ambient temperature, and meets the provisions of the ACI code.

5.4.5 Accident Conditions

For accident conditions no event is postulated that can impact the confinement boundary of the 24PT1-DSC while inside the AHSM. Accidents such as tornado missiles which may effect the shield walls of the AHSM do not effect safe operation; and recovery can be performed in a planned and safe manner. Temporary shielding during removal and replacement of AHSM walls is required or removal of the AHSM from service is required. Estimated recovery times and operations for accident situations must limit doses to As Low As Reasonably Achievable (ALARA) and meet the requirements of 10 CFR Part 72.

The bounding accident condition for the transfer cask considers loss of water from the transfer cask water jacket during a cask drop. This accident has been analyzed for the same OS-197 TC in CoC 1004 (Reference 11.16 of the SAR), for the Standardized NUHOMS® System. Because the normal condition dose rates for the transfer cask containing the 24PT1-DSC are bounded by the previously analyzed condition, the accident dose rates for the 24PT1-DSC will also be less than those analyzed in CoC 1004. Thus, they will be less than the reported value for the Standardized NUHOMS® System of 2128 mrem/hr with respect to gamma and neutron (Section 8.2.5.3.2 of 11.16). For an 8 hour recovery time, the estimated dose rate to a member of the public at 600 meters is approximately 0.04 mrem, which meets the regulatory requirements.

5.4.6 Inadvertent Loading of a Non-analyzed Fuel Assembly

Inadvertent loading of an assembly into the DSC that has not been authorized, or misloading an assembly into the wrong location in the DSC; has been determined by the staff to be a credible event. However, a review by the applicant of all of the Westinghouse 14x14 stainless steel clad UO_2 assemblies and 14x14 Zircaloy clad MOX assemblies fabricated to date has confirmed that the inventory of all fuel assemblies of this type will meet the fuel specification requirements of Section 12 of the SAR. In addition, the staff defines the risk resultant from a misloading is the probability times the consequence. Since the loading of a non-analyzed assembly (either not qualified for storage in the 24PT1-DSC, or in the wrong location inside the DSC) can not affect the confinement boundary integrity of the DSC, no release of radioactive material will occur and the consequence on the shielding characteristics of the system will be minimal.

5.4.7 Occupational Exposures

The analysis in the SAR used the design basis fuel to estimate occupational exposures for the Advanced NUHOMS® System. Section 10 of the SAR presents the estimated occupational exposures that are based on dose rate calculations in Section 5 of the SAR. The staff's evaluation of the occupational exposures is in Section 10 of this SER.

5.4.8 Off-site Dose Calculations

Section 10 of the SAR estimates the offsite dose rates from a single AHSM and a 2x10 array of AHSMs. Tables 10.2-1 through 10.2-5 present the calculated offsite annual doses for these arrays at distances of 6.1 to 500 meters based on 100% occupancy exposure time. These generic off-site calculations demonstrate that the Advanced NUHOMS® System is capable of meeting the offsite dose criteria of 10 CFR 72.104(a). Additionally, the removable contamination limits established in Section 12.5.2.4.c of the SAR of 2,200 dpm/100 cm^2 for β and γ emitting sources and 220 dpm/100 cm^2 for α emitting sources are based on the limits specified in 49 CFR 173.443 and provide reasonable assurance that the user will be able to maintain normal-condition doses ALARA and meet the requirements of 10 CFR Part 72. The applicant also performed a calculation that showed that an instantaneous release of all this contamination would cause a dose of less than 1 mrem 100 meters from the source.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by general licensees. The general licensee must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance. The actual doses to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel 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. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of the general licensee. A general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public as required, by evaluation and measurements.

Any general licensee using an engineered feature for radiological protection, which is considered important to safety (such as a berm) must evaluate the feature to determine the applicable Quality Assurance Category.

5.4.9 Confirmatory Calculations

The staff performed confirmatory analyses of selected dose rates using the SAS4 module of the SCALE system. The staff based its evaluation on the design features and specifications presented in the SAR. Limiting fuel characteristics and the burnup and cooling time are included in Section 12 of the SAR under Tables 12.2-1 through 12.2-3. The staff's calculated dose rates were in general agreement with the SAR values and were somewhat lower due to modeling assumptions used in SAS4, and slightly lower source terms, as described in Section 5.2.2.3 of this SER.

5.5 Evaluation Findings

F5.1 Chapters 2, 5 and 10 of the SAR sufficiently describe the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

F5.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F5.3 The Advanced NUHOMS® System is designed to provide redundant sealing of the confinement system.

F5.4 The staff concludes that the design of the radiation protection system of the Advanced NUHOMS® System, including the AHSM, 24PT1-DSC and the OS-197 TC, 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® 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 applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

5.6 References

1. U.S. Code of Federal Regulations, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Title 10, Part 72.

2. U.S. Code of Federal Regulations, "Standards for Protection Against Radiation," Title 10, Part 20.

3. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

6.0 CRITICALITY EVALUATION

The staff reviewed 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 Standardized Advanced NUHOMS® System meets the following regulatory requirements: 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g).[1] The SAR was also reviewed 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:[2]

1. 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.

2. 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.

3. 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.

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

    a. burnup of the fuel
    b. fuel-related burnable neutron absorbers
    3. 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 upon to prevent criticality are the fuel basket's geometry and permanent neutron-absorbing Boral panels. The Boral panels maintain subcriticality when the canister is flooded with water during loading/unloading. The canister design evaluated for use with the Standardized Advanced NUHOMS® System is the 24PT1, which is a tube and disk design similar to canisters used in the NRC-approved MP-187 transportation cask.

The fuel assemblies are placed in baskets with square fuel cells and Boral panels fixed to the fuel cell walls. TS 4.2.3 and TS 4.2.4 requires a minimum flux trap size of 0.675 to 1.66 inches, depending on the location, and the minimum ^{10}B content of 0.025 g/cm^2 in the Boral panels. The applicant stated that 75 percent credit was taken for the minimum ^{10}B content in the Boral, however, 76 percent credit was used in the applicant's computations. This difference does not affect the overall results. The staff performed independent calculations which confirm that the use of 76 percent credit of the Boral B-10 content instead of 75 percent had a negligible impact on the analysis results.

Spacers are used to maintain the fuel position in the canister because the fuel assembly is approximately 30 inches shorter than the canister cavity. Approximately 25 inches of the canister cavity does not have neutron poison coverage. The fuel spacers were shown to maintain their integrity during normal and off-normal conditions. However, since there are no credible accident conditions during on-site transfer or storage where water can enter the canister, the spacers are not required to withstand any accident condition.

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

The staff also verified that the design-basis off-normal and postulated accident events would not have an adverse effect on the design features important to criticality safety. Chapter 3 of the SAR shows that the basket will remain intact during all normal, off-normal and accident conditions. Therefore, based on the information provided in the SAR, the staff concludes that the Standardized Advanced NUHOMS® System design meets the “double contingency” requirements of 10 CFR 72.124(a).

6.2 Fuel Specification

The Standardized Advanced NUHOMS® System is designed to store 24 PWR assemblies in each canister. The assembly types allowed are limited to Westinghouse 14x14 SC and 14x14 MOX spent fuel assemblies. The assemblies may contain the following integral control components; rod cluster control assemblies, thimble plug assemblies, and neutron source assemblies. These assemblies are discussed in SAR Section 2.1.1 and stated in TS 2.1.b of the TS. The fuel assemblies are described in detail in Section 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 enrichment. The 24PT1 may contain 24 PWR fuel assemblies with maximum initial enrichments up to 4.05 wt% ^{235}U for the UO2 fuel and up to 3.31 wt% fissile Pu for the MOX fuel.

Specifications on the condition of the fuel are also included in the SAR and TS. The 24PT1-DSC is designed to accommodate intact fuel assemblies or up to four damaged fuel assemblies (depending on the fuel type), as defined in the TS. The damaged fuel must be placed in

Damaged Fuel Containers (DFCs) which are designed to confine gross fuel particulates to a known, subcritical geometry. Also, up to two fuel cells may be left empty.

In Section 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 analyses and verified that they are consistent with the specifications given in Sections 1, 2, and 12 of the SAR and the TS.

In response to an RAI, the applicant performed sensitivity studies for various fuel parameters. The results show that k_{eff}= 0.8588 $\pm$ 0.0011 when nominal cladding thickness is used and keff = 0.8631 $\pm$ 0.0012 when minimum cladding thickness is used. Use of bounding tolerance values is consistent with the SRP[2], thus the staff disagrees with the applicant's use of nominal cladding thickness in the criticality models discussed below. However, the calculated k_{eff} for the most limiting normal condition still meets the upper subcritical limit (USL) of 0.9401 when increased to account for changes in k_{eff} due to cladding tolerance. While the most limiting accident condition keff would exceed the USL, the staff has reasonable assurance that the accident scenarios, discussed in Section 6.3.1 below are sufficiently conservative to bound this.

6.3 Model Specification

6.3.1 Configuration

The Standardized Advanced NUHOMS® System consists of the 24PT1-DSC, a TC and an AHSM. The applicant used three-dimensional calculational models in its criticality analyses. The bounding model is based on a fully flooded 24PT1-DSC in a TC. Sketches of the models are given in Section 6.3 of the SAR. The models are based on the engineering drawings in Section 1 of the SAR and consider the dimensional worst-case tolerance values. The design-basis off-normal events do not affect the design of the cask from a criticality standpoint. For the accident conditions, the neutron shield is eliminated and replaced with water. Also, failure of the damaged fuel assemblies was evaluated.

The normal condition model combined the most reactive basket dimensions. The applicant determined that the most reactive basket dimension combinations are as follows; nominal spacer disk cutout pitch as shown in drawing NUH-05-4010, Rev. 1, the maximum fuel cell inner dimension (8.9 inches), the minimum box wall thickness (0.11 inch), maximum boral panel thickness, minimum boral panel width, and all fuel cells moved toward the center of the cask within the spacer disk cutouts. For the failed fuel cases, the applicant conservatively neglected the failed fuel cans.

The calculational models also conservatively assumed the following:

- fresh fuel isotopics (i.e., no burnup credit)
- 76 percent credit for the ^{10}B loading in the Boral panels
- flooding of the fuel rod gap regions with pure water whenever the cask contains water

The active length of the fuel assemblies was modeled explicitly.

The applicant considered various levels of external (interspersed) and internal moderation to determine the most reactive moderating conditions (optimum moderation). The applicant determined that optimum internal moderation occurs when flooded with 100% density unborated water. The applicant also determined that the reactivity of a fully flooded single package is only minimally affected due to changes in the density of interspersed moderation.

The applicant modeled the outer aluminum on the boral sheets as B_4C rather than aluminum. Staff calculations show that modeling of the outer aluminum on the boral can cause a slight increase in the calculated k_{eff}, depending on the scenario modeled, and thus should be considered in any future amendments.

Preferential or uneven flooding within the canister is not a concern because the baskets are designed such that the volume inside and outside the fuel cells will flood and drain at the same rate. For damaged fuel in fuel cans, uneven draining is also not possible because the drainage holes are covered with screens that do not obstruct uniform draining and filling. The screens have a 6x6 mesh size and a 0.047 inch wire diameter.

Based on the results of the applicant's evaluation and the staff's independent confirmatory calculations, the staff concludes that the most reactive moderating conditions have been considered.

The accident condition model considered the removal of the neutron shield which is replaced with water. The k_{eff} of this accident model is less than the USL of 0.9401. For damaged fuel assemblies, the applicant also considered single- and double-ended shearing of fuel rods (with assemblies pushed up into the upper left corner of each fuel cell), and increased pitch. In these scenarios, the applicant ignored the presence of the damaged fuel can. The multiplication factors for these scenarios were all less than the USL of 0.9401, with the increased pitch case being the most reactive.

Staff confirmatory calculations show that moving the sheared fuel assemblies down towards the cask centerline may cause a slight increase in k_{eff} for the shearing cases, however, the increased pitch case was found to be the most limiting case. Also, as discussed in Section 6.2 above, the applicant used nominal cladding thickness rather than the most bounding thickness. However, the bounding accident condition considers that all 24 fuel cells are filled with damaged failed fuel and the fuel can is missing. Since the cask is limited to only four damaged fuel assemblies, the staff has reasonable assurance that the applicant's overall results for accident conditions are bounding. There are no other identified accident conditions that will adversely affect the design features important to criticality safety.

The staff reviewed the applicant's models and agrees that they are consistent with the description of the cask and contents given in Sections 1 and 2, including engineering drawings.

The staff reviewed the applicant's methods, calculations, and results for determining the worst-case manufacturing tolerance. 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 calculational models, or are bounded by the assumptions used in the calculational models.

For its confirmatory analyses, the staff independently modeled the cask using the engineering drawings and bills of material presented in Section 1.5 of the SAR. Specifically, the staff used Drawing No. NUH-05-4010, Revision 1. The staff's fuel assembly models were based on the fuel assembly parameters given in Section 6 of the SAR and in the TS. The staff found its models of the cask and contents to be similar with those of the applicant. Additionally, the staff considered variations in the ^{10}Boron content and explicitly modeled the boral panels with their aluminum covering. The staff's confirmatory analysis show that the Standardized Advanced NUHOMS® System, when used as specified in the SAR and TS, is subcritical and k_{eff} remains below the USL of 0.9401.

6.3.2 Material Properties

The composition and density of the Boral neutron absorber materials considered in the calculational models are provided in Tables 6.3-2 and 6.3-3 of the SAR.

One of the most important materials in the Standardized Advanced NUHOMS® System is the Boral neutron absorber in the basket. In Sections 6.3.2 and 9.1.7 of the SAR, the applicant provided a detailed description of the characteristics, historical applications, service experience, and manufacturing quality assurance of the Boral material. The minimum required ^{10}B content is verified through the acceptance testing program described in Section 9.1.7. As previously stated, a maximum of only 76 percent credit is taken for the ^{10}B content in the Boral panels.

The continued efficacy of the Boral, 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 Boral. In addition, a structural analysis was performed which demonstrates that the Boral panel will remain in place during accident conditions.

The staff reviewed the composition and number densities and found them to be acceptable. While 76 percent credit of the minimum ^{10}B content in the Boral was used in the applicant's computations the SAR states that 75 percent credit was given, this difference does not affect the overall results. The staff performed independent calculations which confirm that the use of 76 percent credit of the Boral B-10 content instead of 75 percent had a negligible impact on the analysis results. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

Based on the information provided on the Boral material, the staff agrees that the continued efficacy of the Boral poison can be assured by the design of the Standardized Advanced NUHOMS® System, thus meeting the requirements of 10 CFR 72.124(b).

The staff reviewed the neutron absorber acceptance test described in Section 9.1.7 of the SAR. The staff's acceptance of the neutron absorber test described in this section is based, in part, on the fact that the criticality analyses assumed only 76 percent of the minimum required ^{10}B content of the Boral. For greater credit allowance, special, comprehensive fabrication tests capable of verifying the presence and uniformity of the neutron absorber are necessary.

6.4 Criticality Analysis

6.4.1 Computer Programs

The applicant's principal criticality analysis code was KENO.Va module of SCALE 4.4, a three-dimensional, Monte Carlo code and the ENDF/B-V 44-group cross-section library.

The staff performed confirmatory analyses with the CSAS/KENO.Va modules of SCALE developed at Oak Ridge National Laboratory.

The SCALE code is a standard in the industry for performing criticality analyses. Thus, the staff agrees that the codes and cross-section sets used are appropriate for this particular application and fuel system.

6.4.2 Multiplication Factor

Results of the applicant's criticality analyses show that the k_{eff} in the Standardized Advanced NUHOMS® System will remain below the USL of 0.9401. The results of the applicant's criticality calculations for the bounding assemblies are given in Tables 6.4-1 through 6.4-4 of the SAR. The maximum k_{eff} calculated for each fuel type are summarized in the table below.

Fuel Assembly type	Maximum k_{eff} Normal Conditions*	σ	Maximum keff Accident Conditions	σ
14x14 SC Intact and Damaged Fuel**	0.8644	0.0011	0.9368	0.0012
14x14 MOX	0.9087	0.0012	0.9086	0.0011

*Nominal cladding thickness used.
**Bounds the damaged MOX fuel since only 1 damaged MOX assembly is allowed per canister

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 percent confidence level or better.

The staff performed independent criticality calculations for the Standardized Advanced NUHOMS® System. The results of the staff's confirmatory calculations 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 will remain subcritical, with an adequate safety margin, under all credible normal, off-normal, and accident conditions.

6.4.3 Benchmark Comparisons

The applicant performed benchmark calculations on selected critical experiments, chosen, as much as possible, to bound the range of variables in the Advanced NUHOMS® design. The parameters in the benchmarks bounded the parameters in the analysis with respect to fuel enrichment, fuel pin pitch, boron areal density in the separator plates, water to fuel volume ratio, assembly separation, and average energy group.

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.

The staff reviewed the benchmark comparisons in the SAR and agrees that the CSAS module of the SCALE computer codes used for the analysis was adequately benchmarked to representative critical experiments.

An USL of 0.9401 was calculated by the applicant. The USL incorporates the biases and uncertainties of the model and computer code into a value that has a 95% confidence level such that any k_{eff} less than the USL is less than 0.95, which is the design criterion.

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

6.5 Supplemental Information

The spent fuel assemblies that can be loaded into the Standardized Advanced NUHOMS® System without compromising criticality safety requirements are listed in the TS. All supportive information has been provided in the SAR, primarily in Sections 1, 2, and 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 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 1, 2, and 6 of the SAR to enable an evaluation of their effectiveness.

F6.2 The Standardized Advanced NUHOMS® System is designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry and fixed neutron poisons. An appraisal of the fixed neutron poisons 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 poisons; therefore, the 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 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 will allow safe storage of spent 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 and High-level Radioactive Waste,” Title 10, Part 72.

2. NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems.”

3. Scale 4.4, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Oak Ridge National Laboratory.

7.0 CONFINEMENT EVALUATION

The staff has reviewed the Standardized Advanced NUHOMS® System confinement features and capabilities to ensure a) that any radiological releases to the environment will be within the limits specified by the regulations[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 application was also reviewed to determine whether the Standardized Advanced NUHOMS® System fulfills the acceptance criteria listed in Section 7 of NUREG-1536, "Standard Review Plan for Dry Cask Storage systems"[2], and the following Interim Staff Guidance documents[3]: ISG-5, "Confinement Evaluation," ISG-4, "Cask Closure Weld Inspections," ISG-1, "Damaged Fuel," ISG-2, "Fuel Retrievability," and ISG-10, "ASME Code Exceptions." The staff's conclusions are based on information provided in the Standardized Advanced NUHOMS® System SAR.

7.1 Confinement Design Characteristics

The design of the Standardized Advanced NUHOMS® System is based on the Standardized NUHOMS® system described in CoC 1004. The 24PT1-DSC is nearly identical to the model FO and FC DSCs licensed for transportation in the NUHOMS® MP187 package (CoC 9255) and the Sacramento Municipal Utility District (SMUD) Rancho Seco plant site-specific license for on-site storage of spent fuel under 10 CFR 72 (SNM-2510; Docket No. 72-11). The 24PT1-DSC design has been modified to improve its resistance to corrosion in marine environments, to permit storage of control components integral with the fuel, to include damaged fuel contents, and to provide higher concrete strength and lower dose rates from the horizontal storage module.

A description of the confinement boundary is given in SAR Sections 1.2.1.1, 2.5.2, 3.1.2.1, 7.1 and Figures 3.1-2 and 7.1-1. The confinement boundary includes a high integrity austenitic stainless steel (Type 316) shell, the top and bottom closure assemblies, including the vent and drain system fabricated from stainless steel and the associated welds. The inner top cover plate has two penetrations for the vent, and siphon ports which are closed with welded cover plates. The outer top and bottom cover plates provide sealing for confining radioactive material within the 24PT1-DSC. The outer top cover plate provides redundant sealing and has a penetration to leak test the closure welds. This is closed with a welded cover plate after testing to complete the redundant sealing of the confinement boundary. The redundant closure of the DSC satisfies the requirements of 10 CFR 72.236(e) for redundant sealing of confinement systems.

The DSC is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code Section III, Subsection NB to the maximum extent practicable. Exceptions to the ASME code are listed SAR Tables 3.1.14 and 3.1.15

The welds forming the confinement boundary are described in Section 7.1.3 and Figure 7.1-1 of the SAR. The fabrication welds that affect the confinement boundary include the weld applied to the inner bottom cover plate and the circumferential and longitudinal seam welds applied to the shell. These welds are inspected volumetrically (radiographic or ultrasonic inspection, and liquid penetrant inspection) according to the requirements of Subsection NB of the ASME code. The vent and siphon block weld is liquid penetrant inspected in accordance with Subsection NB of the ASME code. The welds applied to the vent and siphon port covers, and the inner and outer top cover plates during closure operations define the confinement boundary at the top end of the 24PT1-DSC. These welds are applied using a multiple layer technique with multi level liquid penetrant testing (PT) in accordance with Subsection NB of the ASME Code, ASME Code Case N-595-1 and NRC ISG-4.

A summary of the weld examinations includes: multiple surface and volumetric examinations, pneumatic pressure testing, and leakage rate testing on the finished shell and the inner bottom cover plate at the fabricator; leakage rate testing of the closure welds (inner top cover plate and vent and siphon port cover plates) after loading the spent fuel; and multiple surface and dye penetrant examinations on the redundant confinement boundary. The applicant describes the canister inspection and test acceptance criteria in Section 9 of the SAR. The staff finds that this is acceptable provided that all NDE personnel, both at the fabricator and at the loading site, are qualified in accordance with applicable standards and codes such as SNT-TC-1A. This is a requirement of ASME Section V, Article 1, Paragraph T-140.

The applicant's proposed procedures for drying and evacuating the cask interior during loading operations were reviewed by the staff to ensure that the design is acceptable for the pressures that may be experienced during storage. The staff finds that this design, if fabricated and tested in accordance with the SAR requirements, will maintain the confinement boundary. Maintaining a stable vacuum pressure of 3 torr or less for 30 minutes during vacuum drying provides reasonable assurance that an acceptable level of moisture remains in the Standardized Advanced NUHOMS® System 24PT1-DSC.

The canister is backfilled with an inert gas (helium) to protect against cladding degradation. The Standardized Advanced NUHOMS® System 24PT1-DSC is designed to be leaktight and is tested to a leak rate of $1x10^{-7}$ atm cm^3/s as defined in "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5-1997. This testing confirms that the amount of helium lost from the Standardized Advanced NUHOMS® System 24PT1-DSC over the approved storage period is negligible. Thus, an adequate amount of helium will remain in the canister to maintain an inert atmosphere and to support the heat transfer during the storage period.

The maximum temperature limit for stainless steel clad fuel of 806°F, in accordance with EPRI TR-106440, "Evaluation of Expected Behavior of LWR Stainless Steel Fuel in Long-Term Dry Storage," has been demonstrated by the applicant in Chapter 3 of the SAR.

Section 11.2.10.2 of the SAR discusses the loading of a spent fuel assembly not allowed by Section 12 of the SAR. Although the staff does not agree that a "misloading" is a non-credible event, the safety consequences of an event are negligible. This is based on the administrative controls (i.e. dose measurements) in place, and the applicable SAR analyses in Section 4, which indicate that the integrity of the confinement boundary would remain intact.

For normal storage conditions, the Standardized Advanced NUHOMS® System 24PT1-DSC uses multiple confinement barriers provided by the fuel cladding (for intact fuel) and the Standardized Advanced NUHOMS® System 24PT1-DSC to assure that the confinement system will reasonably maintain confinement and retrievability of radioactive material. Section 3 of the SER shows that all confinement boundary components are maintained within their code-allowable stress limits during normal storage conditions. Section 4 of the SER shows that the peak confinement boundary component and fuel temperatures and pressures are within the design-basis limits for normal conditions of storage. The all-welded construction of the Standardized Advanced NUHOMS® System 24PT1-DSC with the redundant closure, extensive inspection, and testing; ensures that no release of radioactive material for normal storage and on-site transfer will occur.

7.2 Confinement Monitoring Capability

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

7.3 Nuclides with Potential Release

Since the Standardized Advanced NUHOMS® System 24PT1-DSC is designed, fabricated, and tested to meet the leak tight criteria of "American National Standard for Radioactive Materials," ANSI N14.5-1997, there is no contribution to the radiological consequences due to a potential release of the canister contents.

The allowable surface contamination limits are specified in SAR Section 12.5.2.4.c as less than 2,200 dpm/100 cm^2 from beta and gamma emitting sources and less than 220 dpm/100 cm^2 from alpha emitting sources. These limits are based on the allowable removable external radioactive contamination specified in 49 CFR 173.443. The staff agrees that the AHSM will protect the DSC from direct exposure to the environment, limiting the potential releases of this contamination. The applicant also performed a conservative analysis which shows that the dose due to the instantaneous release of all external contamination on a canister would result in a dose of < 1 mrem to an individual 100 meters from the source; much less than the requirements of 10 CFR 72.104(a) of 25 mrem annually.

7.4 Confinement Analysis

The confinement boundary is welded and tested to meet the leak tight criteria of "American National Standard for Radioactive Materials," ANSI N14.5-1997 and is shown to maintain confinement during all normal, off-normal, and hypothetical accident conditions. Also, the temperature and pressure of the canister are within the design-basis limits. Therefore, no discernable leakage is credible. As discussed in Sections 5 and 10 of this SER, the staff finds that the Standardized Advanced NUHOMS® System 24PT1-DSC meets the requirements of 10 CFR 72.104(a) and 10 CFR 106(b).

7.5 Supportive Information

Supportive information or documentation includes drawings of the Standardized Advanced NUHOMS® System 24PT1-DSC confinement boundary and applicable pages from referenced documents.

7.6 Evaluation Findings

F7.1 Section 7 of the SAR describes confinement structures, systems, and components important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the Standardized Advanced NUHOMS® System 24PT1-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 Standardized Advanced NUHOMS® System 24PT1-DSC provides redundant sealing of the confinement system closure joints using dual welds on the canister lid and closure.

F7.4 The 24PT1-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 24PT1-DSC uses an entirely welded, redundant closure system; no direct monitoring of the closure is required.

F7.5 The confinement system is leaktight for normal conditions and anticipated occurrences, thus the confinement system will reasonably maintain confinement of radioactive material. Sections 5 and 10 of the SER shows that the direct dose from the Standardized Advanced NUHOMS® System 24PT1-DSC satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F7.6 The confinement system has been evaluated by analysis. Contingent on successful completion of specified leakage tests and examination procedures, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F7.7 The staff concludes that the design of the confinement system of the Standardized Advanced NUHOMS® System 24PT1-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 Standardized Advanced NUHOMS® System 24PT1-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.

3. U.S. Nuclear Regulatory Commision, “Interim Staff Guidance.”

8.0 OPERATING PROCEDURES EVALUATION

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

8.1 Cask Loading

Detailed loading procedures must be developed by each user.

The loading procedures described in the SAR include appropriate preparation and inspection provisions to be accomplished before cask loading. These include cleaning and decontaminating the transfer cask and other equipment as necessary, and performing an inspection of the Standardized Advanced NUHOMS® System DSC to identify any damage that may have occurred since receipt inspection. However, the staff determined that the SAR does not include inspections of the failed fuel can and that the SAR should be updated to include inspection of the failed fuel can when necessary.

8.1.1 Fuel Specifications

The procedures described in Section 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 Standardized Advanced NUHOMS® System DSC. However, the staff determined that the SAR does not include instructions for use of a failed fuel can should it be required. The SAR should be revised to include instructions for loading damaged fuel into failed fuel cans in the NUHOMS®-24PT1 DSC.

8.1.2 ALARA

The ALARA practices utilized during operations are discussed in Sections 8 and 10 of this SER and are found to be acceptable.

8.1.3 Draining, Drying, Filling and Pressuraization

Section 8 of the SAR clearly describes draining, drying, filling and pressurization procedures for the NUHOMS®-24PT1 DSC that will provide reasonable assurance that an acceptable level of moisture remains in the cask and the fuel is stored in an inert atmosphere. The procedures are similar to those previously approved by the staff for the Standardized NUHOMS® System.

8.1.4 Welding and Sealing

Welding and sealing operations of the NUHOMS®-24PT1 DSC are similar to that previously approved by the staff for other DSCs used with the Standardized NUHOMS® System. The procedures include monitoring for hydrogen during welding operations. Leak checks performed by TS 3.1.3 for the NUHOMS®-24PT1 DSC demonstrate that the inner top cover plate is "leak tight" as defined by ANSI N14.5 - 1997. Sealing operations are performed in accordance with

ASME Code Section III, Division 1, 1992 Edition with Addenda through 1994 including exceptions allowed by Code Case N-595-1 for dye penetrant testing of the closure welds.

8.2 Cask Handling and Storage Operations

All handling and transportation events applicable to moving the NUHOMS®-24PT1 DSC to the storage location are similar to those previously reviewed by the staff for the Standardized NUHOMS® System and are bounded by Section 11 of the SAR. Monitoring operations include daily surveillances of the HSM air inlets and outlets, and temperature performance monitoring in accordance with TS 5.2.5.

Occupational and public exposure estimates are evaluated in Section 10 of the SAR. Each cask user will need to develop detailed cask handling and storage procedures that incorporate ALARA objectives of their site-specifc radiation protection program in accordance with TS 5.2.4.

8.3 Cask Unloading

Detailed unloading procedures must be developed by each user.

Section 8 provides unloading procedures similar to those previously approved by the staff for use with the Standardized NUHOMS® System. The procedures provide a caution on reflooding the DSC to ensure that the cask's vent pressure does not exceed 20 psig to prevent damage to the cask. The procedure also monitors for hydrogen during cutting operations.

Section 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 Standardized Advanced NUHOMS® System is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Section 8 of the applicant's SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.2 The welded cover plates of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.

F8.3 The NUHOMS®-24PT1 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. 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 Standardized Advanced NUHOMS® System 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.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

9.1 Acceptance Tests

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

9.1.1 Visual and Nondestructive Examination Inspections

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

9.1.2 Leakage Testing

The NUHOMS® -24PT1-DSC is designed to be leaktight and is tested to a leak rate of $1x10^{-7}$ atm cm^3/sec, as defined in "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5-1997[1]. The confinement boundary testing includes; leakage rate testing on the finished shell and the inner bottom cover plate at the fabricator; leakage rate testing of the closure welds (inner top cover plate and vent and siphon port cover plates) after loading the spent fuel. The staff finds that this is acceptable provided that all personnel performing leak rate testing, both at the fabricator and at the loading site, are qualified in accordance with applicable standards and codes such as SNT-TC-1A.

9.1.3 Poison Plate Acceptance Testing

The staff has reviewed the acceptance test procedures for the Boral® sheets to be used as thermal neutron absorber materials. The neutron absorber sheets are capable of performing throughout the service life of the 24PT1-DSC. The staff agrees that the environmental conditions of radiation and temperature in the cask are not sufficiently severe, under the conditions of storage service, to damage the aluminum alloys, the aluminum matrix of composite materials, or the boron-containing particles. The staff agrees that these plate materials have been demonstrated to be capable of performing their Important-to-Safety functions within the 24PT1-DSC for a licensing period of 20 years.

During fabrication, the sheet materials to be used as thermal neutron absorbers in the 24PT1-DSC are verified to have their specified minimum total ^{10}B content per unit area. This is the areal density. In production of the sheet materials, samples taken from each sheet are retained for testing and record purposes. The uniformity and areal density of ^{10}B within a sheet are verified by chemical (a destructive) analysis and/or neutron attenuation testing using a sample area of one square centimeter. The neutron attenuation is a non-destructive test method.

Normally, the ^{10}B content is assessed using chemical analyses and samples (of one square centimeter) taken from each end of the rolled product. The sample frequency is determined using standard statistical procedures to assure that the minimum ^{10}B areal densities are achieved, at a specified confidence level. The SAR states that this areal density verification is done with a 95/95 confidence level–the staff interprets this to mean that a statistical analysis of data taken on coupons taken from all plates not rejected for physical (malformation and thickness) reasons is used to ensure that at least 95 percent of the lot will meet 83.3 percent of the required minimum^{10}B content with 95 percent confidence. Accordingly, a one-sided tolerance limit corresponding to the 95 percent probability / 95 percent confidence level for the relevant sample size shall be used. Initial sampling shall be 100 percent of the coupons with reduced (50 percent) sampling being introduced if all coupons in the first 25 percent of the lot are acceptable. A rejection during reduced inspection will require a return to 100 percent inspection of the lot. The staff finds these test methods and procedures for data analysis are acceptable for use of this product at the 83.3 percent level of credit for the reliability of the material.

This level of credit corresponds to 76 percent credit in the criticality analyses for the 24PT1-DSC. The level of credit is less than the demonstrated effectiveness because, in the analyses, only 90 percent of the demonstrated effectiveness is used due to uncertainties associated with the criticality calculations, i.e. neutronic effects. The uncertainty arises from the difference in the neutronic value of the poison plates in the available benchmark experiments versus that in the cask configurations. Thus, a lower neutronic value is assumed to be present for a given amount of poison in the configuration of a commercial poison plate material when compared with that of plates used in benchmark experiments. Thus, the level of credit used in analyses (76 percent), is only 90 percent of the “demonstrated effectiveness” (83.3 percent) of the commercial material. The staff finds that using these procedures, the minimum safety requirement for the neutron absorber sheets are met.

The staff agrees that chemical analyses may be substituted for measurements of the neutron attenuation property for these absorber materials. The staff expects that when this substitution is made, the chemical analyses will be taken on a product produced using the same process and specifications as was used for attenuation tests. The qualification process should involve a correlation between the two types of measurements and ,to the extent possible, should make use of the same measurement point (or spot on the plate) and similar surface areas over which measurements are taken. With these considerations, the staff concludes the correlation and/or a statistical analysis is sufficient to alleviate concerns that the chemical analysis may overestimate the absorptivity of the plate material.

The area of test samples suggested in the SAR is one square inch for test coupons used to measure ^{10}B content. This area is regarded by staff to be unsuitably large for measurements of uniformity of the product. Accordingly, an area of one-square-centimeter, as previously approved for use in the NUHOMS® system that uses the 24PT DSC, has been substituted in the SER for the 21PT1-DSC of the Advanced NUHOMS® system.

The staff agrees that these acceptance test procedures are adequate for demonstrating the efficacy of the thermal neutron absorber sheet materials proposed for use at the 76 percent level of credit for the Boral® taken in the criticality calculations in this application.

9.1.4 Thermal Conductivity Testing

The Standardized Advanced NUHOMS® System material is not tested to verify that the thermal conductivity equals or exceeds the values listed in Section 4.2. However, monitoring of AHSM thermal performance continues, in accordance with TS 5.2.5 throughout the duration that the cask is in service to ensure that the integrity of the AHSM and fuel cladding are maintained.

9.2 Evaluation Findings

The staff concluded that the proposed acceptance testing and maintenance program meet regulatory requirements. Other specific findings are as follows:

F9.1 Sections 3, 7, and 9 of the SAR describe the applicant's proposed program for preoperational testing and initial operations of the Standardized Advanced NUHOMS® System.

F9.2 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. Section 2, Tables 2.5-1 of the SAR identify the safety importance of SSCs and Section 3 of the SAR presents the applicable standards for their design, fabrication, and testing.

F9.3 The user will examine and test the Standardized Advanced NUHOMS® System to ensure that it does not exhibit any defects that could significantly reduce its confinement effectiveness. Section 3, 7, and 9 of the SAR describe this inspection and testing.

F9.4 Cask marking and data plate information are discussed in the Standardized NUHOMS® System FSAR in support of CoC 1004 and were not reevaluated for this application.

F9.5 The staff concludes that the acceptance tests and maintenance program for the Standardized Advanced NUHOMS® System 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 approved period of use. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

9.3 References

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

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

3. ASTM E1461 "Thermal Diffusivity of Solids by the Flash Method."

10. RADIATION PROTECTION EVALUATION

The staff reviewed the radiation protection design features, design criteria, and the operating procedures of the Standardized Advanced NUHOMS® System to ensure that its use will meet the regulatory dose requirements of 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), and 10 CFR 72.236(d).[1] This application was also reviewed to determine whether the Standardized Advanced NUHOMS® System fulfills the acceptance criteria listed in Section 10 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems[2]. The staff's conclusions are based on information provided in the Standardized Advanced NUHOMS® System SAR.

10.1 Radiation Protection Design Criteria and Design Features

10.1.1 Design Criteria

The radiological protection design criteria are the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. This is consistent with NRC guidance. As required by 10 CFR Part 20 and 10 CFR 72.212, each general licensee is responsible for demonstrating site-specific compliance with these requirements. The TS also establish an administrative program which controls dose limits for the TC and the AHSM that are based on calculated dose rate values which are used to determine occupational and off-site exposures. The TS also establish exterior contamination limits for the DSC to keep contamination levels below 2,200 dpm/100 cm^2 for beta and gamma radiation, and 220 dpm/100 cm^2 for alpha radiation.

10.1.2 Design Features

Sections 5 and 10 of the SAR define the radiological protection design features which provide radiation protection to operational personnel and members of the public. The radiation protection design features include the following:

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

- the use of water in the DSC cavity (when possible) to reduce occupational dose rates,
- the low-maintenance design that reduces occupational exposures during ISFSI operation, and
- the implementation of ALARA principles into the cask design and operating procedures that reduce occupational exposures.

The design features that address process instrumentation and controls, control of airborne contaminants, decontamination, radiation monitoring, auxiliary shielding devices and other ALARA considerations are very similiar to the Standardized NUHOMS® System.

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

10.2 Occupational Exposures

Section 8 of the application discusses general operating procedures that general licensees will use for fuel loading, DSC/transfer cask operations, DSC transfer into the HSM, and fuel unloading. Table 10.3-1 of the SAR provides the estimated number of personnel, the estimated time, and the tasks involved and the estimated dose to load one canister. The estimated occupational doses are based on actual Standardized NUHOMS® System operating experience. The dose estimates indicate that the total occupational dose in loading a single canister with design basis fuel into the AHSMs is approximately 3.12 person-rem.

10.3 Public Exposures From Normal and Off-Normal Conditions

Section 10 of the SAR presents the calculated direct radiation dose rates at distances beyond 100 meters from a sample cask array configuration loaded with design basis fuel.
Figure 10.2-1 and 10.2-2 depict estimated dose rate versus distance curves for a single AHSM and an 2x10 array of AHSMs. Tables 10.2-1 through 10.2-5 specify distances at which the regulatory design limit of 25 mrem/yr can be achieved. An array of 20 Advanced NUHOMS® -24PT1-DSCs loaded with design basis fuel and placed in the AHSM is below regulatory limits at approximately 200 meters for a 2x10 array. This assumes 100 percent occupancy for 365 days.

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

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

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

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

Section 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.

The SAR analysis indicates the worst case shielding consequences results in a dose at the controlled area boundary that meets the regulatory requirements of 10 CFR 72.106(b). Section 11 of the 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 are presented in Section 11 of the SER. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents and natural phenomena will be below the regulatory limits in 10 CFR 72.106(b).

10.5 ALARA

Sections 5, 7, and 10 of the SAR present evidence that the Standardized Advanced NUHOMS® System radiation protection design features and design criteria address ALARA requirements, consistent with 10 CFR Part 20 and Regulatory Guides 8.8[3] and 8.10[4]. Each site licensee will apply its existing site-specific ALARA policies, procedures, and practices for cask operations to ensure that personnel exposure requirements in 10 CFR Part 20 are met. Because the transfer cask may have to be drained when used with the NUHOMS® 24PT1-DSC and a 100-ton crane, the occupational dose rates may be higher than when loading other approved canisters. Each plant will have to consider the use of this canister with respect to their particular ALARA implementation philosophy.

The staff evaluated the ALARA assessment of the Standardized Advanced NUHOMS® System and found it acceptable. Section 8 of the SER discusses the staff's evaluation of the operating procedures with respect to ALARA principles and practices. Operational ALARA policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20. In addition, the TS establish an administrative program which controls dose limits dose and surface contamination limits to ensure that occupational exposures are maintained ALARA.

10.6 Evaluation Findings

F10.1 The SAR sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

F10.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F10.3 The Standardized Advanced NUHOMS® System is designed to provide redundant sealing of the confinement system.

F10.4 The Standardized Advanced NUHOMS® System is designed to facilitate decontamination to the extent practicable.

F10.5 The SAR adequately evaluates the Standardized Advanced NUHOMS® System important to safety SSC to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F10.6 The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

F10.7 Operational restrictions necessary to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The Standardized Advanced NUHOMS® System is designed to assist in meeting these requirements.

F10.8 The staff concludes that the design of the radiation protection system of the Standardized Advanced NUHOMS® System 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 Standardized Advanced NUHOMS® 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 applicant’s analyses, the staff’s confirmatory analyses, and acceptable engineering practices.

10.7 References

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

2. U.S. Nuclear Regulatory Commission, “Standard Review Plan for Dry Cask Storage Systems,” NUREG-1536, January 1997.

3. U.S. Nuclear Regulatory Commission, Information Relevant to Ensuring that Occupational Radiation Exposures Will Be As Low As is Reasonably Achievable," Regulatory Guide 8.8, Revision 3, June 1978.

4. U. S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable," Regulatory Guide 8.10, Revision 1-R, May 1977.

11.0 ACCIDENT ANALYSIS

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

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

11.1 Off-Normal Conditions

Off-normal operations are Design Event II as defined by ANSI/ANS 57.9[1]. These events can be described as not occurring regularly, but can be expected to occur with moderate frequency (on the order of once per year). The 24PT1- DSC off-normal events are described in Section 11.1 of the SAR. Two off-normal events are identified and analyzed for the Standardized Advanced NUHOMS® System that are defined as bounding the range of off-normal events. These are inadvertent jamming of the DSC while loading or unloading the AHSM and extreme external ambient temperatures. These events have been analyzed and reported in the appropriate sections of this report such as Sections 3 and 4.

11.2 Accident Events and Conditions

Accident events and conditions are Design Event III and IV as defined in Reference 1. They include natural phenomena and human-induced low probability events. The applicant provided analyses to demonstrate design adequacy for the accident-level events discussed below. The 24PT1-DSC System is designed to accommodate postulated accidents that are described in Section 11 of the SAR. Ten postulated accident conditions are addressed in the section.

11.2.1 Blockage of Air Inlet and Outlet Openings

The Standardized Advanced NUHOMS® System has been designed based on the postulation of the complete blockage of the AHSM ventilation air inlet and outlet openings. The source of the blockage could be debris accumulation at the openings that was transported by events such as floods or tornadoes. The structural loads that may be imposed on the AHSM are small and are bounded by the loads imposed by tornadoes or by earthquakes. The resulting thermal conditions are analyzed and discussed in Section 4. The operating conditions are such that the vents must be cleared within a 40-hour time frame in order to restore ventilation. The thermal-induced stresses are considered in the loading combinations as discussed in Section 3.1.2.1.4.

There are no off-site dose consequences resulting from this accident scenario. The on-site dose received by workers from this accident is estimated at no more than one man-rem per 8 hour period during the removal of debris from the vents.

11.2.2 Accidental Pressurization of 24PT1-DSC

The Standardized Advanced NUHOMS® System design is based on the postulated failure of spent fuel cladding and the release of spent fuel rod fill gas and free fission gas causing the internal pressurization of the 24PT1-DSC. The analysis to determine the magnitude of the pressure is provided in Section 4 and the impact of the pressure on structural loads is presented in Section 3.1.2.1.3.

There are no dose consequences from this accident since the 24PT1-DSC is designed to withstand the accident pressure as a pressure vessel.

11.2.3 Inadvertent Loading of a Newly Discharged Fuel Assembly

The Standardized Advanced NUHOMS® System design is based on the elimination of the possibility of the loading of a spent fuel assembly for storage that has a heat generation rate greater than 0.583 kW into the 24PT1-DSC canister. This accident scenario could be the result of human error in selecting the spent fuel elements for loading or by failure of a single administrative control relative to fuel handling. In order to eliminate the occurrence of this type of accident there will be the use of multiple administrative controls and independent reviews prior to the movement or storage of spent fuel. In addition, the applicant performed a review of all WE 14 x 14 SC and MOX fuel assembly inventories of fuel fabricated by January 2001 and determined that all assemblies meet the requirements of Chapter 12 of the SAR. Therefore, the storage of WE 14 x 14 SC and MOX fuel assemblies fabricated after January 2001 will require evaluation by the general licensee for the consequences of a misload. If the misload is determined to be a credible event, the applicant must request NRC evaluation and approval before that fuel may be loaded.

There are no dose consequences since the occurrence is not considered credible.

11.2.4 Burial of the AHSM

The design of the Standardized Advanced NUHOMS® System is based on the postulated burial of the AHSMs as a result of the effects of an earthquake or a flood or other such phenomenon such that earthen material is deposited on the AHSM. It was assumed the AHSM was completely buried in a medium that restricts cooling and it was determined that there would be no adverse consequences if the AHSM is uncovered within 40 hours. The structural loading from the burial is considered to be bounded by the live loads and seismic loads.

There are no dose consequences provided the AHSM is uncovered within 40 hours.

11.2.5 Accidental Drop of 24PT1-DSC Inside the Transfer Cask

The design basis for an accidental drop of the 24PT1-DSC is for the occurrence of an accidental drop of the transfer cask while an 24PT1-DSC canister loaded with spent fuel is contained within the transfer cask. Other drop scenarios have been considered and determined to not be credible or are controlled under the nuclear power plant licensing basis of the facility's 10 CFR Part 50 license. Plant specific cask drop scenarios to be considered for use on the Standardized Advanced NUHOMS® System include the operations inside the fuel handling building that encompass loading the canister that is inside the transfer cask that is itself in the fuel pool, lifting the cask with a loaded canister from the spent fuel pool and placement of the assembly on the transport skid/trailer including down-ending the cask on the trailer. A cask end drop from the transport skid/trailer during the transport on site or during transfer to the AHSM has been considered and judged to not be credible. The selected design parameters for the Standardized Advanced NUHOMS® System were defined as a loading of 75g from a horizontal side drop and 25g from a oblique corner drop at an angle of 30-degrees with the horizontal. For the purpose of meeting requirements of 10 CFR Part 50 and 10 CFR Part 71, neither of which are part of this safety evaluation, a value of 60g was the design basis for the end drop. With these design conditions there is no damage expected to the confinement boundary of the 24PT1-DSC, but there may be damage to the neutron shielding of the transfer cask.

The dose rate resulting from such damage to the neutron shield has been bounded with an upper limit based on previous calculations (CoC 1004) and those discussed in Section 5 of the SAR. The dose rate at the surface of the transfer cask as a result of the loss of the neutron shield of the transfer cask with the loaded 24PT1-DSC inside is bounded by an upper value of 2128 mrem/hr (gamma and neutron). Based on an 8-hour recovery/return to normal shielding condition the estimated additional dose to the on-site worker at an average distance of 15 feet is less than 2.5 rem. Off-site individuals at a distance of 2000 feet would receive an additional dose of 0.04 mrem for an assumed 8 hour exposure. This represents a dose of approximately twice the annual dose if standing in front of the AHSM at 600 meters. The increase from this accident condition is within the limits for this accident condition.

11.2.6 Fire and Explosion

A credible fire used as the design basis when the Standardized Advanced NUHOMS® System is being used in the transfer mode is the rupture and ignition of the 300 gallon fuel tank of the transfer/support equipment which results in an engulfing fire around the transfer cask loaded with a 24PT1-DSC. This condition bounds any condition of fire and the Standardized Advanced NUHOMS® System once the 24PT1-DSC is loaded and stored in the AHSM. The 300 gallon fire accident evaluation is addressed in Section 4.6 of the SAR and it is concluded that the blockage of air inlet and outlet openings (Section 11.2.1 herein) is a more critical case than the 300 gallon fire. A credible explosion in the vicinity of the AHSM is considered to be bounded by the external design pressure for the Standardized Advanced NUHOMS® System of 22 psi that results from the flooding condition.

There is no breach of the confinement boundary from the accident condition since the design of the 24PT1-DSC is capable of withstanding these conditions. The fire condition may result in the loss of the transfer cask neutron shielding if the fire occurs when the 24PT1-DSC is inside the transfer cask. The effect of this set of conditions is bounded by the results from the cask drop scenarios discussed in Section 11.2.5 herein.

11.2.7 Lightning

Lightning striking the AHSM, which encapsulates and protects the 24PT1-DSC, and causing an accident condition for the functioning of the Standardized Advanced NUHOMS® System is not considered to be credible. A lightning strike in the vicinity of the AHSM will follow a low impedance path offered by the surrounding environment, or by a lightning protection system if the site specific characteristics have resulted in the installation of such a system. The 24PT1-

DSC is protected by the concrete of the AHSM and no mechanical or thermal damage is expected.

There are no dose consequences from such an accident as a lightning strike.

11.2.8 Flood

The structural design basis accident flood conditions are discussed in Section 3.1.2.1.5 of this SER and in Section 11.2.3. Under the flood conditions, including a current of 15 feet per second, a loaded cask has a safety factor of 1.3 against overturning and a safety factor of 1.27 against sliding. The thermal analysis is addressed in Section 4 herein.

The dose due to flooding of the AHSM is negligible since the 24PT1-DSC maintains the confinement boundary. If a flooding event were to occur, there may be the need for some additional temporary shielding during the post-flooding cleanup period for the purposes of ALARA.

11.2.9 Seismic Events

11.2.9.1 Introduction

The application seeks approval for a freestanding cask to slide, as well as tip, on the surface of the concrete pad during a high seismic event. However, the current Spent Fuel Project Office (SFPO) seismic criteria accepts only limited sliding and tipping on a case-by-case basis. Therefore, at SFPO's request, the NRC's Office of Nuclear Regulatory Research (RES) sponsored a project in 1998 to establish general seismic review criteria to assess cask behavior in high and low seismic sites, and under various soil conditions. In addition, the RES program formed an oversight Review Panel whose members consisted of national seismic experts, nuclear industry representatives and NRC staff. The review panel provided direction and recommendations to the project.

The panel recommended that confirmatory analyses of the Standardized Advanced NUHOMS® System for spent (irradiated) nuclear fuel be conducted and the results may be used together with the vendor's submittals and other relevant information as technical bases for the review of this application. The confirmatory analyses was performed by the Sandia National Laboratories and Anatech Corporation.

Chapter 3 of the SAR addresses the stress or forces on the Standardized Advanced NUHOMS® System resulting from seismic loads by linear elastic analyses under normal and off-normal conditions. The AHSM non-linear seismic stability analyses are discussed in Chapter 11 of the SAR and address the accident conditions. It is the non-linear seismic stability analyses that determines the maximum sliding and uplifting of the AHSM array under the dominating accident event for the AHSM array.

The staff review of the SAR and the confirmatory report (Sandia) on the AHSM non-linear seismic stability is addressed below.

11.2.9.2 SAR AHSM Non-linear Seismic Stability Analyses

11.2.9.2.1 AHSM array

An array is defined as a minimum of three AHSMs plus shield walls. Each AHSM has steel rods tying the top shield block to the base storage block in the vertical direction and interlocking concrete keys in the horizontal direction. The keys are designed to resist shear in both horizontal directions. Between the modules there are steel ties at the top and the bottom blocks to prevent tipping and separation. The horizontal and vertical keys at the module wall interfaces are to restrain relative horizontal sliding and vertical movement between AHSMs. With the AHSM row assembly configuration there is a 10 foot clear space required around all sides to prevent any possible contact between rows of AHSMs under the maximum design basis seismic event. This clear space buffer zone sets the limit for AHSM sliding with any extra space being a margin between the maximum calculated movement and the maximum movement that can occur before row arrays could impact one another.

11.2.9.2.2 Models and computer codes

The non-linear analyses are performed using a three-dimensional finite element model and the LS-DYNA computer code. The minimum AHSM array of three AHSMs with each AHSM being modeled as a separate structure forms a portion of the model that includes the 5-foot thick concrete top shield, 3-foot thick rear and end shield walls, the 24PT1-DSC, contact at the concrete keys, and the seismic ties that are also included in the model. The interfacing structural ties between AHSM components and individual AHSMs are modeled with stiffness properties representing the physical characteristics of the ties. The contact interfaces of the shear keys are also modeled as actual shear keys. The ISFSI pad is included in the model since it supports the AHSM arrays and provides the contact surface where any seismically induced movement may occur. Such movement may consist of sliding or uplift due to tipping. Components such as the AHSM base storage block, shield block and shield walls, the pad, etc., are modeled as rigid bodies. This is conservative, since the means of energy dissipation is in the movements that can occur such as sliding, and the dissipation in impact damping.

11.2.9.2.3 Coefficient of Friction

A range of coefficients of friction is used to bound the behavior of the surfaces between the concrete of the bottom of the cask and the concrete of the top of the pad. The range used varied between 0.3 and 0.7. The lower bound value conservatively maximizes the sliding response, while the upper bound value maximizes the rocking or tipping response.

11.2.9.2.4 Seismic analyses

Three sets of acceleration time histories were utilized for the multiple time history analyses. Each set of time histories consisted of three statistically independent orthogonal components. Each component was developed to match the Regulatory Guide (RG) 1.60[2] response spectra shape with 4 percent damping, anchored at 1.5g zpa for the horizontal directions and 1.0g zpa for the vertical direction. The time histories meet the spectral matching requirements and the power spectral density (PSD) target requirements of Section 3.7.1 of the Standard Review Plan.

For each time history, the input to the LS-DYNA model consist of three orthogonal acceleration components applied simultaneously at the top of the pad for determining the sliding/rocking response of the AHSM array. The three sets of acceleration time histories are represented by the three past strong earthquake motions, i.e., the Taiwan earthquake in 1999, the Landers/Lucern earthquake in 1992, and the Tabas earthquake in 1978. All three earthquake time histories have excitation period of 40 seconds.

Six analyses are performed as design cases using the three sets of time histories and upper and lower bound coefficient of friction; another six analyses are performed using the three sets of time histories and incorporating construction tolerance, partially loaded DSC, various coefficients of friction, etc., as sensitivity analysis cases.

11.2.9.2.5 Final results

After enveloping all analyses, the maximum calculated sliding displacements are on the order of 44 inches in the X-direction resulting from the Taiwan time-history and 33 inches in the X-direction resulting from the Tabas time-history, with X-direction being parallel to the three-module doors of the model array. The maximum uplift from tipping was 0.6 inch for the worst sensitivity case.

11.2.9.3 Independent Confirmatory Seismic Stability Analyses

The Sandia National Laboratories and Anatech Corporation jointly performed independent confirmatory analyses for the Standardized Advanced NUHOMS® design. The purpose of the confirmatory analyses was to develop a coupled finite element model of the module/cask, pad, and soil foundation; to provide methodologies and techniques as recommended by the Review Panel to perform a full scope soil-structure-interaction (SSI) analyses; and, to verify the sliding and rocking response of the Standardized Advanced NUHOMS® System cask under postulated seismic excitations. The panel recommended two specific time history accelerations and three coefficients of friction (i.e., 0.3, 0.5, and 0.75) to be used in the analyses.

11.2.9.3.1 Coupled models and computer codes

The non-linear analyses were performed using three-dimensional coupled finite element models of the three-module assembly, a flexible concrete pad, and an underlying soil foundation. The geometry modeling of the assembly is based on Transnuclear West drawings[4]. The model of the concrete pad covers the area directly underneath the assembly plus 10 feet clearance around the four sides. Since the casks are freestanding on top of the pad, the cask/pad (concrete slab) interface becomes the dominant source of non-linearity controlling the overall behavior of the system. The ability to predict the behavior of the interface response requires a comprehensive overall model and a computer code that can reasonably handle the interface problem. The ABAQUS computer code[5] was used to examine the non-linear behavior of the assembly and the SSI effect. Two sets of time history (T/H) accelerations were recommended by the panel as seismic excitations. The first time history was the artificial T/H provided by the staff based on RG 1.60 with a 80-second excitation duration. The second T/H used was the data from the Tabas earthquake records. Both seismic excitations are defined at the free surface, but are applied at the base of the soil foundation in the coupled model.

11.2.9.3.2 Soil Model

The size of the soil foundation model plays an important role in assessing the SSI effect. The soil-structure-interaction modeling techniques described in “US Corps of Engineers, Engineer Technical Letter No. 1110-2-339, March 1993,"[6] together with sensitivity studies using Tabas records provide guidelines in determining the lateral dimensions of the soil model. The depth of the soil foundation model is recommended to be at least twice the width of the concrete pad (45'-3"), according to Reference 7 so as to simulate the behavior of a semi-infinite soil foundation underneath the concrete pad. Therefore, a depth of 100 feet is chosen. The soil model dimensions are 397'-1" x 436'-9" x 100' (depth).

11.2.9.3.3 Soil Foundation

Modeling the soil mass properly is essential for achieving the response solution. The soil model boundary conditions affect the soil wave propagation and hence the shaking intensity on the cask. Sandia used the edge-column concept at the four vertical side boundaries of the finite element mesh constraining the horizontal response of the elements to respond in a shear deformation mode, so that the edge column would behave as an infinite boundary element. Sensitivity studies using the Tabas earthquake records (5 percent damping) and the friction coefficient of 0.3 yields a consistent solution at the top corners of the edge-columns, thereby demonstrating the validity of the analysis methodology of the soil model.

The 100-foot depth of the soil foundation is divided into six sub-regions. Its soil properties, such as shear wave velocity and damping profiles, are developed in accordance with the EQE report, “Soil-Structure-Interaction of the Independent Spent Fuel Storage Installation for San Onofre Nuclear Generation Station Unit 1."[7] The final model uses the best-estimate strain-compatible shear wave velocity and damping profiles for high intensity shaking.

11.2.9.3.4 Deconvolution Procedure

The seismic time-histories are defined at the free surface, yet the seismic excitations were applied at the base of the soil foundation in the coupled model. The Review Panel recommended a procedure to approximate the seismic loading at the foundation base and eventually achieve the desired surface shaking intensity by successive iterations. This is known as the deconvolution procedure. The frequency domain deconvolution procedure developed by Anatech was applied to the time-history of seismic accelerations so that the time-history magnitudes and frequency contents are adjusted simultaneously. Therefore, when deconvoluted seismic motions are applied at the base of soil foundation, the dynamic characteristics of the original seismic motions are preserved, the desired surface shaking intensity can be achieved, and the SSI effect is reflected. The validity of the deconvolution procedure has been demonstrated by the edge column surface response solution discussed in SER Section 11.2.9.3.3.

11.2.9.3.5 Final Results

A total of 12 SSI analyses were performed. The maximum cask sliding displacements are on the order of 58 inches resulting from the Tabas time-history and 53 inches from the NRR time-history in the E-direction. The E-direction is parallel to the three module-doors. The resulting maximum uplift was 0.4 inch from the Tabas time-history.

11.2.9.4 Comparison of results and conclusion

Both the Standardized Advanced NUHOMS® System application (SAR) and the independent confirmatory analyses (Sandia) used bounding (strong) earthquake time-history accelerations for predicting sliding and rocking responses. In the former case, Taiwan, Landers/Lucern and Tabas time-histories were applied at the top of the pad with the coefficient of friction ranging between 0.3 to 0.7; whereas in the latter case, NRR and Tabas time-histories were applied at the base of the soil model which is 100 feet below the pad. The coefficient of friction ranged between 0.3 to 0.75.

In both cases, the maximum sliding response is less than half the distance between the module assemblies (5 feet). The maximum rocking response is negligibly small. For comparison purposes, Tabas time-history records were used as earthquake input motion in the SAR and RES confirmatory analyses. As a result, the maximum calculated cask sliding displacements are 33 inches (SAR) and 58 inches (Sandia), respectively. The larger displacement (58 inches) reflects the SSI effect with the Sandia's coupled finite element model which includes the soil foundation.

The staff concludes that by comparing the results contained in the "Final Report on Seismic Analysis of Three-Module Rectangular Transnuclear West Module/Cask" by Sandia with the AHSM Non-linear Seismic Stability Analyses presented in the SAR, the evaluation of the SAR non-linear seismic stability analyses provides reasonable assurance that the AHSM stability analyses results are adequate and acceptable based on the following review findings:

a. The five postulated earthquakes in both analyses are of extreme strong shaking that bound with significant margin those of most nuclear power plant sites in the U. S.

b. The requirement of three coupled AHSM assemblies and the use of coefficients of friction between the concrete surfaces (0.3 for sliding, 0.7 for tipping) are conservative in predicting AHSM sliding and rocking response.

c. The ISFSI pad layout requires each AHSM row assembly to have 10 feet spacing around all sides. This provides space for the maximum cask slide under design basis earthquakes (SAR).

d. The 3-dimensional, coupled finite element model used in the Sandia analysis included the structural features of the three-module assembly, the cask/pad interface, and the soil model. The soil model dimensions are determined based on rigorous analyses and sensitivity studies.

e. The validity of the analysis methodology and the deconvolution procedure used in the Sandia confirmatory calculations has been demonstrated by sensitivity studies. The best estimated strain compatible soil properties are based on EQE report which utilizes the actual soil properties.

f. The computer code, ABAQUS, used in the Sandia analyses, is capable of handling the interface response, i.e., the separation between the cask base and the pad, a unique non-linearity issue that exists with freestanding casks.

g. The maximum AHSM sliding displacements corresponding to the five strong earthquakes are within half the distance between the row assemblies, and are based on a coefficient of friction of 0.3 between the concrete surfaces, an extremely conservative analytic assumption. Furthermore, the analyses performed in the SAR and by Sandia are totally independent, in that they use different approaches in the dynamic response modeling, different methodologies, different analytical procedures, and different computer codes.

h. When the cask/pad model is decoupled from the soil foundation the dynamic response does not address potential soil-structural-interaction amplification from the soil, per 10 CFR 72.212(b)(2), a general licensee is required to perform written evaluations, prior to using the Standardized Advanced NUHOMS® System at a site, that validates that the effects of the site-specific ISFSI soil-structure-interaction falls within the response spectra described in the SAR. The licensee shall also evaluate for potential site-specific liquefaction or other soil instability due to vibratory ground motion. (10 CFR 72.102)

11.2.10 Tornado Wind and Tornado Missiles

The Standardized Advanced NUHOMS® System is designed to be located anywhere within the continental United States of America and therefore was designed to meet the most severe tornado wind and tornado missile criteria for the accident conditions. The criteria are based on those specified in NUREG-0800[3] and NRC Regulatory Guide 1.76[8]. The specific design parameters are discussed in Section 3.1.2.1.6 herein. These resulted in equivalent wind pressures of 397 psf and 196 psf with a suction of 357 psf. The range of tornado driven missiles included a utility pole, steel pipe, an armor piercing artillery shell and a 4000 pound automobile with a 20 square foot frontal area moving at 195 feet per second.

No dose consequences are associated with these postulated accident conditions since the Standardized Advanced NUHOMS® System is designed to resist the imposed loads.

11.3 Evaluation of Findings

F11.1 Structures, systems, and components of the Standardized Advanced NUHOMS® System are adequate to prevent accidents and to mitigate the consequences of accidents and natural phenomena events that do occur.

F11.2 The spacing of casks is discussed in Sections 1 and 11 of the Standardized NUHOMS® System FSAR. The staff has previously reviewed and approved the cask spacing to ensure accessibility of the equipment and services required for emergency response.

F11.3 The applicant has evaluated the Standardized Advanced NUHOMS® System to demonstrate that it will reasonably maintain confinement of radioactive material under credible accident conditions.

F11.4 An accident or natural phenomena event will not preclude the ready retrieval of spent fuel for further processing or disposal.

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

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

11.4 References

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

2. NRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, 1973.

3. NRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Revision 2, July 1981.

4. Transnuclear West Drawings, Nos. SEE-01.2002 and SCE-01.2003, Preliminary Version.

5. ABAQUS/Explicit: User's Manual, Version 5.8-19, Hibbitt, Karlson and Sorensen, Inc., 1998, Pawtucket, RI.

6. US Army, Corps of Engineers, Engineering Technical Letter No. 1110-2-339, March 1993.

7. "Soil-Structure-Interaction of the Independent Spent Fuel Storage Installation for San Onofre Nuclear Generating Station, Unit 1," EQE Report 201038.02 - R - 001, Rev. 0, April 3, 2000.

8. NRC Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants, " April 1974.

12.0 CONDITIONS FOR CASK USE - TECHNICAL SPECIFICATIONS

In this section the staff evaluated the operating controls and limits or the TS, including their bases and justification, that the applicant established as conditions of use for the Standardized Advanced NUHOMS® System.

For simplicity in defining the acceptance criteria and review procedures, the term technical specifications may be considered synonymous with operating controls and limits. The conditions for use and TS define the conditions that are deemed necessary for safe Standardized Advanced NUHOMS® System use. Specifically, they define operating limits and controls, monitoring instruments and control settings, surveillance requirements, design features, and administrative controls that ensure safe operation of the Standardized Advanced NUHOMS® System. As such, these conditions for use and TS are included in a DCSS certificate of compliance.

12.1 Conditions for Use

The conditions for use of the Standardized Advanced NUHOMS® System were developed by NRC staff in accordance with guidance provided in NUREG 1745, “Standard Format and Content for 10 CFR Part 72 Cask Certificates of Compliance. The conditions were derived from analysis and evaluations provided in the Standardized Advanced NUHOMS® System SAR and pertain to the design, construction and operation of the system.

12.2 Technical Specifications

Section 12 of the SAR describes the TS required to ensure that the Standardized Advanced NUHOMS® System is operated safely. The TS were established to implement requirements for the design, construction, and operation of the Standardized Advanced NUHOMS® System by a licensee using a general license in accordance with 10 CFR Part 72. The TS, as approved by the staff, are contained in an appendix to the Certificate of Compliance for the Standardized Advanced NUHOMS® System and address the following areas:

- Use and Application
- Approved Contents
- Limiting Conditions for Operation (including Surveillance Requirements)
- Design Features
- Administrative Controls

Table 12-1 of this SER lists the TS to be implemented for the Standardized Advanced NUHOMS® System.

12.3 Evaluation Findings

F12.1 Table 12-1 of the SER lists the Technical Specifications for the use of the Standardized Advanced NUHOMS® System. These Technical Specifications are contained as part of the Certificate of Compliance.

F12.2 The staff concludes that the conditions for use of the Standardized Advanced NUHOMS® System, identify necessary Technical Specifications to satisfy 10 CFR Part

72 and that the applicable acceptance criteria have been satisfied. The Technical Specifications provide reasonable assurance that the cask will provide for safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

TABLE 12-1

Standardized Advanced NUHOMS® System
Technical Specifications

TABLE 12-1 (cont.)

5.0 Administrative Controls
- 5.1 Procedures
- 5.2 Programs
  - 5.2.1 Safety Review Program
  - 5.2.2 Training Program
  - 5.2.3 Radiological Environmental Monitoring Program
  - 5.2.4 Radiation Protection Program
  - 5.2.5 AHSM Thermal Monitoring Program
- 5.3 Lifting Controls
  - 5.3.1 Cask Lifting Heights
  - 5.3.2 Cask Drop

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether TN West has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G. The staff has previously reviewed and accepted the TN West QA program. TN West operates under the same QA program as its parent company Transnuclear, Inc. (TN). The staff has performed inspections of the QA program, as implemented at both TN West and TN, and found that both met regulatory requirements. Therefore, while TN West originally submitted the application for the Standardized Advanced NUHOMS® System, the staff concluded the CoC may be issued to TN as requested by letter on October 4, 2001.

14.0 DECOMMISSIONING

The purpose of this review is to assess the applicant's conceptual decommissioning plan to assess whether the cask system provides adequate provisions to facilitate decommissioning of the ISFSI once the spent fuel has been transferred to the Department of Energy or some other location for storage or reprocessing. The applicable 10 CFR Part 72 requirements for decommissioning are 72.130 and 72.236(i)

14.1 Decommissioning Activities

Section 14 of the SAR discusses two options to remove spent fuel from the ISFSI for storage/disposal at another NRC approved location. The first option assumes that the NUHOMS®-24PT1-DSC can be transferred in a transportation package that meets the requirements of 10 CFR Part 71. The second option requires the spent fuel to be removed from the NUHOMS®-24PT1-DSC and transferred to another NRC approved transportation package.

The first option would require very little facility decommissioning. The Standardized Advanced NUHOMS® System has been designed and evaluated to maintain containment during normal, off-normal, and accident conditions. In addition, the NUHOMS®-24PT1-DSC and associated equipment are decontaminated before being moved from the fuel handling building to the ISFSI. Therefore, contamination at the ISFSI should be very low.

The second option would require the NUHOMS®-24PT1-DSC to be unloaded in a spent fuel pool or a dry transfer facility. Decontamination of the ISFSI would then be the same as the first option with the addition of decontamination and disposal of the empty DSC.

14.2 Evaluation Findings

The applicant's proposed cask design includes adequate provisions for decontamination and decommissioning. As discussed in Section 14 of the SAR, these provisions include facilitating decontamination of the Standardized Advanced NUHOMS® System, if needed; storing the remaining components, if no waste facility is expected to be available; and disposing of any remaining low-level radioactive waste.

Section 14 of the SAR presents information concerning the proposed practices and procedures for decontaminating the cask and disposing of residual radioactive materials after all spent fuel has been removed. This information provides reasonable assurance that the applicant will conduct decontamination and decommissioning in a manner that adequately protects the health and safety of the public.

The staff concludes that the decommissioning considerations for the Standardized Advanced NUHOMS® System are in compliance with 10 CFR Part 72. This evaluation provides reasonable assurance that the Standardized Advanced NUHOMS® System will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

15.0 CONCLUSIONS

15.1 Overall Conclusion

The staff performed a detailed safety evaluation of the application for a 10 CFR Part 72 CoC for the Standardized Advanced NUHOMS® System. The staff performed the review in accordance with the guidance in NUREG 1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997. Based on the statements and representations contained in the SAR and the conditions in the CoC, the staff concluded that the Standardized Advanced NUHOMS® System meets the requirements of 10 CFR Part 72.

15.2 Conclusions Regarding Analytical Methods

The staff determined that all analytical methods used by the applicant in the design of the Standardized Advanced NUHOMS® System, as described in the SAR, are acceptable with the following exceptions:

15.2.1 Shielding Methodology

The shielding analysis for the Advanced NUHOMS® system was performed with DORT, a 2-D discrete ordinates code used to calculate the dose rates on and around the AHSM and OS-197 TC. The staff has considered this 2-D code acceptable for this application under the following limitations:

- the utilization of already proven technology in the Standardized NUHOMS® System
- the thickness improvements in the concrete of the AHSM,
- the relatively low design basis source term, and
- the relatively long fuel cooling times of 10 and 20 years.

However, the staff determined that use of a 2-D code for complex design configurations, such as those associated with dry cask storage systems, may not accurately characterize all possible radiation dose levels. Therefore, for future amendment applications, and safety evaluations performed in accordance with 10 CFR 72.48, a 3-D shielding analysis that has been validated against actual data should be performed. A 2-D shielding model analysis that has been validated against actual data for the same or similar configuration (materials, dimensions, geometry, etc.) may also be used provided that the analysis demonstrates that it is conservative with respect to a 3-D analysis.

15.2.2 Thermal Evaluation Methodologies

15.2.2.1 Calculation of Insolation for Normal Conditions

The staff determined that the value used by the applicant for solar insolation was well below the value recommended in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," for insolation upon a flat surface and, therefore, non-conservative. The staff determined that the value used by the applicant was derived from a misapplication of the information provided in the American Society of Heating, Refrigerating, and Air Conditioning Engineers, Inc. (ASHRAE) 1981 Fundamentals Handbook. The applicant revised its analysis to meet the guidelines presented in NUREG-1536. The staff found the applicant's revised analysis acceptable, however, the staff considers the applicant's misapplication of ASHRAE values as non-conservative and should not be used in future amendment applications or safety evaluations performed in accordance with 10 CFR 72.48.

15.2.2.2 Calculation of Peak Clad Temperature

As discussed in Section 4.5.4.4 of this SER, the analysis performed by PNNL indicates that the fuel cladding temperature is within the calculated fuel temperature limits, and is therefore, acceptable. The staff has confidence in the analysis conducted by PNNL due to the fact that the COBRA-SFS code is a validated thermal code because it has shown through comparison, with available fuel temperature data, to be a best-estimate code

However, the staff determined that the applicant's methodology for calculating maximum fuel cladding temperatures, as presented in the SAR, is non-conservative for the following reasons:

- The applicant's model uses a homogenized region for the fuel assembly with an effective thermal conductivity or "smeared" property approach in their fuel assembly model. This method uses data from spent fuel assemblies to determine the effective thermal conductivity, taking into account radiation, convection, and conduction within the assembly. This model provides an average temperature for the fuel assembly and does not provide a peak cladding temperature for the hottest fuel rod within the assembly. The result reported by the applicant in the SAR is actually a maximum average temperature for the fuel assembly region.

- The applicant provided a limited number of nodes in the fuel assembly model, which does not accurately capture the temperature gradient that exists across the fuel assembly, nor capture the location of the hottest individual fuel rod.

- The applicant's DSC thermal model has not been validated against actual fuel temperature data applicable to the fuel assemblies to be stored and fill gas to be used in the 24 PT1- DSC. Therefore, the applicant's fuel assembly model cannot be considered reliable for predicting peak fuel cladding temperatures given the current fuel parameters.

Therefore, the staff concludes that the HEATING7 model may not be used for future amendment applications, and safety evaluations performed in accordance with 10 CFR 72.48, until it has been validated against actual data.

15.2.3 Criticality Evaluation Methodologies

15.2.3.1 Sensitivity Studies

The applicant performed sensitivity studies for various fuel parameters for the WE 14x14 SS304. The results show that k_{eff}= 0.8588 $\pm$ 0.0011 when nominal cladding thickness is used and k_{eff} = 0.8631 $\pm$ 0.0012 when minimum cladding thickness is used. Use of bounding tolerance values is consistent with the NUREG-1536, thus the staff disagrees with the applicant's use of nominal cladding thickness in the criticality models discussed below. However, the calculated k_{eff} for the most limiting normal condition for the NUHOMS 24PT1-DSC meets the upper subcritical limit (USL) of 0.9401 when increased to account for changes in k_{eff} due to cladding tolerance. While the most limiting accident condition k_{eff} would exceed the

USL, the staff has reasonable assurance that the accident scenarios, discussed in Section 6.3.1 of this SER are sufficiently conservative to bound this.

The staff determined for future amendment applications, and safety evaluations performed in accordance with 10 CFR 72.48, the applicant should use the bounding tolerance values as recommended by the NUREG-1536.

15.2.3.2 Material Properties

The applicant modeled the outer aluminum on the boral sheets as B_4C/Aluminum mixture rather than as aluminum. Staff calculations determined that modeling of the outer aluminum on the boral can cause a slight increase in the calculated k_{eff}, depending on the scenario modeled, and thus should be considered in any future amendments, and safety evaluations performed in accordance with 10 CFR 72.48.

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