72-1029

TRANSNUCLEAR
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Mr. L. Raynard Wharton
Spent Fuel Project Office, NMSS
U. S. Nuclear Regulatory Commission
11555 Rockville Pike M/S O13-D-13
Rockville, MD 20852

Subject: TN Review Comments on the Preliminary Certificate of Compliance and Safety Evaluation Report for the Standardized Advanced NUHOMS® System (Amendment No. 1 to CoC 1029, TAC No. L23606)

Reference: Preliminary Certificate of Compliance and Safety Evaluation Report for the Amended Standardized NUHOMS® System, dated October 7, 2004 (Amendment No. 1 to CoC 1029, TAC No. L23606)

Dear Mr. Wharton:

Enclosed herewith is a marked up copy of the reference document which reflects TN's review comments. Please note that only those pages with comments have been included herewith.

Should you or your staff require additional information to support review of this application, please do not hesitate to contact me at 510-744-6053.

Sincerely,

UBChopra

U. B. Chopra
Licensing Manager

Docket 72-1029

Enclosure: As stated

NMSS01

1.0 GENERAL INFORMATION

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

1.1 General Description and Operational Features

(including reconstituted)

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

unlimited number of

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The NUHOMS® 24PT4-DSC system consists of two different basket configurations. These configurations differ in the boron loading in the Boral® plates. Type A is the designation for the standard loading basket, Type B for the high loading basket. Minimum Boron-10 concentration for Type A is 0.025 g/cm^2, and for Type B is ~~0.069~~ g/cm^2. Fuel to be stored in these baskets is limited to an initial ^{235}U enrichment of 4.1 wt. % for the Type A basket, and to an initial ^{235}U enrichment of 4.85 wt.% for the Type B basket.

and poison rodlets

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

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

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

1.2 Drawings

Section A.1 of the SAR contains the non-proprietary drawings for the NUHOMS® 24PT4-DSC, including drawings of the structures, systems and components (SSC) important to safety. The staff determined that the drawings contain sufficient detail on dimensions, materials, and specifications to allow for a thorough evaluation of the NUHOMS® 24PT4-DSC. Specific SSC are evaluated in other sections of this SER.

1.3 DCSS Contents

(including reconstituted)

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

(~~unlimited~~ any number per assembly)

1.4 Qualification of the Applicant

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

1.5 Evaluation of Findings

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

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

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

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

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

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

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

2.0 PRINCIPAL DESIGN CRITERIA

The objective of evaluating the principal design criteria related to the SSC important to safety is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72 (Ref. 1).

2.1 Structures, Systems, and Components Important to Safety

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

2.2 Design Basis for SSCs Important to Safety

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

2.2.1 Spent Fuel Specifications

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

The NUHOMS® 24PT4-DSC system consists of two different basket configurations. These configurations differ in the boron loading in the Boral® plates. Type A is the designation for the standard loading basket, Type B for the high loading basket. Minimum Boron-10 concentration for Type A is 0.025 g/cm², and for Type, it is ~~0.009~~ g/cm². Fuel to be stored in the Type A basket is limited to an initial ^{235}U enrichment of 4.1 wt. %, and is limited to 4.85 wt.% ^{235}U for Type B.

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2.2.2 External Conditions

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

2.3 Design Criteria for Safety Protection Systems

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

3.0 STRUCTURAL EVALUATION

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

The review was conducted against the appropriate regulations as described in 10 CFR 72.11, 10 CFR 72.122, and 10 CFR 72.236. The structural evaluation shows that the 24PT4-DSC design is compatible with requirements of 10 CFR 72.236 for maintaining the spent fuel in a subcritical condition, providing adequate radiation shielding and confinement, having adequate passive heat removal capability, providing a redundant sealing of the confinement system, and providing wet or dry transfer capability.

3.1 Structural Design of the Dry Storage Canister 24PT4-DSC (Type 304) (Type 316)

The 24PT4-DSC canister assembly is made of several steel structural components and the related weld filler metal, all of which are important to safety. The individual structural components are as follows: for the canister shell assembly - a cylindrical shell, the top and bottom cover plates, and shield plug assemblies; for the internal basket assembly - 24 stainless steel guide sleeve assemblies, 28 carbon steel spacer discs, and four-support rod/spacer sleeve assemblies. The 24PT4-DSC shell assembly's top and bottom ends include stainless steel forgings and ~~Type 316 stainless steel~~ plates that encase the lead (ASTM B29) shield plugs. The cylindrical shell is fabricated from SA240, Type 316 stainless steel. ASME SA-533 Grade B Class 1 carbon steel material is used for fabrication of the 24PT4-DSC basket assembly spacer discs. This is different from the SA-537, Class 2 used for the 24PT1-DSC spacer discs in order to accommodate the higher disc temperatures. ASME Code Case N-499- 1 3 ~~13~~ provides the basis for limited elevated temperature service up to 1000°F for the SA-533 Grade B Class 1 carbon steel and the material properties are shown in SAR Table A.3.3-2. The support rod and spacer sleeves are fabricated of SA-479, Type XM-19 stainless steel. Damaged fuel assemblies are stored in Failed Fuel Cans. Failed Fuel Cans are provided with screens at the top and bottom to contain fuel debris and allow water to fill or drain from the can. The Failed Fuel Can and the guide sleeves are all fabricated from SA-240, Type 304 stainless steel.

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

During ~~fabrication, leak tests of the 24PT4-DSC~~ shell assembly are performed in accordance with ANSI N14.5-1997 to demonstrate that the shell is leaktight. The 24PT4-DSC inner top closure welds, including the vent and siphon block subassembly welds, is leak-tested after fuel loading to demonstrate that ANSI N14.5 leaktight criteria are met.

3.2 Materials

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The applicant provided a general description of the materials of construction in SAR Sections A.1.2, A.3.1, and A.3.3. Additional information regarding the materials, fabrication details and testing programs can be found in SAR Section A.9.1. The staff reviewed the information contained in these sections; Table A.3.1-5, ASME Code Alternatives and the information presented in the SAR drawings, to determine whether the NUHOMS 24PT4-DSC meets the requirements of 10 CFR 72.24(c) (3) and (4), 72.122(a), (b), (h) and (l), and 72.236(g) and (h). In particular, the following aspects were reviewed: materials selection; brittle fracture; applicable codes and standards; weld design and specifications; and chemical and galvanic corrosion.

3.2.1 Structural Materials

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

3.2.2 Nonstructural Material

guide sleeve assemblies

The basket assembly structure consist t of ~~a grid assembly of~~ welded stainless steel ~~plates or tubes~~ that make up the fuel compartments. Each fuel compartment accommodates neutron absorber ~~absorbing plates~~ for criticality control. The neutron absorber ~~plate~~ for criticality control ~~are~~ is either Boral or boron carbide encapsulated in stainless steel tubes. In accordance with Section A.9.1, appropriate acceptance testing will be used to ensure that the neutron absorbers have the minimum specified ^{10}B loading.

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

Boral plus

The DSC top closure welds are performed in accordance with

3.2.3 Welds

The DSC cylindrical shell is assembled using full penetration longitudinal welded joints and circumferential welded joints at the junction between the inner bottom plate and the shell. These welds are performed in accordance with ASME Code, Section III, Subsection NB. ~~and~~ ASME Code Case N595-1.

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

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

3.2.4 Bolting Materials

The DSC is an all-welded canister.

3.2.5 Coatings

No zinc, zinc compounds, or zinc-based coatings are used on the ~~carbon~~ austenitic steel, encased top shield plug of the DSC. The carbon steel spacer discs will be coated with an electroless nickel plating which has been used in the nuclear industry. The coating will protect the steel from excessive oxidation of the surface.

3.2.6 Mechanical Properties

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

3.2.7 Chemical and Galvanic Reactions

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

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

3.3 Normal and Off-Normal Conditions

3.3.1 Analysis Methods

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

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

3.3.2 Loading Conditions Analyzed

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

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

3.3.3 Analysis Results

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

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

3.4 Accident Conditions

3.4.1 Analysis Methods

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

Stress analyses of the guide sleeve assemblies are performed using a combination of closed-form calculations and finite element analyses using the same ANSYS finite element model of the guidesleeve used in the normal and off-normal loading analyses. Elastic-plastic analyses the bounding 24PT1-DSC 2467.5

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

perpendicular to

3.4.2 Loading Cases Analyzed

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

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

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

3.4.3 Analysis Results

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

Tables A.3.6-5 and A3.6-6 of the SAR provide the summary results for the enveloping loading cases for the accident load conditions (i.e., Service Levels C and D loadings) for the 24PT4-DSC shell assembly. The calculated maximum stress intensities for the various components of the shell assembly under accident loading condition are provided in Table A.3.6-3. Tables 3.6-

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

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

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

3.5 Evaluation Findings

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

F3.2 The 24PT4-DSC is designed to allow handling and retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage to the DSC that will prevent retrieval of the stored spent nuclear fuel.

F3.3 The 24PT4-DSC is designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions. The configuration of the stored spent fuel is unchanged. Additional criticality evaluations are discussed in Section 6 of this SER.

F3.4 The 24PT4-DSC is evaluated to demonstrate that it has a redundant seal and that it will maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

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

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

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

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

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

F3.10 The staff concludes that the structural design of the 24PT4-DSC is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that addition of a 24PT4-DSC in the Standardized Advanced NUHOMS® System will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable industry codes and standards, accepted practice and confirmatory analysis.

3.6 References

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

2. Transnuclear, Inc, Final Safety Analysis Report (FSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision ~~0~~ 8, ~~October~~ June ~~2001~~ 2004, USNRC Docket Number 72-1004.

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

in addition

4.0 THERMAL EVALUATION

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

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

or ZIRLO

4.1 Spent Fuel

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

4.1.1 Spent Fuel Cladding

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

The staff reviewed the discussion on material temperature limits with respect to the following regulatory requirements:

- 10 CFR §72.122(h)(1) requires the spent fuel cladding to be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate.

For the PWR fuel assemblies, the allowable temperature limits are based on Interim Staff Guidance No. 11, Rev. 3 (ISG-11) (U.S. Nuclear Regulatory Commission, November, 2003). For normal conditions (long-term) of storage and short-term fuel loading and storage operations (which includes drying, backfilling with inert gas, and transfer of the cask to the storage pad), the temperature limit of the fuel cladding is maintained below 400°C. This is done to ensure that circumferential hydrides in the cladding will not dissolve and go into solution during fuel loading operations, and that re-precipitation of radial hydrides does not occur in the cladding during storage (see ISG-11, Rev. 3 for a discussion on hydride reorientation). The applicant established a temperature limit of 570°C (1058°F) for off-normal and hypothetical accident conditions for Zircaloy fuel cladding and invoked the thermal cycling criteria contained in ISG-11. Fuel with burnup greater than 60 GWd/MTU is unacceptable for storage in the 24PT4-DSC system. ~~Fuel cladding types are limited to Zircaloy. Zirconium based alloys (i.e., M5, Optin, ZIRLO, etc.) are unacceptable for storage in the 24PT4-DSC system at this time.~~

4.2 Cask System Thermal Design

4.2.1 Design Criteria

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

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

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

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

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

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

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

4.2.2 Design Features

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

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

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

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

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

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

4.3 Thermal Load Specifications

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

4.3.1 Storage Conditions

Initially, the applicant provided thermal analyses based on the 24PT1 methodology (previously submitted and evaluated by the staff) to demonstrate that the system would perform satisfactorily for normal, off normal and accident conditions. The staff had concerns with these analyses (as stated in a formal request for additional information), particularly with regard to the models' capability to predict temperature distributions and convection flow patterns. To address these concerns, the applicant withdrew the 24PT1-based analyses and performed new analyses using a robust computational fluid dynamics (CFD) code.

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

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

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

4.3.2 Normal Conditions

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

The *minimum normal storage condition considers* a 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~~ 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 4.4.3 as siting parameters that must be evaluated by the storage system user.

4.3.3 Off-Normal Conditions

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

4.3.4 Accident Conditions

4.3.4.1 Blocked Vents

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

4.3.4.2 Transfer Cask Loss of Neutron Shield and Sunshade

The second accident condition postulated by the applicant is a loss of water neutron shield in the annular region of the OS-197H transfer cask, as well as a loss of the required sunshade during transfer operations ~~at the~~ extreme off-normal ambient temperature condition of 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 above.

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

4.3.4.3 Fire

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

A 15 *minute fire with an average flame temperature of* 1475°F (800°C), *an average convective* heat transfer coefficient of 5.21E-4 Btu/min-in²-°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

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

4.4 Model Specification

4.4.1 Configuration

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

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

4.4.1.1 AHSM Model

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

4.4.1.2 24PT4- DSC Basket Section/Fuel Assembly Model

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

4.4.1.3 24PT4-DSC in Transfer Cask Model

This model is described in SAR Section A.4.4.3. The TC model developed by the applicant to simulate the DSC within the TC is a two-dimensional axisymmetric model which includes the DSC shell assembly, and the DSC cavity modeled as a homogenous region. The model is based on an analysis conducted and submitted in the 72-1004 SAR (Revision X dated ~~November 2001~~). The maximum DSC shell temperature is extracted from this model and used in the DSC basket analysis. The DSC 3D slice model described above was also used for the TC/DSC analyses.

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4.4.2 Material Properties

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

4.4.3 Boundary Conditions

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

4.4.3.1 Accident Conditions - Blocked Vent

For the postulated blocked vent accident conditions, the HEATING7 AHSM model described in SAR Section 4.4.2.2 for the storage condition was used, and the inlet and outlet vents were blocked. ~~The~~ boundary conditions include the DSC off-normal condition temperature distribution before the postulated accident, and ambient temperatures and insolation as outlined in Table 4-~~2~~ 1 (above).

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

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

4.4.3.3 Accident Conditions-Fire

The postulated fire accident conditions and the model of the DSC in the transfer cask are described in SAR Section A.4.6.4. The boundary conditions for the fire accident are described in Section ~~4.3.6~~ 4.3.4.3 above. The boundary conditions include the DSC and transfer cask normal condition temperature distribution before the postulated fire and the maximum off-normal ambient ~~conditions~~ after the fire.

4.4.3.4 Cask Heatup Analysis

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

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

4.5 Thermal Analysis

4.5.1 Temperature Calculations

4.5.1.1 Storage Conditions

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

4.5.1.2 Accident Conditions- Blocked Vents

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

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Table 4-~~3~~ 2

Temperatures of Key Components in the Advanced NUHOMS® Storage System[1]

Component	Normal Storage Conditions		Transfer Condition (off normal)	Normal Allowable Range (°F)	Accident Conditions	
	Maximum (°F)	Minimum[2] (°F)	Maximum (°F)		Maximum (°F)	Allowable Range (°F)
AHSM Concrete	232	0	N/A	0 to 300	~~3923~~	-40 to 300
AHSM Support Steel	281	0	N/A	0 to 800	615	-40 to 800
AHSM Heat Shield	314	0	N/A	0 to 800	542	-40 to 800
DSC Shell	459	0	443	0 to 800	642	-40 to 800
Guidesleeve or Can	662	0	675	0 to 800	845	-40 to 900[4]
DSC Oversleeve	662	0	675	0 to 800	845	-40 to 900[4]
DSC Spacer Disk	653	0	668	0 to 700	836	-40 to 1000
DSC Support Rod/Spacer Sleeve	560	0	580	0 to 800	738	-40 to 800[4]
DSC Boral® Sheet	662	0	675	0 to 850	845	-40 to 1000
Zircaloy Cladding	697	0	712	0 to 752	880	-40 to 1058

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

Extreme ambient conditions were used as boundary conditions for this analysis. The analysis included a heat source of 24 kW for qualification of the AHSM concrete and the 24PT4-DSC. Maximum DSC shell temperature and concrete temperatures were obtained in this analysis. None of the components of the DSC exceeded their temperature limits. The maximum concrete temperature reported was above the limit specified by the applicant. The applicant has committed to testing the ~~concrete~~ used to fabricate the AHSM at an elevated temperature to demonstrate that the concrete will perform satisfactorily. The results for this accident analysis are summarized in Table 4-~~3~~ 2 above. Based on this analysis and the TS, the staff finds reasonable assurance that the fuel cladding integrity and the confinement boundary will not be compromised during the blocked vent transient.

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

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

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

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

4.5.1.4 Accident Conditions - Fire

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

4.5.1.5 Cask Heatup Analyses

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

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

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

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

4.5.2 Pressure Analysis

4.5.2.1 Storage/Off Normal/Accident Conditions

In SAR Section A.4.4.8, the applicant evaluated internal pressurization for normal conditions. The applicant assumed a fully loaded DSC. A 1% failure of fuel rods and control components is assumed. For the ruptured rods, a 100 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 17.5 psig, which is below the applicant's criteria of 20 psig for normal conditions.

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

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

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

The staff reviewed the applicant's calculations and determined that the applicant's calculations used appropriate meth~~ods and~~ cover gas temperatures determined in SAR Section A.4. The highest predicted pressure was 80.7 psig at a cavity gas temperature of 713°F for the accident condition, which is below the DSC thermal criteria pressure of 81 psig. (The pressure used in the stress analysis was ~~90~~ 100 psig as stated in Table A.3.1-~~6~~ 4.)

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

4.5.2.2 Pressure During Unloading of Cask

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

4.5.2.3 Pressure During Loading of Cask

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

4.5.3 Confirmatory Analyses

The staff reviewed the initial approaches used by the applicant in the thermal analyses. The results of these analyses were compared to the detailed confirmatory analyses (see Sections ~~4.5.4.1~~ below) conducted by Pacific Northwest National Labs (PNNL) for the staff. The staff concluded (as stated in the formal Request for Additional Information) that some of the applicant's models were not robust enough to accurately predict the temperature distributions in the system.

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

4.5.3.1 and 4.5.3.2

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

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

5.3 Shielding Model Specifications

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

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

5.3.1 Shielding and Source Configuration

The shielding source is divided into four axial regions: bottom nozzle, in-core, plenum, and top nozzle. The lengths of these regions are specified in Table ~~5.2-1~~ A.5.2-1. The source regions were homogenized and cross-sectional area was preserved.

5.3.2 Material Properties

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

5.3.3 Staff Evaluation

The staff reviewed the shielding models and found them acceptable. The material compositions and densities used were appropriate and provide reasonable assurance that the

6.0 CRITICALITY EVALUATION

The staff reviewed Amendment 1 to the Standardized Advanced NUHOMS® System criticality analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the following regulatory requirements are met: 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g) (Ref.1). The staff also reviewed the SAR to determine whether the cask system was consistent with the following acceptance criteria listed in Section 6 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" (Ref. 2):

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

6.1 Criticality Design Criteria and Features

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

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

The fuel assemblies are placed in baskets with square fuel cells and Boral® panels fixed to the fuel cell walls. The 24PT4-DSC may contain up to 12 damaged fuel assemblies and up to 5 poison rodlets per undamaged fuel assembly. TS 4.2.3 has been revised to describe the two alternate minimum ^{10}B areal density specifications for the 24PT4-DSC Boral® panels: a standard loading of 0.025 g/cm^2 and a high loading of 0.068 g/cm^2. TS 4.2.3 also references Table 2-8 of the TS which shows the maximum fuel enrichment versus minimum Boral® panel areal density and number of poison rodlets required for various configurations of damaged and intact fuel assemblies. Damaged fuel assemblies contained in failed fuel cans must be loaded in the basket as shown in TS Figure 2-4. The applicant stated that 75 percent credit was taken for the minimum ^{10}B content in the Boral® panels, and 64 percent credit was taken for the minimum ^{10}B content in the B_4C poison rodlets.

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

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

Additionally, the staff verified that the design-basis off-normal and postulated accident events would not have an adverse effect on the design features important to criticality safety. Section A.3 of the SAR shows that the basket will remain intact during all normal, off-normal, and accident conditions. Based on the information provided in the SAR, the staff concludes that the Standardized Advanced NUHOMS® System design with the 24PT4-DSC meets the double contingency requirements of 10 CFR 72.124(a).

6.2 Fuel Specification

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

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

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

Specifications on the condition of the fuel are also included in the SAR and TS. The 24PT4-DSC is designed to accommodate intact fuel assemblies or up to 12 damaged fuel assemblies (depending on the minimum ^{10}B areal density of the Boral® panels and the number of B_4C poison rodlets included with the intact assemblies), as defined in the TS. The damaged fuel must be placed in individual failed fuel cans, which are designed to confine gross fuel particles to a known, subcritical geometry. Up to 12 failed fuel cans may be placed in the perimeter basket locations of the 24PT4-DSC, as shown in TS Figure 2-4. Reconstituted fuel assemblies, with up to 8 rods replaced with stainless steel rods or any number of rods replaced with zirconium-clad uranium rods, may be stored as intact or damaged assemblies.

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

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

6.3 Model Specification

6.3.1 Configuration

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

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

The calculation models also conservatively assumed the following:

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

#5

7.4 Confinement Analysis

TN has demonstrated that the welds and applicable non-destructive examinations meet the applicable requirements demonstrating DSC integrity as set forth in ~~ISG-1, Rev. 2, and~~ ISG-18, as follows:

1. The DSC is fabricated from austenitic stainless steel;

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

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

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

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

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

7.5 Supportive Information

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

7.6 Evaluation Findings

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

F7.2 The design of the NUHOMS® 24PT4-DSC adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the SER discusses the relevant temperature considerations.

F7.3 The design of the NUHOMS® 24PT4-DSC provides redundant sealing of the confinement system closure joints using dual welds on the canister lid and closure.

8.2 Cask Handling and Storage Operations

All handling and transportation events applicable to moving the NUHOMS® 24PT4-DSC to the storage location are similar to those previously reviewed by the staff for the Standardized Advanced NUHOMS® System are bounded by Section A.11 of the SAR. Monitoring operations include surveillance of the AHSM air inlets and outlets in accordance with TS 5.2.5, and temperature performance in accordance with TS 5.2.5. Occupational and public exposure estimates are evaluated in Section A.10 of the SAR. Each cask user will need to develop detailed cask handling and storage procedures that incorporate ALARA objectives of their site-specific radiation protection program.

8.3 Cask Unloading

Detailed unloading procedures must be developed by each user.

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

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

8.4 Evaluation Findings

F8.1 The NUHOMS® 24PT4-DSC is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Section A.8 of the applicant's SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

DSC

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

F8.3 The DSC geometry and general operating procedures facilitate decontamination. Only routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F8.4 No significant radioactive waste is generated during operations associated with the independent spent fuel storage installation (ISFSI). Contaminated water from the spent fuel pool will be governed by the 10 CFR Part 50 license conditions.

F8.5 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading will be governed by the 10 CFR Part 50 license conditions.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAMS

9.1 Acceptance Tests

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

9.1.1 Visual and Nondestructive Examination Inspections

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

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

9.1.2 Leakage Testing

Nqe .

The NUHOMS® 24PT4-DSC is designed and tested to be leaktight, as specified in "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5-1997 (Ref. 1).

9.1.3 Neutron Absorber Tests

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

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

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

9.2 Evaluation Findings

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

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

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

F9.4 The staff concludes that the acceptance tests and maintenance program for the 24PT4-~~DCS~~ DSC are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the acceptance tests and maintenance program provides reasonable assurance that the cask will allow safe storage of spent fuel throughout its licensed or certified term. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

9.3 References

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

None

11.0 ACCIDENT ANALYSES

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

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

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

11.1 Off-Normal Operations

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

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

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

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

11.2 Accident -Level Events and Conditions

The application states that the discussion in Section 11.2 of the FSAR for the 24PT1-DSC also applies to 24PT4-DSC. The earthquake stress evaluations for the AHSM presented in Sections 3.6 and 11.2.1 for the 24PT1-DSC were based on a bounding weight of 85,000 lbs. Thus, the analysis results presented in the FSAR are applicable to the AHSM loaded with a 24PT4-DSC. For tornado wind, tornado missile, and flooding loading conditions, the calculated overturning, sliding, and missile impact analyses are bounding for the AHSM loaded with a 24PT4-DSC because the stabilizing moment against overturning and the force required to slide the AHSM were based on the slightly lower weight of the 24PT1-DSC. The applicant performed a new

Table 12-1

Standardized Advanced NUHOMS® Horizontal Modular Storage System
Technical Specifications for use with the NUHOMS® 24PT4-DSC

Operating Controls and Limits

1.0 Use and Application
- 1.1 Definitions
- 1.2 Logical Connectors
- 1.3 Completion Times
- 1.4 Frequency

2.0 Functional and Operational Limits
- 2.1 Fuel to be Stored in the NUHOMS® 24PT1-DSC
- 2.2 Fuel to be Stored in the NUHOMS® 24PT4-DSC
- 2.3 Functional and Operating Limits Violations

3.0 Limiting Condition for Operation (LCO) and Surveillance Requirement (SR) applicability
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  - 3.1.2.b NUHOMS® 24PT4-DSC Helium Backfill Pressure

4.0 Design Features
- 4.1 Site
  - 4.1.1 Site Location
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4.4.1. Storage Configuration
4.4.2 Concrete Pad Properties to Limit DSC Gravitational Loading Due to Postulated Drops
4.4.3 Site Specific Parameters and Analyses.

NRC

APPENDIX A TO CERTIFICATE OF COMPLIANCE NO. 1029

PRELIMINARY
TECHNICAL SPECIFICATIONS FOR THE ADVANCED NUHOMS® SYSTEM

OPERATING CONTROLS AND LIMITS

TABLE OF CONTENTS

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1.1 Definitions

-- NOTE --

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required ACTIONS to be taken under designated Conditions within specified Completion Times.
ADVANCED HORIZONTAL STORAGE MODULE (AHSM)	The AHSM is a reinforced concrete structure for storage of a loaded 24PT1-DSC or 24PT4-DSC (DSC) at a spent fuel storage facility
DAMAGED FUEL ASSEMBLY	A DAMAGED FUEL ASSEMBLY is a FUEL ASSEMBLY with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial or missing rods.
DRY SHIELDED CANISTER (DSC)	A 24PT1-DSC or 24PT4-DSC is a welded pressure vessel that provides confinement of INTACT or DAMAGED FUEL ASSEMBLIES in an inert atmosphere.
FAILED FUEL CAN	A FAILED FUEL CAN confines any loose material and gross fuel particles to a known, subcritical volume during normal, off-normal and accident conditions and facilitates handling and retrievability.
FUEL DEBRIS	An intact or partial fuel rod not contained in a FUEL ASSEMBLY grid or an individual intact or partial fuel pellet not contained in a fuel rod. FUEL DEBRIS may be inserted in a ROD STORAGE BASKET.
INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within a perimeter fence licensed for storage of spent fuel within AHSMs.

INTACT FUEL ASSEMBLY	Spent Nuclear FUEL ASSEMBLIES without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means.
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a DSC while it is being loaded with INTACT or DAMAGED FUEL ASSEMBLIES, and on a TRANSFER CASK while it is being loaded with a DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. LOADING OPERATIONS begin when the first INTACT or DAMAGED FUEL ASSEMBLY is placed in the DSC and end when the TRANSFER CASK is ready for TRANSFER OPERATIONS.
RECONSTITUTED FUEL ASSEMBLY	RECONSTITUTED FUEL ASSEMBLIES include assemblies in which leaking fuel rods are replaced with either stainless steel rods or intact fuel rods prior to return to the reactor. RECONSTITUTED FUEL ASSEMBLIES may contain from one to eight stainless steel rods per assembly.
ROD STORAGE BASKET	A 9x9 array of tubes in a lattice that has approximately the same dimensions as a standard FUEL ASSEMBLY.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI while a DSC containing INTACT or DAMAGED FUEL ASSEMBLIES is located in an AHSM on the storage pad within the ISFSI perimeter.
TRANSFER CASK (TC)	The TRANSFER CASK will consist of a licensed NUHOMS® OS197 or OS197H onsite transfer cask. The TRANSFER CASK will be placed on a transfer trailer for movement of a DSC to the AHSM.
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all licensed activities involving the movement of a TRANSFER CASK loaded with a DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. TRANSFER OPERATIONS begin when the TRANSFER CASK is placed on the transfer trailer following LOADING OPERATIONS and end when the DSC is located in an AHSM on the storage pad within the ISFSI perimeter.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a DSC to unload INTACT or DAMAGED FUEL ASSEMBLIES. UNLOADING OPERATIONS

begin when the DSC is removed from the AHSM and end when the last INTACT or DAMAGED FUEL ASSEMBLY has been removed from the DSC.

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors. Logical connectors are used in Technical Specifications (TS) to

discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

para

When logical connectors are used to state a Condition, Completion Time,

delete space

Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors:

space

EXAMPLE 1.2-1:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Remove . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO are not met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability. Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions:

EXAMPLE 1.3-1:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2	12 hours 36 hours

delete space

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

EXAMPLES (continued)

EXAMPLE 1.3-2:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2.	12 hours 36 hours

When a system is determined to not meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

EXAMPLES (continued)

EXAMPLE 1.3-3:

ACTIONS

-------------------------NOTE-------------------------

Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The "Specified Frequency" is referred to throughout this section and each of the Specifications of Section 12.3, Surveillance Requirement (SR) Applicability. The "Specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 12.3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With a SR satisfied, SR 12.3.0.4 imposes no restriction.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified:

EXAMPLE 1.4-1:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify Pressure within limit.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 12.3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 12.3.0.1 (such as when the equipment is determined to not meet the LCO, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 12.3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 12.3.0.1.

If the interval as specified by SR 12.3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 12.3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 12.3.0.4.

EXAMPLES (continued)

EXAMPLE 1.4-2:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one-time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 12.3.0.2.

"Thereafter" indicates future performances must be established per SR 12.3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

EXAMPLES (continued)

EXAMPLE 1.4-3:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------NOTE------ Not required to be met Not required to be met until 96 hours after verifying the helium leak rate is within limit. Verify 24PT1-DSC vacuum drying pressure is within limit.	6 Once after verifying the helium leak rate is within limit.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the vacuum drying pressure not be met immediately following verification of the helium leak rate while in LOADING OPERATIONS, this Note allows 96 hours to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency."

Once the helium leak rate has been verified to be acceptable, 96 hours, plus the extension allowed by SR 12.3.0.2, would be allowed for completing the Surveillance for the vacuum drying pressure. If the Surveillance was not performed within this 96 hour interval, there would then be a failure to perform the Surveillance within the specified Frequency, and the provisions of SR 12.3.0.3 would apply.

2.0 Functional and Operating Limits

2.1 Fuel To Be Stored In The 24PT1-DSC

The spent nuclear fuel to be stored in each 24PT1-DSC/AHSM at the ISFSI shall meet the following requirements:

a. Fuel shall be INTACT FUEL ASSEMBLIES or DAMAGED FUEL ASSEMBLIES. DAMAGED FUEL ASSEMBLIES shall be placed in screened confinement cans (FAILED FUEL CANS) inside the 24PT1-DSC guidesleeves. DAMAGED FUEL ASSEMBLIES shall be stored in outermost guidesleeves located at the 45, 135, 225 and 315 degree azimuth locations.

b. Fuel types shall be limited to the following:

   UO_2 Westinghouse 14x14 (WE 14x14) Assemblies (with or without IFBA fuel rods), as specified in Table ~~12.2-1~~ 2-1.

   WE 14x14 Mixed Oxide (MOX) Assemblies, as specified in Table ~~12.2-1~~ 2-1

   Fuel burnup and cooling time is to be consistent with the limitations specified in Table ~~12.2-4~~ 2-4 for UO2 fuel.

   Control Components stored integral to WE 14x14 Assemblies in a 24PT1-DSC, shall be limited to Rod Cluster Control Assemblies (RCCAs), Thimble Plug Assemblies (TPAs), and Neutron Source Assemblies (NSAs). Location of control components within a 24PT1-DSC shall be selected based on criteria which does not change the radial center of gravity by more than 0.1 inches.

c. The maximum heat load for a single FUEL ASSEMBLY, including control components, is 0.583 kW for SC FUEL ASSEMBLIES and 0.294 kW for MOX FUEL ASSEMBLIES. The maximum heat load per 24PT1-DSC, including any integral Control Components, shall not exceed 14 kW when loaded with all SC FUEL ASSEMBLIES and 13.706 kW when loaded with MOX FUEL ASSEMBLIES.

d. Fuel can be stored in the 24PT1-DSC in any of the following configurations:

   1) A maximum of 24 INTACT WE 14x14 MOX or SC FUEL ASSEMBLIES; or
   2) Up to four WE 14x14 SC DAMAGED FUEL ASSEMBLIES, with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES; or
   3) One MOX DAMAGED FUEL ASSEMBLY with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES.

   A 24PT1-DSC containing less than 24 FUEL ASSEMBLIES may contain dummy FUEL ASSEMBLIES in FUEL ASSEMBLY slots. The dummy FUEL ASSEMBLIES are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a FUEL ASSEMBLY. The effect of dummy assemblies or empty FUEL ASSEMBLY slots on the radial center of gravity of the DSC must meet the requirements of Section 2.1.b.

No more than two empty FUEL ASSEMBLY slots are allowed in each DSC. They must be located at symmetrical locations about the 0-180° and 90-270° axes.

No more than 14 fuel pins in each assembly may exhibit damage. A visual inspection of assemblies will be performed prior to placement of the fuel in the 24PT1-DSC, which may then be placed in storage or transported anytime thereafter without further fuel inspection.

e. Fuel dimensions and weights are provided in Table 2-2.

f. The maximum neutron and gamma source terms are provided in Table 2-3.

2.2 Fuel to Be Stored in the 24PT4-DSC

a. The spent fuel to be stored in the NUHOMS® 24PT4-DSC consists of INTACT (including RECONSTITUTED) Westinghouse-CENP 16x16 (CE 16x16) and/or DAMAGED CE 16x16 FUEL ASSEMBLIES with Zircaloy or ZIRLO™ cladding and UO_2, $(U, Er)O_2$ or $(U, Gd)O_2$ fuel pellets. Assemblies are with or without integral burnable poison rods or integral fuel burnable absorber (IFBA) rods.

b. Each 24PT4-DSC can accommodate a maximum of 12 DAMAGED FUEL ASSEMBLIES, with the remaining assemblies being intact.

RECONSTITUTED ASSEMBLIES containing up to eight replacement stainless steel rods in place of DAMAGED FUEL Rods or replacement Zircaloy clad uranium rods (any number per assembly) are acceptable for storage in the 24PT4-DSC as either INTACT or DAMAGED ASSEMBLIES.

DAMAGED FUEL may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). DAMAGED FUEL ASSEMBLIES shall be encapsulated in individual FAILED FUEL CANS placed in locations as shown in Figure 2-4.

FUEL DEBRIS and DAMAGED FUEL Rods that have been removed from a DAMAGED FUEL ASSEMBLY and placed in a ROD STORAGE BASKET are also considered as DAMAGED FUEL. A ROD STORAGE BASKET is a 9x9 array of tubes in a lattice that has approximately the same dimensions as a standard FUEL ASSEMBLY. ROD STORAGE BASKETS may also include IFBA and Integral Burnable Poison Rods. Loose FUEL DEBRIS not contained in a ROD STORAGE BASKET may be placed in a FAILED FUEL CAN for storage provided the size of the debris is larger than the FAILED FUEL CAN screen mesh opening. FUEL DEBRIS may be associated with any type of UO_2 fuel provided that the maximum uranium content and enrichment limits are met.

c. The INTACT and/or DAMAGED CE 16x16 FUEL ASSEMBLIES acceptable for storage in 24PT4-DSC are specified in Table 2-5, Table 2-6, and Table 2-7. The fuel to be stored in the 24PT4-DSC is limited to a maximum initial enrichment of 4.85 wt. % ^{235}U. The maximum allowable assembly burnup is given as a function of initial fuel enrichment but does not exceed 60,000 MWd/MTU. The minimum cooling time is 5 years.

d. A 24PT4-DSC containing less than 24 FUEL ASSEMBLIES may contain dummy FUEL ASSEMBLIES in FUEL ASSEMBLY slots, or empty slots. The dummy FUEL ASSEMBLIES are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a FUEL ASSEMBLY.

e. The 24PT4-DSC may store PWR assemblies in any one of the three alternate configurations shown in Figure 2-1 through Figure 2-3 with a maximum heat load of 1.26 kW per assembly and a maximum heat load of 24 kW per DSC. Table 2-9 through Table 2-12 define the FUEL ASSEMBLY cooling time (in years) based on FUEL ASSEMBLY burnup and initial fuel enrichment for the assembly, assuming that no reconstituted fuel with stainless steel rods is present. The fuel qualification tables to be used for reconstituted assemblies with stainless steel rods are provided in Table 2-13 through Table 2-16. These tables ensure that the FUEL ASSEMBLY decay heat load is less than that specified for each table and that the corresponding radiation source term is bounded by that analyzed in Chapter A.5.

f. Two different 24PT4-DSC basket configurations are provided as shown in Table 2-8. These configurations differ in the boron loading in the Boral® plates. The minimum areal boron –10 (^{10}B) concentrations for the standard (Type A basket) and high (Type B basket) loadings are 0.025 and 0.068 g/cm^2, respectively. Fuel to be stored in the standard ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 4.1 wt. %. Fuel to be stored in the high ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 4.85 wt. %.

g. Up to four DAMAGED FUEL ASSEMBLIES may be stored in a 24PT4-DSC of either ^{10}B loading without impact upon the maximum allowed ^{235}U enrichment and without the use of additional poison rodlets. The DAMAGED ASSEMBLIES shall be stored in FAILED FUEL CANS located at the 45, 135, 225 and 315 degree azimuth locations (Zone A of Figure 2-4).

Five to twelve DAMAGED FUEL ASSEMBLIES may be stored in a 24PT4-DSC of either ^{10}B loading without the use of poison rodlets if the maximum allowed ^{235}U enrichment is reduced for the DAMAGED ASSEMBLIES. The intact assembly enrichment limits remain at their nominal values of 4.1 and 4.85 wt. % for the standard and high ^{10}B loadings, respectively. DAMAGED FUEL to be stored in the standard ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 3.7 wt. %, and DAMAGED FUEL to be stored in the high ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 4.1 wt. %. All DAMAGED ASSEMBLIES shall be stored in FAILED FUEL CANS located in Zones A and B of Figure 2-4.

Five to twelve DAMAGED FUEL ASSEMBLIES may be stored in a 24PT4-DSC

of either ^{10}B loading without impact upon the maximum allowed ^{235}U enrichment if poison rodlets are utilized. For the standard ^{10}B loading, a single poison rodlet is inserted into the center guide tube of each INTACT FUEL ASSEMBLY located in Zone C of Figure 2-4. For the high ^{10}B loading, a poison rodlet is inserted into each of the five guide tubes in each INTACT FUEL ASSEMBLY located in Zone C of Figure 2-4. All DAMAGED ASSEMBLIES shall be stored in FAILED FUEL CANS located in Zones A and B of Figure 2-4.

The poison rodlets consist of B_4C (pellets or powder) encased in ~~a 0.75" nominal OD~~ stainless steel tube ~~with a wall thickness of 0.035~~". The minimum linear B_4C content is 0.70 g/cm with sufficient length to cover the active fuel length.

Para break

Fuel Assembly poison rods installed within the guide tubes for criticality control in the spent fuel pool racks may be stored with any INTACT FUEL ASSEMBLY or DAMAGED FUEL ASSEMBLIES as long as the total assembly weight is less than that specified in Table 2-5.

Each poison rodlet may include a lifting mechanism to allow insertion into the selected SFA guide tube.

A summary of the storage configurations analyzed is presented in Table 2-8.

2.3 Functional and Operating Limits Violations

If any Functional and Operating Limit of 2.1 is violated, the following actions shall be completed:

a. The affected FUEL ASSEMBLIES shall be placed in a safe condition.

b. Within 24 hours, notify the NRC Operations Center.

c. Within 30 days, submit a special report which describes the cause of the violation and the actions taken to restore compliance and prevent recurrence.

Table 2-1 Fuel Specifications (24PT1-DSC)

Fuel Type	Maximum Initial Enrichment	Cladding Material	Minimum Cooling Time	Minimum Initial Enrichment	Maximum Burnup
UO_2 WE 14x14 (with or without IFBA fuel rods)	4.05 weight % U-235	Type 304 Stainless Steel	10 years	See Table 2-4 for Enrichment, Burnup, and Cooling Time Limits.	
WE 14x14 MOX	2.84 weight % Fissile Pu - 64 rods 3.10 weight % Fissile Pu - 92 rods 3.31 weight % Fissile Pu - 24 rods	Zircalloy-4	20 years	2.78 weight % Fissile Pu –64 rods 3.05 weight % Fissile Pu –92 rods 3.25 weight % Fissile Pu – 24 rods	25,000 MWd/MTU
Integral Control Components	N/A	N/A	10 years	N/A	N/A

Table 2-2 Fuel Dimension and Weights (24PT1-DSC)

Parameter	WE 14x14 SC[1]	WE 14x14 MOX[1]
Number of Rods	180	180
Number of Guide Tubes/Instrument Tubes	16	16
Cross Section (in)	7.763	7.763
Unirradiated Length (in)	138.5	138.5
Fuel Rod Pitch (in)	0.556	0.556
Fuel Rod O.D. (in)	0.422	0.422
Clad Material	Type 304 SS	Zircaloy-4
Clad Thickness (in)	0.0165	0.0243
Pellet O.D. (in)	0.3835	0.3659
Max. Initial ^{235}U Enrichment (%wt)	4.05	Note 2
Theoretical Density (%)	93-95	91
Active Fuel Length (in)	120	119.4
Max. U Content (kg)	375	Note 3
Assembly Weight (lbs)	1210	1150
Max. Assembly Weight incl. NFAH[4] (lbs)	1320	1320

(1) Nominal values shown unless stated otherwise
(2) Mixed-Oxide assemblies with 0.71 weight % U-235 and maximum fissile Pu weight of 2.84 weight % (64 rods), 3.10 weight % (92 rods), and 3.31 weight % (24 rods)
(3) Total weight of Pu is 11.24 kg and the total weight of U is 311.225 kg
(4) Weights of TPAs and NSAs are enveloped by RCCAs

Table 2-3 Maximum Neutron and Gamma Source Terms (24PT1-DSC)

Parameter	WE 14x14 SC	WE 14x14 MOX
Gamma Source (γ/sec/assy)	3.43E+15	9.57E+14
Neutron Source (n/sec/assy)	2.84E+08	4.90E+07

Parameter	RCCAs	TPAs	NSAs
Gamma Source (γ/sec/assy)	7.60E+12	5.04E+12	1.20E+13
Decay heat (Watts)	1.90	1.2	1.66

Table 2-4 Fuel Qualification Table (24PT1-DSC)

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

Burnup GWd/MTU	Initial Enrichment (weight % U-235)			
	3.12	3.36	3.76	3.96
45.0	Not Analyzed		15.2	15.2*
43.3			15.2	11.5
40.0		10.9	10.9**	10.9**
36.8		10.9	10.0***	10.0***
35.0 or less	10.0***	10.0***	10.0***	10.0***

Notes
- * Cooling time based on 3.76 weight % enrichment is conservatively used.
- ** Cooling time based on 3.36 weight % enrichment is conservatively used.
- *** Cooling time based on shielding analysis source term.

General Notes:
- Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Example: An assembly with an initial enrichment of 3.90 w/o U-235 and a burnup of 37 GWd/MTU is acceptable for storage after a 10.9 year cooling time as defined at the intersection of 3.76 weight % U-235 (rounding down) and 40 GWd/MTU (rounding up) on the qualification table.

Table 2-5 PWR Fuel Specification of Intact Fuel to be stored in NUHOMS® 24PT4-DSC

Fuel Design:	INTACT CE 16x16 PWR FUEL ASSEMBLY or equivalent reload fuel that is enveloped by the FUEL ASSEMBLY design characteristics as listed in Table 2-7 and the following requirements:
Fuel Damage:	Fuel with known or suspected cladding damage in excess of pinhole leaks or hairline cracks or an assembly with partial and/or missing rods is not authorized to be stored as "INTACT PWR FUEL."
Physical Parameters(1)	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
~~Assembly Weight (lbs)~~	1500(2)
~~No. of Assemblies per DSC~~	≤ 24 intact assemblies
Max. U Content (kg)	455.5
Fuel Cladding	Zircaloy-4 or ZIRLO™
RECONSTITUTED FUEL ASSEMBLIES	DAMAGED FUEL Rods replaced by either stainless rods ~~(up to 8 rods~~ per assembly) or Zircaloy clad ~~low-enrichment~~ uranium rods (any number of rods per assembly)
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt %)	Per Table 2-8 and Figure 2-4
Fuel Burnup and Cooling Time	Per Table 2-9, Table 2-10, Table 2-11 and Table 2-12 For RECONSTITUTED FUEL with stainless steel replacement rods per Table 2-13, Table 2-14, Table 2-15 and Table 2-16
Decay Heat	Per Figure 2-1, Figure 2-2 or Figure 2-3.
Gamma Source (γ/sec/assembly)	7.5E+15
Neutron Source (n/sec/assembly)	3.696E+08

Notes:

(1) Nominal values shown unless stated otherwise.
(2) Does not include weight of Poison Rodlets (25 lbs each).

Table 2-6 PWR Fuel Specifications of DAMAGED FUEL to be Stored in NUHOMS® 24PT4-DSC

Fuel Design:	DAMAGED CE 16x16 PWR FUEL ASSEMBLY or equivalent reload fuel that is enveloped by the FUEL ASSEMBLY design characteristics as listed in Table 2-7 and the following requirements:
Fuel Damage:	DAMAGED FUEL may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). DAMAGED FUEL ASSEMBLIES shall be encapsulated in individual FAILED FUEL CANS and placed in Zones A and/or B as shown in Figure 2-4. FUEL DEBRIS and DAMAGED FUEL Rods that have been removed from a DAMAGED FUEL ASSEMBLY and placed in a ROD STORAGE BASKET are also considered as DAMAGED FUEL. Loose FUEL DEBRIS not contained in a ROD STORAGE BASKET, may also be placed in a FAILED FUEL CAN for storage, provided the size of the debris is larger than the FAILED FUEL CAN screen mesh opening. FUEL DEBRIS may be associated with any type of UO_2 fuel provided that the maximum uranium content and enrichment limits are met.
Physical Parameters[1]	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500[2]
No. of Assemblies per DSC	≤ 12 DAMAGED ASSEMBLIES, balance INTACT
Max. U Content (kg)	455.5
Fuel Cladding	Zircaloy-4 or ZIRLO™
RECONSTITUTED FUEL ASSEMBLIES	DAMAGED FUEL Rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly)
Nuclear and Radiological Parameters	
Initial ^{235}U Enrichment (wt %)	Per Table 2-8 and Figure 2-4.
Fuel Burnup and Cooling Time	Per Table 2-9, Table 2-10, Table 2-11 and Table 2-12 For RECONSTITUTED FUEL with stainless steel replacement rods per Table 2-13, Table 2-14, Table 2-15 and Table 2-16
Decay Heat	Per Figure 2-1, or Figure 2-2 or Figure 2-3
Gamma Source (γ/sec/assembly)	7.5E+15
Neutron Source (n/sec/assembly)	3.696E+08

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of Poison Rodlets (25 lbs each).

Table 2-7 PWR Fuel Assembly Design Characteristics (24PT4-DSC)

Assembly Class	CE 16x16[1]
Assembly Length	Table 2-5 or Table 2-6
Max. Initial ^{235}U Enrichment (wt %)	4.85
Fissile Material	UO_2, or $(U, Er)O_2$, or $(U, Gd)O_2$
Number of Rods	236
Fuel Rod Pitch (in)	0.506
Fuel Rod O.D. (in)	0.382
Clad Thickness (in)	0.025
Nominal Pellet O.D., (in)	0.3255[2]
Number of ~~Guides~~ Guide Tubes	5

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Bounds pellets with a nominal OD of 0.325".

Table 2-8 Maximum Fuel Enrichment v/s Neutron Poison Requirements for 24PT4-DSC

Storage Configuration	Maximum No. of DAMAGED FUEL ASSEMBLIES(1)	Maximum ^{235}U Fuel Enrichment (wt %)	DSC Basket, Minimum BORAL© Areal Density (gm/cm^2)	Minimum No. of Poison Rodlets Required(2)
All INTACT FUEL ASSEMBLIES	0	4.1	.025 (Type A Basket)	0
	0	4.85	.068 (Type B Basket)	0
Combination of DAMAGED and INTACT FUEL ASSEMBLIES	4	4.1	.025 (Type A Basket)	0
	4	4.85	.068 (Type B Basket)	0
	12	3.7 (DAMAGED) 4.1 (INTACT)	.025 (Type A Basket)	0
	12	4.1 (DAMAGED) 4.85 (INTACT)	.068 (Type B Basket)	0
	12	4.1	.025 (Type A Basket)	1(2) (Located in center guide tube of each INTACT ASSEMBLY)
	12	4.85	.068 (Type B Basket)	5(2) (Located in all five guide tubes of each INTACT ASSEMBLY)

Notes:

(1) See Figure 2-4 for location of DAMAGED FUEL ASSEMBLIES within the 24PT4-DSC basket (Zones A and/or B only).

(2) Poison rodlets are only required for a specific DSC configuration with a payload of 5-12 DAMAGED ASSEMBLIES in combination with maximum fuel enrichment levels as shown. The poison rodlets are to be located within the guide tubes of the inner Zone C INTACT ASSEMBLIES as shown in Figure 2-4.

Table 2-9 PWR Fuel Qualification Table for 1.26 kW per Assembly for the NUHOMS® 24PT4-DSC
(Minimum required years of cooling time after reactor core discharge)

BU (GWd MTU)	Initial Enrichment																														
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
38											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
39											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
40											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
41			Not Analyzed								5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
42											6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
43																5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
44																6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5
45																					5	5	5	5	5	5	5	5	5	5	5
48																					6	6	6	6	6	6	6	6	6	6	6
51																					7	7	7	7	6	6	6	6	6	6	6
54																					7	7	7	7	7	7	7	7	7	7	7
57																					8	8	8	8	8	8	8	8	8	8	8
60																					9	9	9	9	9	9	9	9	9	9	9

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a five-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-10 PWR Fuel Qualification Table for 1.2 kW per Assembly for the NUHOMS® 24PT4-DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd/MTU)	Initial Enrichment																														
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
38											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
39											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
40											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
41			Not Analyzed								5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
42											6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
43																6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5
44																6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5
45																					6	6	6	6	6	6	6	6	6	6	6
48																					6	6	6	6	6	6	6	6	6	6	6
51																					7	7	7	7	7	7	7	7	7	7	7
54																					8	8	8	8	8	8	8	8	8	7	7
57																					9	9	9	9	9	9	9	9	8	8	8
60																					10	10	10	10	10	10	10	10	10	10	9

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a five-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-11 PWR Fuel Qualification Table for 1.0 kW per Assembly for the NUHOMS® 24PT4-DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
38											6	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5
39											6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5
40											6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
41				Not Analyzed							6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
42											7	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
43																7	7	7	7	7	7	7	6	6	6	6	6	6	6	6	6
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	6
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					9	9	9	9	9	9	9	9	9	9	9
54																					11	11	11	11	11	10	10	10	10	10	10
57																					13	13	13	13	12	12	12	12	12	12	12
60																					15	15	15	15	15	15	15	14	14	14	14

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a six-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-12 PWR Fuel Qualification Table for 0.9 kW per Assembly for the NUHOMS® 24PT4-DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	6	6	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5
38											6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
39											7	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	6	6	6	6	6	6	6
41			Not Analyzed								7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7
44																8	8	8	8	8	8	8	8	8	8	8	8	8	8	7	7
45																					8	8	8	8	8	8	8	8	8	8	8
48																					10	10	9	9	9	9	9	9	9	9	9
51																					11	11	11	11	11	11	11	11	11	11	11
54																					14	14	14	13	13	13	13	13	13	13	13
57																					17	16	16	16	16	16	16	16	15	15	15
60																					20	19	19	19	19	19	19	18	18	18	18

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-13 PWR Fuel Qualification Table for 1.26 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

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

BU (GWd/MTU)	Initial Enrichment																														
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41			Not Analyzed								7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					8	8	8	8	8	8	8	8	8	8	8
54																					9	9	9	9	9	9	9	9	9	9	9
57																					10	10	10	10	10	10	10	10	10	10	10
60																					12	12	12	12	12	12	12	12	12	12	12

Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-14 PWR Fuel Qualification Table for 1.2 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

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

BU (GWd/MTU)	Initial Enrichment																														
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41			Not Analyzed								7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					8	8	8	8	8	8	8	8	8	8	8
54																					9	9	9	9	9	9	9	9	9	9	9
57																					10	10	10	10	10	10	10	10	10	10	10
60																					12	12	12	12	12	12	12	12	12	12	12

Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-15 PWR Fuel Qualification Table for 1.0 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

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

BU (GWd/ MTU)	Initial Enrichment																														
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41			Not Analyzed								7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					9	9	9	9	9	9	9	9	9	9	9
54																					11	11	11	11	11	10	10	10	10	10	10
57																					13	13	13	13	12	12	12	12	12	12	12
60																					15	15	15	15	15	15	15	14	14	14	14

Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

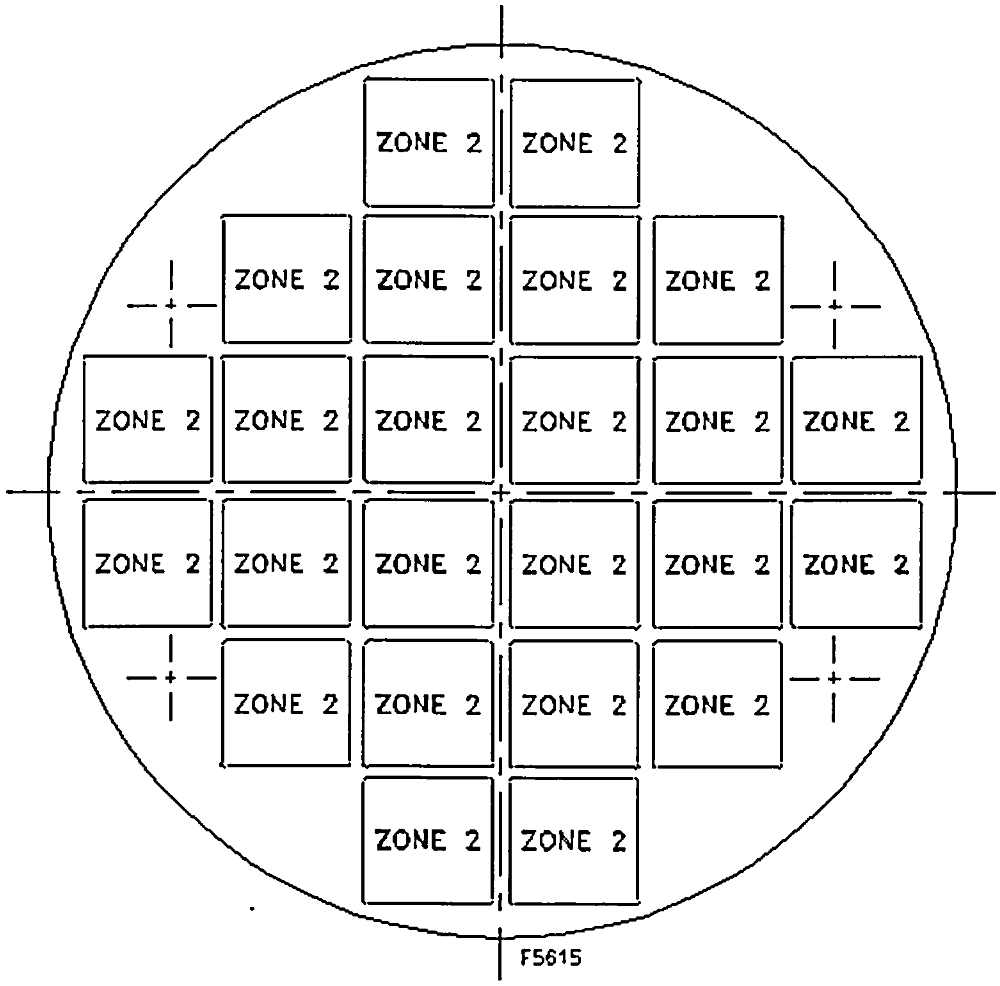
Table 2-16 PWR Fuel Qualification Table for 0.9 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

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

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41			Not Analyzed								7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7
44																8	8	8	8	8	8	8	8	8	8	8	8	8	8	7	7
45																					8	8	8	8	8	8	8	8	8	8	8
48																					10	10	9	9	9	9	9	9	9	9	9
51																					11	11	11	11	11	11	11	11	11	11	11
54																					14	14	14	13	13	13	13	13	13	13	13
57																					17	16	16	16	16	16	16	16	15	15	15
60																					20	19	19	19	19	19	19	18	18	18	18

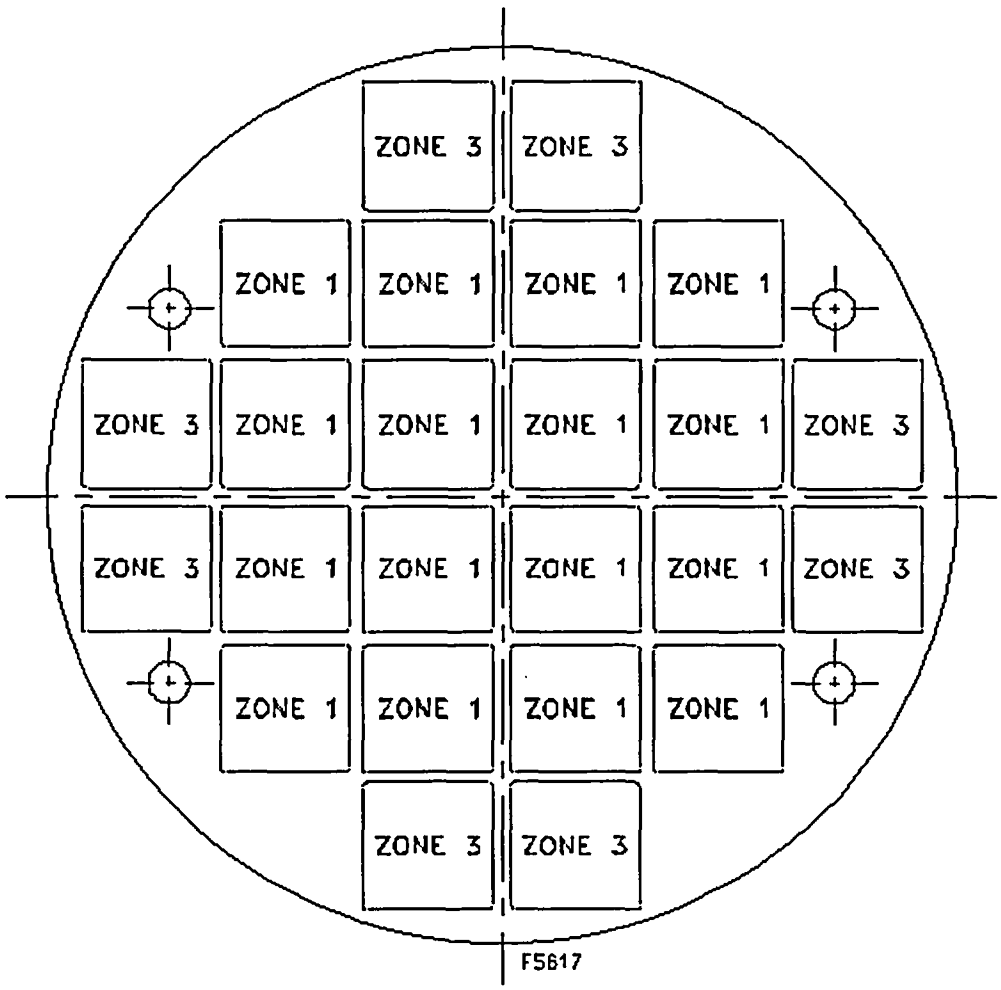
Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU and is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 47 GWd/MTU is acceptable for storage after a nine-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 48 GWd/MTU (rounding up) on the qualification table.



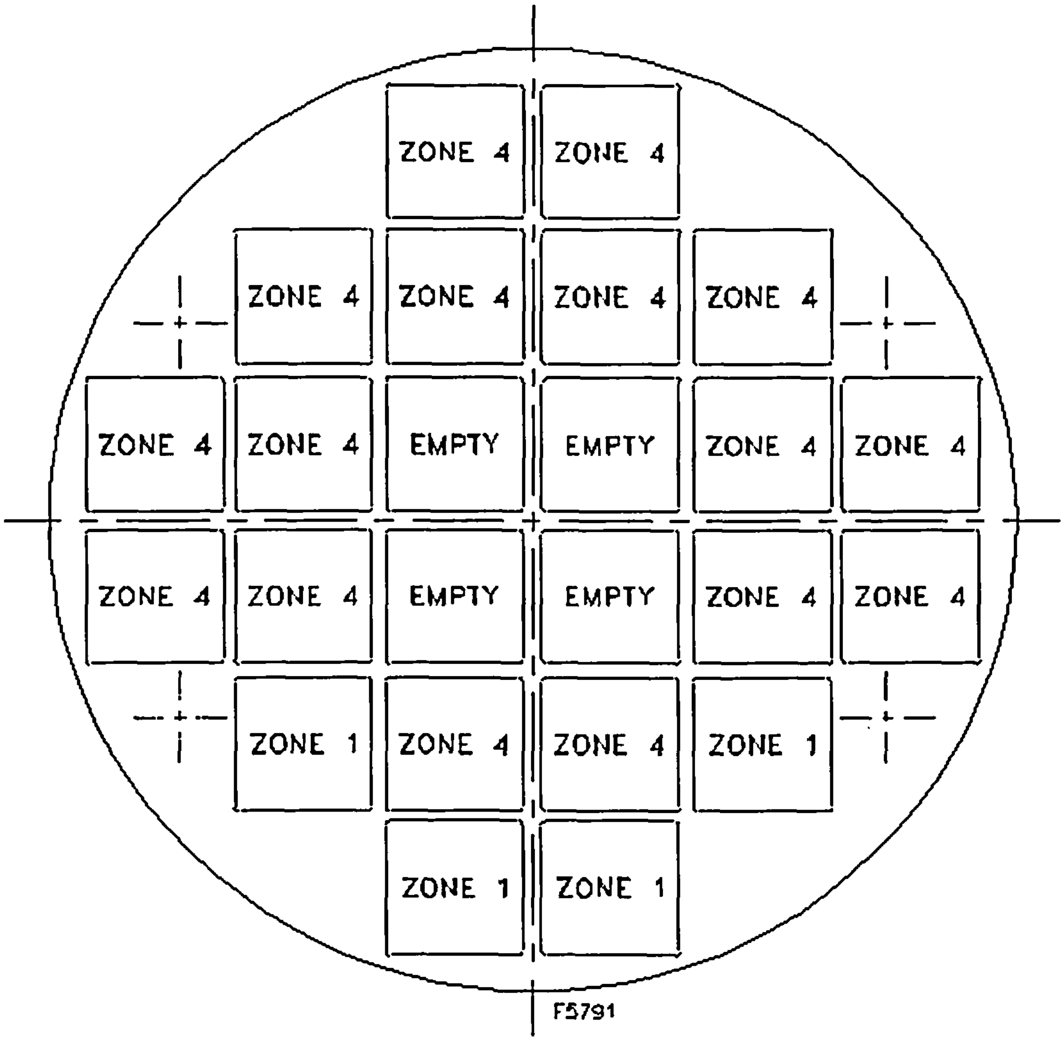
	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts / FA)	NA	1.0	NA	NA
Maximum Decay Heat per Zone (kWatts)	NA	24.0	NA	NA

Figure 2-1 24PT4-DSC Heat Load Configuration #1, kW/Assembly



	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts / FA)	0.9	NA	1.2	NA
Maximum Decay Heat per Zone (kWatts)	14.4	NA	9.6	NA

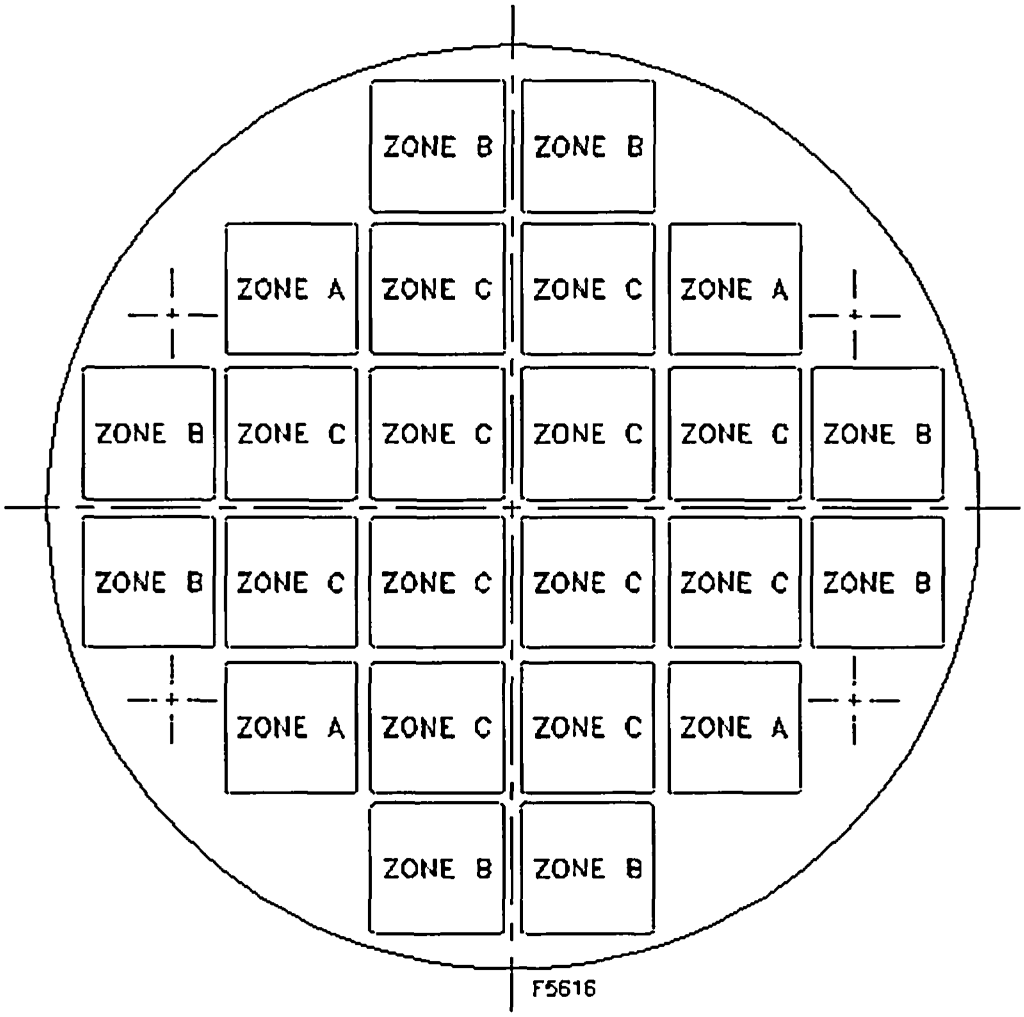
Figure 2-2 24PT4-DSC Heat Load Configuration #2, kW/Assembly



	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts / FA)	0.9	NA	NA	1.26
Maximum Decay Heat per Zone (kWatts)	3.6	NA	NA	20.16

Note: FUEL ASSEMBLIES with a heat load of 0.9 kW (Zone 1) may also be placed anywhere in Zone 4.

Figure 2-3 24PT4-DSC Heat Load Configuration #3, kW/Assembly



Notes:

1. Locations identified as Zone A are for placement of up to 4 DAMAGED FUEL ASSEMBLIES.
2. Locations identified as Zone B are for placement of up to 8 additional DAMAGED FUEL ASSEMBLIES (Maximum of 12 DAMAGED FUEL ASSEMBLIES allowed, Zones A and B combined).
3. Locations identified as Zone C are for placement of up to 12 intact FUEL ASSEMBLIES, including 4 empty slots in the center as shown in Figure 2-3.
4. Poison Rodlets are to be located in the guide tubes of intact FUEL ASSEMBLIES placed in Zone C only per Table 2-4.

Figure 2-4 Location of FAILED FUEL CANS inside 24PT4-DSC

3.0 Limiting Condition for Operation (LCO) and Surveillance Requirement (SR) Applicability

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
delete space	
LCO 3.0.3	Not applicable to a spent fuel storage cask.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS, or that are related to the unloading of a 24PT1-DSC or 24PT4-DSC.
Para break	Exceptions to this Specification are stated in the individual Specifications.
delete space	These exceptions allow entry into specified conditions in the Applicability when the associated ACTIONS to be entered allow operation in the specified condition in the Applicability only for a limited period of time.
LCO 3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate that the LCO is met.
LCO 3.0.6	Not applicable to a spent fuel storage cask.
LCO 3.0.7	Not applicable to a spent fuel storage cask.

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of a DSC.

Add space

delete space

3.1 DSC Integrity

3.1.1.a <u>24PT1-DSC Vacuum Drying Time (Duration) and Pressure</u>

LCO 3.1.1.a Duration: Vacuum Drying of the 24PT1-DSC shall be achieved with the following time durations after the start of bulk water removal (blowdown):

Heat Load (kW)	Time Limit
kW ≤ 12	No limit
12 < kW ≤ 13	71 Hours
13 < kW ≤ 14	54 Hours

Pressure: The 24PT1-DSC vacuum drying pressure shall be sustained at or below 3 Torr (3 mm Hg) absolute for a period of at least 30 minutes following stepped evacuation.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

-------------------------------- NOTE --------------------------------

This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 24PT1-DSC vacuum drying pressure limit not met within 47 hours for a DSC with heat load greater than 12 kW and ≤ 13 kW or within 30 hours for a DSC with heat load greater than 13 kW and ≤ 14 kW.	A.1 Establish helium pressure of at least 1 atm and no greater than 20 psig in the 24PT1-DSC.	24 hours
	<u>OR</u>	
	A.2 Flood the 24PT1-DSC with water submerging all FUEL ASSEMBLIES.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.a.1	Verify that the 24PT1-DSC vacuum pressure is less than, or equal to, 3 Torr (3 mm Hg) absolute for at least 30 minutes, within the specified total time duration based on heat load.	Once per 24PT1-DSC, after an acceptable NDE of the inner top cover plate weld.

3.1.1.b 24PT4-DSC Vacuum Drying Time (Duration) and Pressure

LCO 3.1.1.b Duration: Vacuum Drying of the 24PT4-DSC shall be achieved within the following durations (depending upon the 24PT4-DSC specific heat load configuration) following completion of blowdown using air. No time limits apply for vacuum drying of 24PT4-DSC if helium is used for blowdown. Transfer between air and helium blowdown within the time limits specified below is acceptable. Blowdown with helium with a volume equal to the DSC free volume is required within the air time limit.

Heat Load Configuration	Time Limit Using Air	Time Limit Using Helium
1	35 Hours	No Limit
2	35 Hours	No Limit
3	26 Hours	No Limit

Pressure: The 24PT4-DSC vacuum drying pressure shall be sustained at or below 3 Torr (3 mm Hg) absolute for a period of at least 30 minutes following stepped evacuation.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

------------------------------- NOTE --------------------------------
This specification is applicable to all 24PT4-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 24PT4-DSC vacuum drying pressure limit not met when using air for blowdown within 33 hours (Configurations #1 or 2) or 24 hours (Configuration #3).	A.1 Establish helium pressure of at least 1 atm and no greater than 20 psig in the 24PT4-DSC. Vacuum drying can proceed with no time limit.	2 hours
	<u>OR</u>	
	A.2 Flood the 24PT4-DSC with water submerging all FUEL ASSEMBLIES.	2 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.b.1	Verify that the 24PT4-DSC vacuum pressure is less than, or equal to, 3 Torr (3 mm Hg) absolute for at least 30 minutes, within the specified total time duration based on heat load.	Once per 24PT4-DSC, after an acceptable NDE of the inner top cover plate weld.

3.1.2.a 24PT1-DSC Helium Backfill Pressure

LCO 3.1.2.a 24PT1-DSC helium backfill pressure shall be 1.5 ± 1.5 psig (stable for 30 minutes after filling).

delete space

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

------------------------------------ NOTE ------------------------------------

This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 3.1.2.a.1 is performed. A. The required backfill pressure cannot be obtained or stabilized.	A.1 Establish the 24PT1-DSC helium backfill pressure to within the limit. OR A.2 Flood the 24PT1-DSC with water submerging all fuel assemblies.	24 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.a.1 Verify that the 24PT1-DSC helium backfill pressure is 1.5 ± 1.5 psig.	Once per 24PT1-DSC, after the completion of TS 3.1.1.a actions.

3.1.2.b 24PT4-DSC Helium Backfill Pressure

LCO 3.1.2.b 24PT4-DSC helium backfill pressure shall be 6.0 + 1.0 / -0.0 psig (stable for 30 minutes after filling).

APPLICABILITY: During LOADING OPERATIONS. delete space

ACTIONS

------------------------------------ NOTE ------------------------------------

This specification is applicable to all 24PT4-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 3.1.2.b.1 is performed.		
A. The required backfill pressure cannot be obtained or stabilized.	A.1 Establish the 24PT4-DSC helium backfill pressure to within the limit.	24 hours
	OR	
	A.2 Re-initiate vacuum drying.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.b.1	Verify that the 24PT4-DSC helium backfill pressure is 6.0 + 1.0 / -0.0 psig.	Once per 24PT4-DSC, after the completion of TS 3.1.1.b actions.

4.0 Design Features

The specifications in this section include the design characteristics of special importance to each of the physical barriers and to maintenance of safety margins in the Advanced NUHOMS® System design. The principal objective of this section is to describe the design envelope that may constrain any physical changes to essential equipment. Included in this section are the site environmental parameters that provide the bases for design, but are not inherently suited for description as LCOs.

4.1 Site

4.1.1 Site Location

Because this FSAR is prepared for a general license, a discussion of a site-specific ISFSI location is not applicable.

4.2 Storage System Features

4.2.1 Storage Capacity

The total storage capacity of the ISFSI is governed by the plant-specific license conditions.

4.2.2 Storage Pad

For sites for which soil-structure interaction is considered important, the licensee is to perform site-specific analysis considering the effects of soil-structure interaction. Amplified seismic spectra at the location of the AHSM center of gravity (CG) is to be developed based on the SSI responses. The AHSM center of gravity is shown in Table 3.2-1. The site-specific spectra at the AHSM CG must be bounded by the spectra presented in Chapter 2.

The storage pad location shall have no potential for liquefaction at the site-specific SSE level earthquake.

Additional requirements for the pad configuration are provided in Section 4.4.2.

4.2.3 Canister Neutron Absorber

For a 24PT1-DSC basket, neutron absorber with a minimum ^{10}B loading of 0.025 grams/square centimeter is provided for criticality control.

For a 24PT4-DSC basket, two alternate neutron absorber specifications are provided for criticality control depending upon the number of DAMAGED ASSEMBLIES and/or the maximum fuel enrichment of the payload as shown in Table 2-8:

- Type A Basket (minimum areal ^{10}B loading of 0.025 gm/cm^2)
- Type B Basket (minimum areal ^{10}B loading of 0.068 gm/cm^2)

4.2.4 Canister Flux Trap Configuration

The canister flux trap configuration is defined by the spacer disc ligament width dimensions. Figure 4-1 (applicable to 24PT1-DSC and 24PT4-DSC) shows the location and dimensions of the ligaments (the dimensions shown in the one quadrant are applicable to all four quadrants).

4.2.5 Fuel Spacers

Bottom fuel spacers are required to be located at the bottom of the DSC below each FUEL ASSEMBLY stored in the 24PT1-DSC. Top fuel spacers are required to be located above each INTACT FUEL ASSEMBLY stored in the 24PT1-DSC (the FAILED FUEL CAN design includes an integral top fuel spacer and therefore does not require a top fuel spacer).

No fuel spacers are required for 24PT4-DSC.

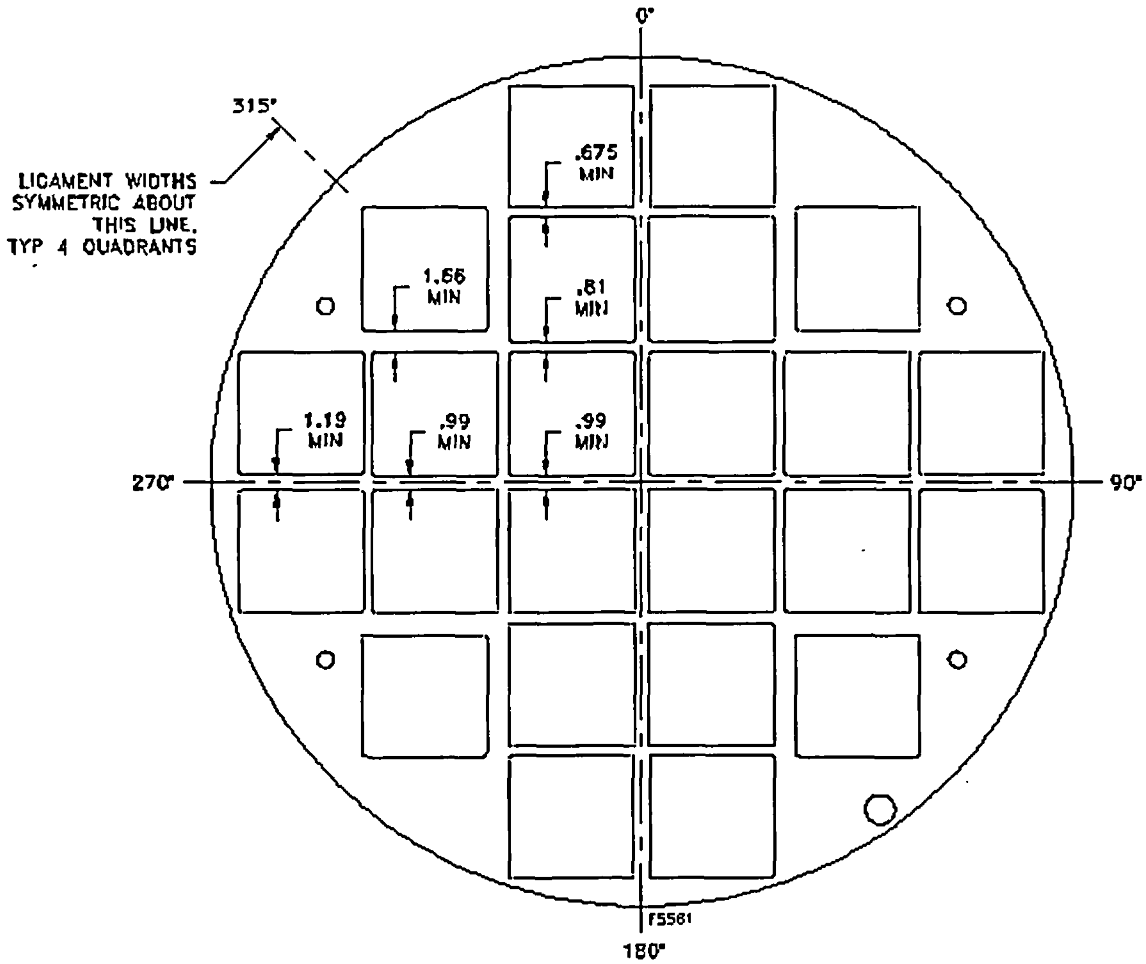


Figure 4-1 Minimum Spacer Disc Ligament Widths

4.3 Codes and Standards

4.3.1 Advanced Horizontal Storage Module (AHSM)

The reinforced concrete AHSM is designed to meet the requirements of ACI 349-97. Load combinations specified in ANSI 57.9-1984, Section 6.17.3.1 are used for combining normal operating, off-normal, and accident loads for the AHSM.

4.3.2 Dry Shielded Canister, 24PT1-DSC or 24PT4-DSC (DSC)

The DSC is designed fabricated and inspected to the maximum practical extent in accordance with ASME Boiler and Pressure Vessel Code Section III, Division 1, 1992 Edition with Addenda through 1994, including exceptions allowed by Code Case –595-1, Subsections NB, NF, and NG for Class 1 components and supports. In addition, Code Case –499-1 applies to 24PT4-DSC spacer discs. Code Alternatives are discussed in 4.3.4.

4.3.3 Transfer Cask

The TRANSFER CASK (OS197 or OS197H) shall meet the codes and standards that are applicable to its design under Certificate of Compliance C of C 1004.

A solar shield is required for cask TRANSFER OPERATIONS at temperatures exceeding 100°F.

4.3.4 Alternatives to Codes and Standards

ASME Code ~~exceptions~~ alternatives for the 24PT1-DSC or 24PT4-DSC (DSC) are listed below:

DSC Shell Assembly Alternatives to ASME Code, Subsection NB

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NB-1100	Requirements for Code Stamping of Components	The DSC shell is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical. However, Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-2130 NB-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the FSAR drawings are obtained from ASME approved MM or MS supplier(s) with ASME CMTR's. Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability & certification are maintained in accordance with TN's NRC approved QA program.

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
NB-6111	All completed pressure retaining systems shall be pressure tested	The shield plug support ring and vent and siphon block are not pressure tested due to the manufacturing sequence. The support ring is not a pressure-retaining item and the siphon block weld is helium leak tested after fuel is loaded and the inner top closure plate installed in accordance with Code Case N-595-1.
NB-7000	Overpressure Protection	No overpressure protection is provided for the DSC. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum internal pressure considering 100% fuel rod failure at maximum accident temperature. The DSC is pressure tested to 120% of normal operating design pressure. An overpressure protection report is not prepared for the DSC.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. In lieu of code stamping, QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TN's approved QA program.

Basket Alternatives to ASME Code, Subsection NG/NF

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NG/NF-1100	Requirements for Code Stamping of Components	The DSC baskets are designed & fabricated in accordance with the ASME Code, Section III, Subsection NG/NF to the maximum extent practical as described in the FSAR, but Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME N or NPT stamp or be ASME Certified.
NG/NF-2130 NG/NF-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the FSAR drawings are obtained from ASME approved MM or MS supplier with ASME CMTR's. Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NG/NF-2130 is not possible. Material traceability & certification are maintained in accordance with TN's NRC approved QA program.
Table NG-3352-1	Permissible Joint Efficiency Factors	Joint efficiency (quality) factor of 1 is assumed for the guidesleeve longitudinal weld. Table NG-3352-1 permits a quality factor of 0.5 for full penetration weld with visual inspection. Inspection of both faces provides n= (2*0.5)=1. This is justified by this gauge of material (0.12 inch) with visual examination of both surfaces which ensures that any significant deficiencies would be observed and corrected.

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
NG/NF-8000	Requirements for nameplates, stamping & reports per NCA-8000	The DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. In lieu of code stamping, QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TN's approved QA program.
N/A	N/A	Oversleeve to guidesleeve welds are non-code welds which meet the requirements of AWS D1.3-98, the Structural Welding Code-Sheet Steel.
NG-3000 / Section II, Part D, Table 2A	Maximum temperature limit for Type 304 plate material is 800°F	For 24PT4-DSC only: The DSC guidesleeves, oversleeves and failed fuel cans do not comply with ASME Code limit of 800°F for Type 304 steel for the postulated blocked vent accident for approximately 25 hours. The maximum predicted temperature of those components for this event is less than 900°F. In accordance with Table I-14.5 of Article NH, the expected reduction in material strength is small (less than 1 ksi) and the calculated stress ratio is very small.

Proposed alternatives to the ASME code, other than the aforementioned ASME Code alternatives may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards, or designee. The applicant should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or
2. Compliance with the specified requirements of ASME Code, Section III, 1992 Edition with Addenda through 1994 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for alternatives in accordance with this section should be submitted in accordance with 10CFR 72.4.

4.4 Storage Location Design Features

The following storage location design features and parameters shall be verified by the system user to assure technical agreement with this FSAR.

4.4.1 Storage Configuration

AHSMs are to be tied together in single rows or back to back arrays with not less than 3 modules tied together (side by side). Any 2 of the 3 modules may be empty (not contain a loaded DSC). Each group of modules not tied together must be separated from other groups by a minimum of 20 feet to accommodate possible sliding during a seismic event. The distance between any module and the edge of the ISFSI pad shall be no less than 10 feet.

4.4.2 Concrete Storage Pad Properties to Limit DSC Gravitational Loadings Due to Postulated Drops

The TC/DSC has been evaluated for drops of up to 80 inches onto a reinforced concrete storage pad. The evaluations are based on the concrete parameters specified in EPRI Report NP-4830, "The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent Fuel Casks," October 1986.

4.4.3 Site Specific Parameters and Analyses

The following parameters and analyses shall be verified by the system user for applicability at their specific site.

1. Tornado maximum wind speeds: 290 mph rotational
   70 mph translational

2. Flood levels up to 50 ft. and water velocity of 15 fps.

3. One-hundred year roof snow load of 110 psf.

4. Normal ambient temperatures of 0°F to 104°F.

5. Off-normal ambient temperature range of –40°F without solar insolation to 117°F with full solar insolation.

6. The potential for fires and explosions shall be addressed, based on site-specific considerations.

7. Supplemental Shielding: In cases where engineered features (i.e., berms, shield walls) are used to ensure that the requirements of 10CFR 72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

8. Seismic restraints shall be provided to prevent overturning of a loaded TC in a vertical orientation in the plant's decontamination area during a seismic event if a certificate holder determines that the horizontal acceleration is 0.40 g or greater. The determination of horizontal acceleration acting at the center of gravity (CG) of the loaded TC must be based on a peak horizontal ground acceleration at the site.

9. The effects of lightning, tsunamis, hurricanes and seiches, based on site-specific conditions shall be shown to be bounded by the design capability of the storage cask system.

5.0 Administrative Controls

5.1 Procedures

Each user of the Advanced NUHOMS® System will prepare, review, and approve written procedures for all normal operations, maintenance, and testing at the ISFSI prior to its operation. Written procedures shall be established, implemented, and maintained covering the following activities that are important to safety:

- Organization and management
- Routine ISFSI operations
- Alarms and annunciators
- Emergency operations
- Design control and facility change/modification
- Control of surveillances and tests
- Control of special processes
- Maintenance
- Health physics, including ALARA practices
- Special nuclear material accountability
- Quality assurance, inspection, and audits
- Physical security and safeguards
- Records management
- Reporting
- All programs specified in Section 5.2

5.2 Programs

Each user of the Advanced NUHOMS® System will implement the following programs to ensure the safe operation and maintenance of the ISFSI:

- Safety Review Program
- Training Program
- Radiological Environmental Monitoring Program
- Radiation Protection Program
- AHSM Thermal Monitoring Program

5.2.1 Safety Review Program

Users shall conduct safety reviews in accordance with 10CFR 72.48 to determine whether proposed changes, tests, and experiments require NRC approval before implementation. Changes to the Technical Specification Bases and other licensing basis documents will be conducted in accordance with approved administrative procedures. Changes may be made to Technical Specification Bases and other licensing basis documents without prior NRC approval, provided the changes meet the criteria of 10CFR 72.48.

The safety review process will contain provisions to ensure that the Technical Specification Bases and other licensing basis documents are maintained consistent with the FSAR.

Proposed changes that do not meet the criteria above will be reviewed and approved by the NRC before implementation. Changes to the Technical Specification Bases implemented without prior NRC approval will be provided to the NRC in accordance with 10CFR 72.48.

5.2.2 Training Program

Training modules shall be developed as required by 10CFR 72. Training modules shall require a comprehensive program for the operation and maintenance of the Advanced NUHOMS® System and the INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI). The training modules shall include the following elements, at a minimum:

- Advanced NUHOMS® System design (overview)
- ISFSI Facility design (overview)
- Systems, Structures, and Components Important to Safety (overview)
- Advanced NUHOMS® System Safety Analysis Report (overview)
- NRC Safety Evaluation Report (overview)
- Certificate of Compliance conditions
- Advanced NUHOMS® System Technical Specifications
- Applicable Regulatory Requirements (e.g., 10CFR 72, Subpart K, 10CFR 20, 10 CFR Part 73)
- Required Instrumentation and Use
- Operating Experience Reviews
- Advanced NUHOMS® System and Maintenance procedures, including:
    Fuel qualification and loading, as applicable
    Rigging and handling,
    LOADING OPERATIONS as described in Chapters 8, A.8, and Sections 9.2 and A.9.2 of the FSAR,
    UNLOADING OPERATIONS including reflooding,
    Auxiliary equipment operations and maintenance (i.e., welding operations, vacuum drying, helium backfilling and leak testing, reflooding),
    TRANSFER OPERATIONS including loading and unloading of the Transfer Vehicle,
    ISFSI Surveillance operations,
    Radiation Protection,
    Maintenance,
    Security,
    Off-normal and accident conditions, responses and corrective actions.

5.2.3 Radiological Environmental Monitoring Program

a) A radiological environmental monitoring program will be implemented to ensure that the annual dose equivalent to an individual located outside the ISFSI controlled area does not exceed the annual dose limits specified in 10CFR 72.104(a).

b) Operation of the ISFSI will not create any radioactive materials or result in any credible liquid or gaseous effluent release.

c) In accordance with 10CFR 72.212(b)(2), a periodic report will be submitted by the licensee that specifies the quantity of each of the principal radionuclides released to the environment in liquid and gaseous effluents during the previous year of operation.

5.2.4 Radiation Protection Program

The Radiation Protection Program will establish administrative controls to limit personnel exposure to As Low As Reasonably Achievable (ALARA) levels in accordance with 10CFR Part 20 and Part 72.

a. As part of its evaluation pursuant to 10CFR 72.212, the licensee shall perform an analysis to confirm that the limits of 10CFR 20 and 10CFR 72.104 will be satisfied under the actual site conditions and configurations considering the planned number of DSCs to be used and the planned fuel loading conditions.

b. A monitoring program to ensure the annual dose equivalent to any real individual located outside the ISFSI controlled area does not exceed regulatory limits is incorporated as part of the environmental monitoring program in the Radiological Environmental Monitoring Program of Section 5.2.3.

c. Following placement of each loaded TRANSFER CASK into the cask decontamination area and prior to transfer to the ISFSI, the DSC smearable surface contamination levels on the outer surface of the DSC shall be 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.

   The contamination limits specified above are based on the allowed removable external radioactive contamination specified in 49 CFR 173.443 (as referenced in 10 CFR 71.87(i)) the system provides significant additional protection for the DSC surface than the transportation configuration). The AHSM will protect the DSC from direct exposure to the elements and will therefore limit potential releases of removable contamination. The probability of any removable contamination being entrapped in the AHSM air flow path released outside the AHSM is considered extremely small.

d. TC surface dose rates with 24PT4-DSC payload as specified below shall be confirmed prior to 24PT4-DSC closure to assure proper loading and consistency with the offsite dose analysis.

a. ≤ 260 mrem/hr (gamma) at 3 feet from the centerline of the top of the welder

delete space

neutron shield prior to wet welding operations, with the shield plug in place and approximately 4" of water drained and the welder with its neutron shield in place.

b. ≤ 95 mrem/hr (gamma) at 3 feet from the surface of the TC neutron shield at the

centerline (mid-height) of the TC prior to wet welding operations

5.2.5 AHSM Thermal Monitoring Program

This program provides guidance for temperature measurements that are used to monitor the thermal performance of each AHSM. The intent of the program is to prevent conditions that could lead to exceeding the concrete and fuel clad temperature criteria.

a) AHSM Concrete Temperature

The temperature measurement will be a direct measurement of the AHSM concrete temperature, or other means that would identify and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria. A temperature measurement of the thermal performance for each AHSM will be taken on a daily basis for the 24PT1-DSC with a 40 hour blocked vent time limit and twice a day for the 24PT4-DSC with a 25 hour blocked vent time limit.

If the temperature of the AHSM at the monitored location rises by more than 80°F for the 24PT1-DSC and 30°F for the 24PT4-DSC, based on this surveillance, then it is possible that some type of an inlet and or outlet vent blockage has occurred. Visual inspection of the vents will be initiated and appropriate corrective actions will be taken to avoid exceeding the concrete and cladding temperature limits. The 80°F/30°F values are obtained from a review of a transient thermal analysis of the AHSM with a 24 kW heat load to ensure that the rapid heatup is detected in time to initiate corrective action prior to exceeding concrete or DSC basket material temperature limits for the respective AHSM DSC payloads.

In addition, if the temperature of the AHSM at the monitored location is greater than 225°F, then it is possible that some type of an inlet and or outlet vent blockage has occurred. Visual inspection of the vents will be initiated and appropriate corrective actions need to be taken to avoid exceeding the concrete and cladding temperature limits. The 225°F temperature limits are chosen based on the expected concrete temperature for the 24 kW blocked vent scenarios to ensure that the associated fuel clad temperature is not exceeded.

The AHSM Thermal Monitoring Program provides a positive means to identify conditions that could approach the temperature criteria for proper AHSM operation and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria.

b) AHSM Air Temperature Difference

Following initial DSC transfer to the AHSM, the air temperature difference between ambient temperature and the roof vent temperature will be measured 24 hours (plus or minus 8 hours) after DSC insertion into the AHSM and again 5 to 7 days after insertion into the AHSM and prior to removing the AHSM door to perform the DSC retainer adjustment. If the air temperature differential is greater than 100°F, the air inlets and exits should be checked for blockage. If after removing any blockage found, the temperature differential is still greater than 100°F, corrective actions and analysis of existing conditions will be performed in accordance with the site corrective action program to confirm that conditions adversely affecting the concrete or fuel cladding do not exist.

The specified air temperature rise ensures the fuel clad and concrete temperatures are maintained at or below acceptable long-term storage limits. If the temperature rise is within the ≤100°F, then the AHSM and DSC are performing as designed and no further temperature measurements are required.

c) AHSM Air Vents

Since the AHSMs are located outdoors, there is a possibility that the AHSM air inlet and outlet openings could become blocked by debris. Although the ISFSI security fence and AHSM bird screens reduce the probability of AHSM air vent blockage, the ISFSI FSAR postulates and analyzes the effects of air vent blockage.

The AHSM design and accident analyses demonstrate the ability of the ISFSI to function safely if obstructions in the air inlets or outlets impair airflow through the AHSM for extended periods. This specification ensures that blockage will not exist for periods longer than assumed in the analyses.

Site personnel will conduct a daily visual inspection of the air vents to ensure that AHSM air vents are not blocked for more than 40 hours (with 24PT1-DSC). For the 24PT4-DSC credit will be taken for the temperature measurement taken in Section 5.2.5.a. Visual inspection of the AHSM air vents with the 24PT4-DSC will be performed only if the temperature monitoring system data is unavailable or if the temperature limits specified in Section 5.2.5.a are exceeded to ensure that AHSM air vents are not blocked for more than 25 hours.

5.3 <u>Lifting Controls</u>

5.3.1 <u>Cask Lifting Heights</u>

The lifting height of a loaded TC/DSC, is limited as a function of location and temperature as follows:

a) The maximum lift height of the TC/DSC inside the Fuel Handling Building shall be 80 inches if the ambient temperature is below 0°F but higher than -80°F.

b) No lift height restriction other than 10CFR50 administrative controls, is imposed on the TC/DSC during LOADING OPERATIONS provided that a single-failure-proof crane is used and if the ambient temperature is higher than 0°F.

c) The maximum lift height and handling height for all TRANSFER OPERATIONS shall be 80 inches if the ambient temperature is greater than 0°F.

These restrictions ensure that any DSC drop as a function of location or low temperature is within the bounds of the accident analysis. If the ambient temperature is outside of the specification limits, LOADING and TRANSFER OPERATIONS will be terminated.

5.3.2 Cask Drop

Inspection Requirement

The DSC will be inspected for damage after any TRANSFER CASK drop of fifteen inches or greater.

Background

TC/DSC handling and loading activities are controlled under the 10CFR 50 license until a loaded TC/DSC is placed on the transporter, at which time fuel handling activities are controlled under the 10CFR 72 license. Although the probability of dropping a loaded TC/DSC while en route from the Fuel Handling Building to the ISFSI is small, the potential exists to drop the cask 15 inches or more.

Safety Analysis

The analysis of bounding drop scenarios shows that the TRANSFER CASK will maintain the structural integrity of the DSC confinement boundary from an analyzed drop height of 80 inches. The 80-inch drop height envelopes the maximum vertical height of the TRANSFER CASK when secured to the transfer trailer while en route to the ISFSI.

Although analyses performed for cask drop accidents at various orientations indicate much greater resistance to damage, requiring the inspection of the DSC after a drop of 15 inches or greater ensures that:

1. The DSC will continue to provide confinement

2. The TRANSFER CASK can continue to perform its design function regarding DSC transfer and shielding.