SAFETY EVALUATION REPORT OF VECTRA TECHNOLOGIES, INC. a.k.a. PACIFIC NUCLEAR FUEL SERVICES, INC. SAFETY ANALYSIS REPORT FOR THE STANDARDIZED NUHOMS HORIZONTAL MODULAR STORAGE SYSTEM FOR IRRADIATED NUCLEAR FUEL

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR MATERIAL SAFETY
AND SAFEGUARDS

December 1994

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1.0 INTRODUCTION, GENERAL DESCRIPTION

1.1 Introduction

VECTRA Technologies, Inc. (VECTRA), formerly Pacific Nuclear Fuel Services, Inc. (PNFS) has submitted a Safety Analysis Report (SAR) and supplementary docketed material (Reference 1) to support issuance of a Certificate of Compliance under 10 CFR Part 72, Subpart L (Reference 2). The application does not request NRC approval for installation at any specific site.

The subject of the SAR is the "standardized NUHOMS horizontal modular cask" storage system. For the purposes of this review, the system will be referred to as the "standardized NUHOMS system" to distinguish it from two previous versions of horizontal storage systems designed by NUTECH Engineers. The standardized NUHOMS system is different in many ways from the previous designs. Consequently the NRC staff reviewed all features of the standardized NUHOMS.

10 CFR 72.238 provides that a Certificate of Compliance for a cask model will be issued by NRC on a finding of compliance with 10 CFR 72.236(a) through (i). In addition, 10 CFR 72.234(a) through (f) contain conditions of approval of the spent fuel cask design, including requirements for compliance with 10 CFR 72.236, quality assurance requirements according to Subpart G of 10 CFR Part 72, and other administrative requirements for which the vendor is responsible.

The review focused on the specific requirements for spent fuel storage casks contained in 10 CFR 72.236(a) through (m). These requirements cover fuel specifications, design criteria and administrative aspects. As noted, issuing the Certificate of Compliance will be based on an NRC finding that the requirements in 10 CFR 72.236(a) through (i) are met. Additionally, the staff will address whether the requirement of 10 CFR 72.236(m) has been considered. Issuance of the certificate is also subject to compliance with conditions specified in 10 CFR 72.236(j) through (l) for which the vendor is responsible.

The objectives of this Safety Evaluation Report (SER) are to document the NRC staff's review and evaluation of the Safety Analysis Report (SAR) (Reference 1), and to clearly state the compliance (or noncompliance) of the license application to the applicable requirements of 10 CFR Part 72, Subpart L.

1.2 Context

This SER provides NRC staff analyses and conditions on the SAR submitted by PNFS in conjunction with an application for certification of the standardized NUHOMS system described in the SAR.

The SAR was submitted in accordance with the requirements of 10 CFR Part 72, Subpart L. This SER is based on review for compliance with 10 CFR Part 72. Changes, clarifications, and additional information submitted to the NRC subsequent to the SAR during the review process (as listed at Reference 1) are considered to have the full effect and to express the same type of commitments as if they were included in the SAR itself.

The SAR presumes that the standardized NUHOMS system will be used on the site of a nuclear power reactor licensed by the NRC under 10 CFR Part 50, and that fuel loading and unloading will occur within a fuel pool of the licensed facility. The SER does not include identification of additional requirements should fuel loading and unloading not be within the fuel pool of a facility licensed under 10 CFR Part 50.

Information incorporated by reference or included by subsequent submittal (Reference 1) is considered as if it were information set out in the SAR itself. Where such information is already the subject of NRC staff approval, as by approval of a SER (e.g., References 3 and 4), that approval is considered to extend to the document incorporating the information by reference, to the extent of such incorporation, and subject to any qualifiers included in the referenced document and/or the corresponding SER.

Use of the proposed standardized NUHOMS system will include operations and use of equipment related to safety within a fuel pool of the facilities licensed under 10 CFR Part 50. Fuel handling operations for an independent spent fuel storage installation (ISFSI) may require amendments to existing license technical specifications for the facility licensed under 10 CFR Part 50. This SER does not constitute the formal safety evaluation review for the safety of operations and equipment within fuel pool facilities. This SER does, however, examine the suitability of the transfer cask and DSC for mutual compatibility and for satisfaction of 10 CFR Part 72, Subpart L requirements.

The proposed standardized NUHOMS system uses designs for its components which have evolved from designs in use or under construction as ISFSIs at existing facilities. The approval of these ISFSIs have involved NRC SERs for topical reports and license application SARs prepared in compliance with 10 CFR Part 72 and Regulatory Guide 3.48 (Reference 5). These documents have provided a context to the review which assisted in determining suitable criteria and design acceptability.

1.3 General Discussion of Reference Materials and Role of Inspection

This SER refers in several places to fabrication specifications and engineering drawings for major components of the standardized NUHOMS system. The following paragraphs provide a general explanation for these references; they indicate the referenced specifications and drawings were not a basis for the staff's safety approval of a particular design topic in the SER. Rather, the staff reviewed aspects of the specifications and drawings to verify they accurately incorporated information that was part of the staff's basis for approving the cask design.

In basic terms, the cask vendor's design commitment, contained in codes, standards, and design criteria, is identified in the SER and serves as a design input for the vendor's design calculations. The vendor's calculations both demonstrate compliance with design inputs and produce design details, e.g., reinforcing steel sizing, shield lid thickness, and many other results called design outputs. Much of the design output is contained in the vendor's engineering drawings and fabrication specifications. These drawings and fabrication specifications provide the vendor's constructor and component fabricator with detailed instructions for constructing the standardized NUHOMS system and its components. These drawings and specifications are not approved by the NRC as a part of the staff's review of the vendor's standardized NUHOMS system.

As reflected in the SER, from the vendor's entire set of design information, the staff's design approval mainly relies on the vendor's criteria and design commitments and certain calculations or parts of calculations. The staff generally uses these portions of the vendor's design information to conduct an independent review and analysis to determine whether there is reasonable assurance that the vendor's design will perform its intended safety function.

Another aspect of the staff's activities reflected in the SER, separate from and related to its safety review, is the delineation of the requirements for NRC inspections. For instance, the staff may prepare inspection procedures for the regional or headquarters vendor inspection staffs to conduct certain types of inspections of spent fuel storage cask vendors, fabricators, and constructors. These inspection procedures may specify, in addition to the information in the procedures and the design commitments contained in the SER, that inspectors should use information in the vendor's fabrication specifications, engineering drawings, procurement documents, and material certifications to perform their field inspections.

Where the inspection procedures refer to the vendor's drawings and specifications, the staff has typically reviewed selected aspects of the vendor's drawings and fabrication specifications to verify that the results contained in the vendor's design calculations have been accurately transposed into the drawings and specifications. By so doing, the staff provides added assurance that the inspectors will have accurate documentation to inspect the adequacy of construction. It is important to note that this NRC inspection does not constitute an additional NRC review of the standardized NUHOMS system design or a further NRC safety determination of the adequacy of the standardized NUHOMS system design. Rather, inspection activities address the adequacy of component construction, fabrication, and quality assurance (QA). Therefore, as previously noted, while the staff did not rely upon the fabrication specifications or drawings in approving the design, the SER will reflect the staff's check of portions of these documents to verify they contain accurate design output information to be used by the fabricator and checked by NRC inspectors.

A further NRC check on the validity of the design output information is through QA requirements that review, approve, and link the individual QA programs of utility, vendor, fabricator, and constructor. Among other things, these QA programs ensure the control of changes to drawings and specifications for accuracy and ensure proper engineering review.

10 CFR 72.234(a) through (f) which contain the conditions of approval for spent fuel cask design, require compliance with the specific design criteria of 72.236 and the quality assurance requirements in subpart G and identify other administrative requirements for which the system vendor is responsible.

10 CFR 72.236(a) through (m) contain the specific requirements for spent fuel storage cask approval, including spent fuel specification, design criteria, and administrative requirements. As noted, the Certificate of Compliance is issued by the NRC on a finding that the requirements of 72.236(a) through (i) are met, and after the staff determines that the requirement of 72.236(m) has been considered. The issuance is also subject to the conditions specified in 72.236(j), (k), and (l) for which the vendor is responsible.

1.3.1 General Description of Standardized NUHOMS System

The following descriptions of the standardized NUHOMS system are based on the more complete descriptions provided by Reference 1 and are only included here for the convenience of readers of the SER. The SER is based on the descriptions provided in the SAR. The standardized NUHOMS system components for irradiated fuel assemblies (IFA) storage at an ISFSI are the Dry Storage Canister (DSC) and the Horizontal Storage Module (HSM). Additional systems required for the DSC closure and transfer include the transfer cask (TC), the skid and skid positioning system, the trailer, the hydraulic ram system, and the DSC vacuum drying system.

1.3.2 Horizontal Storage Module

The standardized NUHOMS system uses HSMs assembled from standardized units, as illustrated in Figure 1.1. These are:

- Base Unit Assembly, consists of the monolithically poured reinforced concrete (RC) base unit of floor and four walls, with DSC access opening, inlet and outlet ventilation openings, and embedments for attachment of restraints, the DSC support structure roof slab, heat shields, spacers and shield walls. The base unit side walls are 0.46 m (1'-6"), the front wall is 0.76 m (2'-6"), and the rear wall and floor are 0.30 m (1') thick.

- DSC Support Structure, a structural steel frame with rails installed within the base unit to provide for sliding the DSC in and out of the HSM, supports the DSC within the HSM, and resists and transfers forces associated with a jammed DSC or a design basis earthquake.

- Roof Slab Assembly, a rectangular 0.91 m (3 foot) thick RC slab which is bolted to the base unit to complete the shielded enclosure for DSC storage. It includes embedments for attachment to the base unit for positioning, for lifting, and for attachment of screens between adjacent modules and between modules and external separate shield walls.

- Second Shield Wall, a rectangular 0.61 m (2 foot) thick RC slab installed vertically at the outer side of HSM at the ends of rows of HSMs. The end shield walls are installed with channel spacers, shielded bolt assemblies attaching them to the HSM, and screens across the gaps between the walls.

- Single Module Rear Shield Wall, used when HSMs are placed in single rows (the alternative placement is with two rows back-to-back). The rear shield wall is a rectangular 0.46 m (1'-6") thick RC slab installed vertically against a base unit rear wall without an intervening space. The rear shield walls are installed with shielded bolt assemblies.

- Shielded Door, composed of a 5.1 cm (2") thick steel plate and 14.9 cm (5-7/8") of RC, which closes the DSC access opening and provides radiation shielding and resistance to natural phenomena.

- Basemat, cast-in-place RC foundation on which the HSMs rest. The HSMs are not connected to the basemat and are held in place against any horizontal forces by friction. Thickness of the basemat is to be determined by site foundation analysis.

- Approach Slabs, a cast-in-place RC slab providing for access and support of the DSC transport and transfer systems. This slab is structurally connected to the Basemat. Thickness of the approach slab is to be determined by site foundation analysis.

The HSM protects the DSC from the potentially adverse effects of natural phenomena, such as earthquake, tornado, tornado missiles, flood, and temperature.

The modular HSM system is considered acceptable for layout variations from a single HSM to unrestricted numbers of HSMs in single or back-to-back rows, without additional shielding, as approved in this SER, if criteria of the SAR are also met.

The HSM dissipates decay heat from the spent fuel by a combination of radiation, conduction, and convection. Natural convection air flow enters at the bottom of the HSM, circulates around the DSC, and exits through the flow channels between the HSM roof slab and side walls. A thermal radiation shield is used to reduce the HSM concrete temperatures to within acceptable limits for all conditions.

1.3.3 Dry Shielded Canister

The DSC is illustrated in Figure 1.2. A DSC is shown in its storage position in Figure 1.1. The principal component subassemblies of the DSC are the shell with integral bottom cover plate and shield plug and ram/grapple ring, top shield plug, top cover plate, and basket assembly. The main component of construction of the DSC is a type 304 stainless steel cylindrical confinement vessel.

The internal basket assembly for the PWR fuel is comprised of 24 guide sleeves supported by 8 spacer discs at intervals corresponding to the fuel assembly spacer grids. Support rods maintain the spacer disks in location. The internal basket assembly for the channelized BWR fuel is similar to the PWR except that 52 guide sleeves are used for the BWR application, supported by 9 spacer discs. Borated stainless steel poison plates are used for all BWR baskets. Steel shielding is used in both the top and bottom end shield plugs.

Criticality safety during wet loading operations for the PWR fuel is maintained through the geometric separation of the fuel assemblies within the internal basket assembly, the inherent neutron absorption capability of the steel guide sleeves, the proper selection of sufficiently depleted fuel assemblies, and adequate boron concentration in the pool water. For BWR fuel assemblies, criticality safety during wet loading operations is maintained by similar means except that borated stainless steel plates are used in the guide sleeve assemblies and borated water is not required. Credit for burnup is not currently permitted by the NRC staff.

The DSC provides mechanical confinement for the stored fuel assemblies and all radioactive materials for two purposes: to prevent the dispersion of particulate or gaseous radionuclides from the fuel, and to maintain a barrier of helium around the fuel in order to mitigate corrosion of the fuel cladding and prevent further oxidation of the fuel.

The DSC provides radiological shielding in both axial directions. The top shield plug serves to protect operating personnel during the DSC drying and sealing operations. The bottom shielding reduces the HSM door area dose rates during storage. The DSC shielding is designed for a maximum contact dose of 2 mSv/hr (200 mrem/hr) before draining the DSC cavity.

The DSC is designed to slide from the transfer cask into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces. This is accomplished by a combination of surface finishes and dry film lubricant coatings applied to the DSC and the DSC support assembly in the HSM. The transfer operation is illustrated in Figure 1.6.

1.3.4 Transfer Cask

The principal components of the transfer cask (TC) are shown in Figure 1.3 (SAR Figure 1.3-2b). Figures 1.4 and 1.5 (SAR Figures 4.2-9 and 4.2-9a) show the TC with DSC. Figure 1.6 (SAR Figure 1.1-2) shows the TC in position for DSC transfer to the HSM.

The transfer cask is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. The cask's cylindrical walls are formed from three concentric steel shells with lead poured between the inner liner and the structural shell to provide gamma shielding during DSC transfer operations. The structural and outer shells form an annular pressure vessel. A solid neutron absorbing material is cast between the structural shell and outer shell to provide neutron shielding when the DSC is in the TC.

The cask bottom end assembly is welded to the cylindrical shell assembly. It includes two closure assemblies for the ram/grapple access penetration. A watertight bolted top cover plate, with a core of solid neutron absorbing material, is used for transfer operation within the Auxiliary Building (or Spent Fuel Storage Building in some plants). The bolted ram access penetration bottom cover plate assembly is replaced, after the TC is horizontal on the transport trailer and while still in the Auxiliary Building, by a two-piece neutron shield plug assembly for transfer operations from/to the Auxiliary Building to/from the HSM. The inner plug of this assembly is bolted to the TC. The outer plug is held in brackets by gravity. At the HSM site, the outer plug of the assembly is removed to provide access for the ram/grapple to push/pull the DSC into/from the HSM.

The top plate cover is bolted to the top flange of the cask during transport from/to the Auxiliary Building to/from the ISFSI. The top cover plate assembly consists of a thick structural plate with a thin shell encapsulating solid neutron shielding material. Two upper lifting trunnions are located near the top of the cask for downending/uprighting and lifting of the cask in the Auxiliary Building. Two lower trunnions, located near the base of the cask, serve as the axis of rotation during downending/uprighting operations and as supports during transport to/from the ISFSI. The TC is not designed as a pressure vessel.

The neutron shield material is BISCO Products NS-3. NS-3 is a shop castable, fire resistant material with a high hydrogen content which is designed for nuclear applications. The material is used in the cask outer annulus, top and bottom covers, and temporary shield plug. It produces water vapor and a small quantity of non-condensible gases when heated above 100°C (212°F). The off-gassing produces an internal pressure which increases with temperature. As the temperature is reduced, the off-gas products are reabsorbed into the matrix, and the pressure returns to atmospheric. The annular neutron shield containment is designed for an internal pressure of 655 kPag (95 psig). Pre-set safety relief valves are included to protect the neutron shield cover in the event that its design pressure is exceeded.

1.3.5 Fuel Transfer Equipment

With the exception of the TC, fuel transfer and auxiliary equipment necessary for ISFSI operations are not included as a part of the standardized NUHOMS system to be reviewed for a Certificate of Compliance under 10 CFR 72, Subpart L. However this equipment will be described for general information only. Fuel is transferred in ISFSI operations by means of the TC. Inside the fuel pool facility, the TC with loaded DSC is transferred from the fuel pool to a position where decontamination, drying, sealing, and installation of the TC cover take place. The TC and DSC are then transferred to the transfer trailer, still within the Auxiliary Building. The TC with DSC is moved to position for coupling with the HSM access opening by the transfer trailer, with final positioning by movement of the TC support shield over the trailer. The DSC is transferred from the TC to the HSM by use of the ram acting through the ram access opening of the TC.

Equipment used to physically grip, lift, inspect, and position the IFAs in the fuel pool is the same as that already in place and in use for Auxiliary Building IFA handling.

There is special equipment involved with fuel transfer within the Auxiliary Building unique to the ISFSI application. Of this, only the TC lifting yoke is used exclusively within the Auxiliary Building and is thereby subject to evaluation as part of the 10 CFR Part 50 license review of updates to the FSAR.

The lifting yoke is a special lifting device which provides the means for performing all cask handling operations within the plant's Auxiliary Building. It is designed to support a loaded transfer cask weight up to 90.7 t (100 tons). A lifting pin connects the Auxiliary Building cask handling crane hook and the lifting yoke. The lifting yoke is a passive, open hook design with two parallel lifting beams fabricated from thick, high-strength carbon steel plate material with a decontaminable coating. It is designed to be compatible with the Auxiliary Building crane hook and load block. The lifting yoke engages the outer shoulder of the transfer cask lifting trunnions. To facilitate shipment and maintenance, all yoke subcomponent structural connections are bolted.

Lifting slings are used in the Auxiliary Building for placement and removal of the DSC and TC shield plugs and covers. Eyebolts are installed on the items to be lifted to facilitate rigging for lifting.

The transfer trailer is used to transport the transfer cask skid and the loaded transfer cask from the Auxiliary Building to the ISFSI. The transfer trailer is an industrial heavy-haul trailer with pneumatic tires, hydraulic suspension and steering, and brakes on all wheels. Four hydraulic jacks are incorporated into the transfer trailer design to provide vertical elevation adjustment for alignment of the cask at the HSM. The transfer trailer is shown in Figure 1.6. It is pulled by a conventional tractor.

The trailer is pulled using a drawbar steering unit. The steering unit includes hydraulic master cylinders to provide motive force for the slave steering cylinders in the trailer. The trailer may also be steered manually using a remote steering control located on a pendant. This feature allows precise control as the trailer is backed up to the HSM. The pendant allows the operator the freedom to observe the trailer from the side and also reduces the operational exposure by increasing operator distance from the DSC and reducing operator time.

The trailer incorporates a skid positioning system which holds the TC support skid. The functions of the skid positioning systems (SPS) are to hold the TC support skid stationary (with respect to the transport trailer) during cask loading and transport, and to provide alignment between the transfer cask and the HSM before insertion or withdrawal of the DSC. It is composed of tie down or travel lock brackets, bolts, three hydraulically powered horizontal positioning modules, four hydraulic lifting jacks, and a remotely located hydraulic supply and control skid.

The hydraulic jacks are designed to support the cask setdown load and the loads applied to them during the HSM loading and unloading. Their purpose is to provide a solid support for the trailer frame and skid. Three measures are taken to avoid accidental lowering of the trailer payload: the hydraulic pump will be de-energized after the skid has been aligned (the jacks are also hydraulically locked out during operation of the horizontal cylinders); there are mechanical locking collars on the cylinders; and pilot-operated check valves are located on each jack assembly to prevent fluid loss in the event of a broken hydraulic line.

Three positioning modules provide the motive force to horizontally align the skid and cask with the HSM before insertion or retrieval of the DSC. The positioning module controls are manually operated and hydraulically powered. The system is designed to provide the capability to align the cask to within the specified alignment tolerance.

The hydraulic power supply and controls for the SPS are located on a skid which is normally stored on the hydraulic ram utility trailer. Directional metering valves are used to allow precise control of cylinder motions. The SPS is manually operated and has three operational modes: simultaneous actuation of the four vertical jacks or any pair of jacks, actuation of any single vertical jack, or actuation of any one of the three horizontal actuators. Simultaneous operation of the vertical jacks and the horizontal actuators is not possible. Fourteen small hydraulic quick-connect lines provide power to the seven SPS hydraulic cylinders.

The hydraulic ram system provides the motive force for transferring the DSC between the TC and the HSM. The hydraulic ram consists of a double-acting hydraulic cylinder with a capacity of 36,290 kg (80,000 lb.) in either push or pull mode and stroke of 6.4 m (21 feet). The ram will be supported during operation by a frame assembly attached to the bottom of the transfer cask and a tripod assembly resting on the concrete slab. The operational loads of the hydraulic ram are grounded through the transfer cask. The hydraulic ram system includes a grapple at the end of the piston which is used to engage a grapple ring on the DSC for retrieval operations. Figure 1.6 shows main components of the hydraulic ram system (SAR Figure 1.1-2).

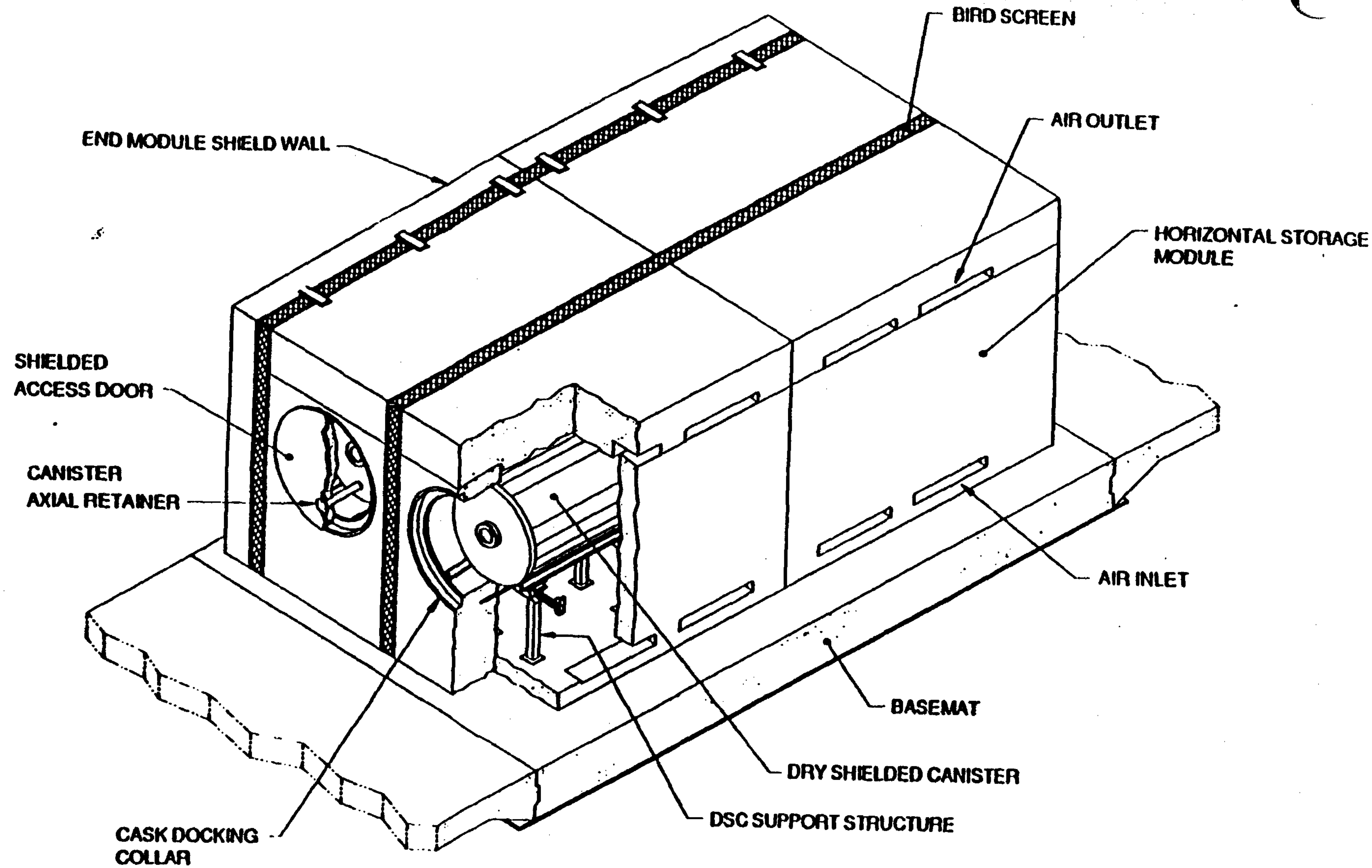

Figure 1.1

NUHOMS® Horizontal Storage Module Arrangement

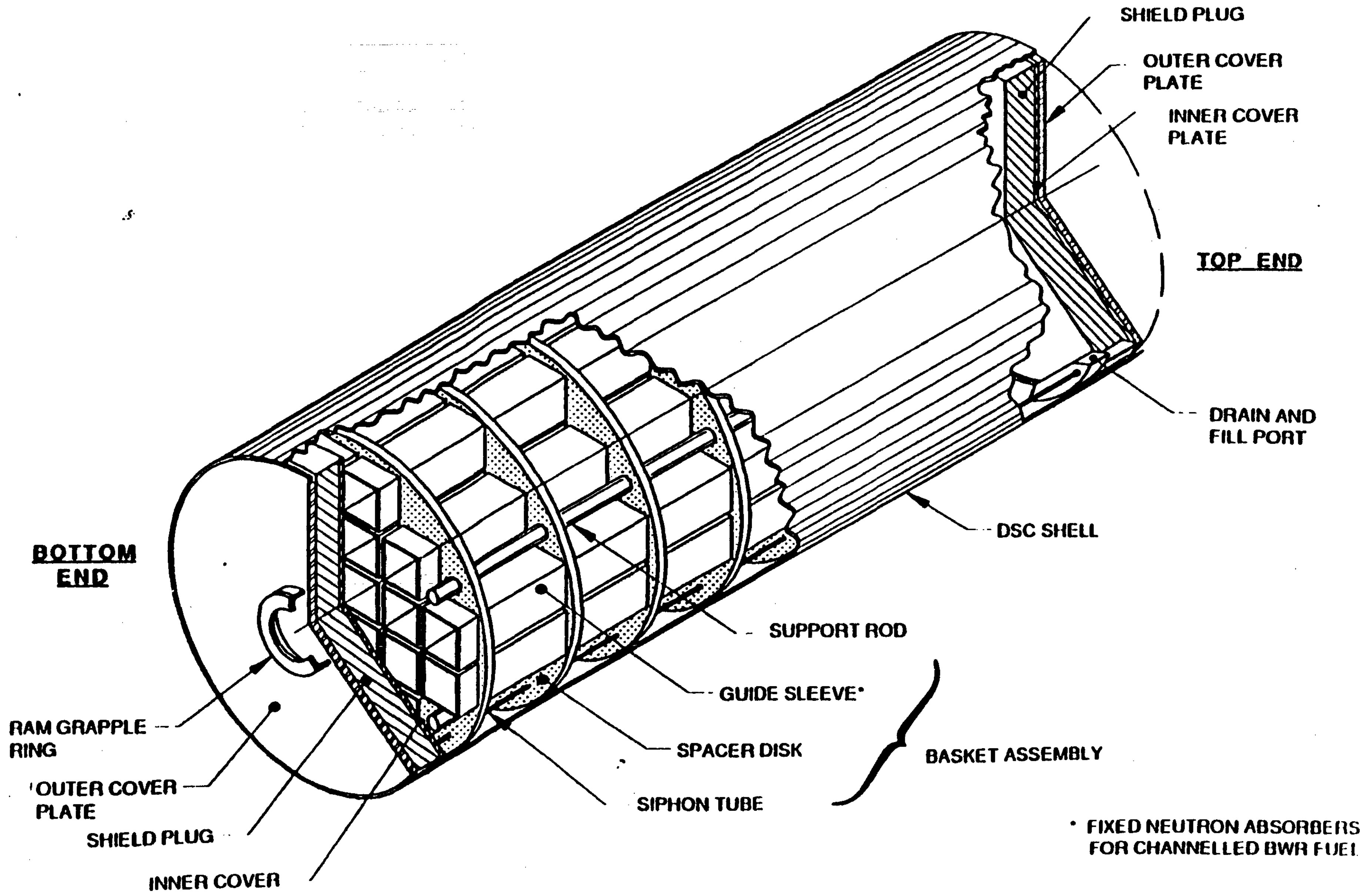

Figure 1.2

NUHOMS® Dry Shielded Canister Assembly Components

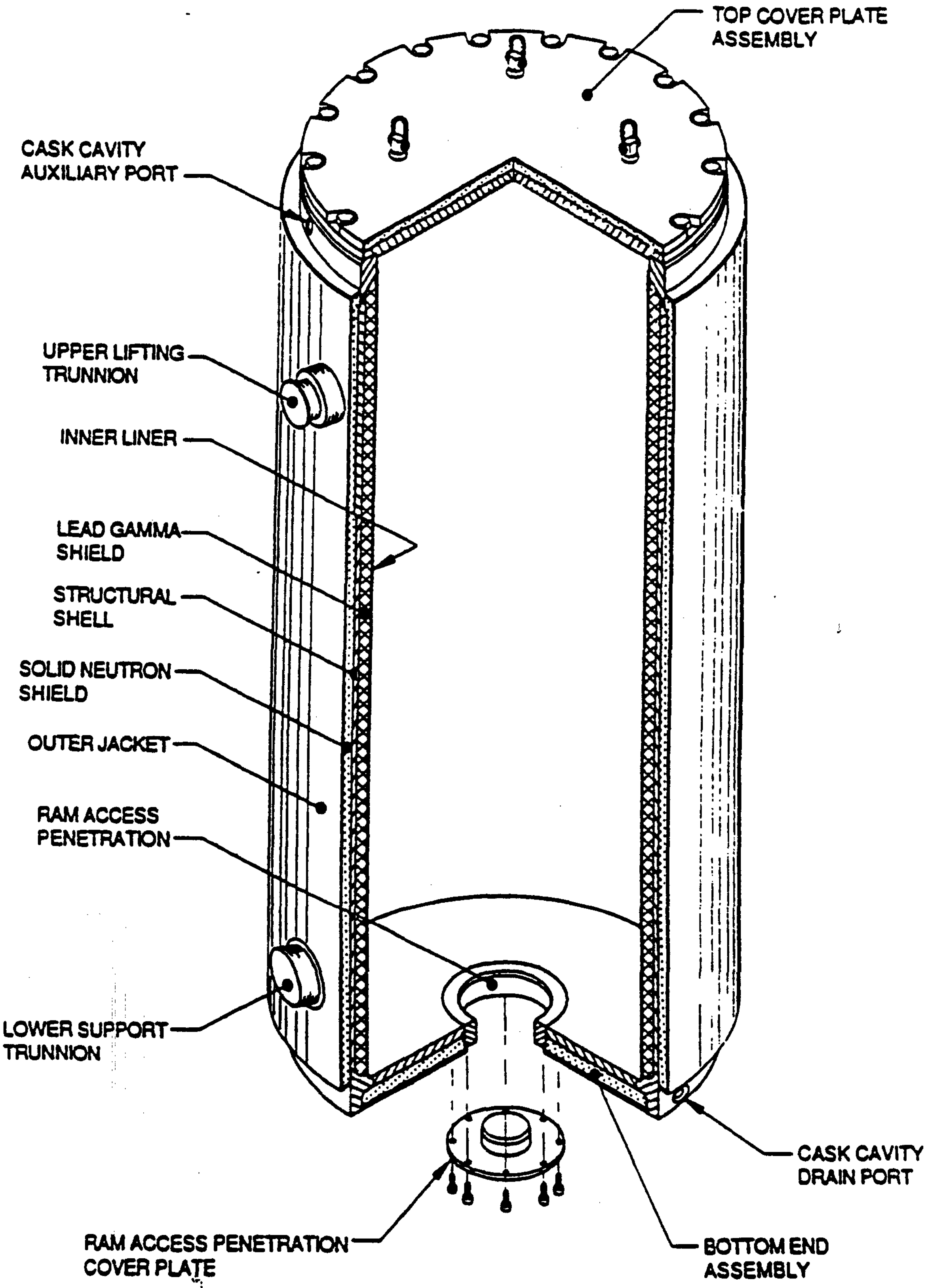

Figure 1.3

NUHOMS® On-Site Transfer Cask

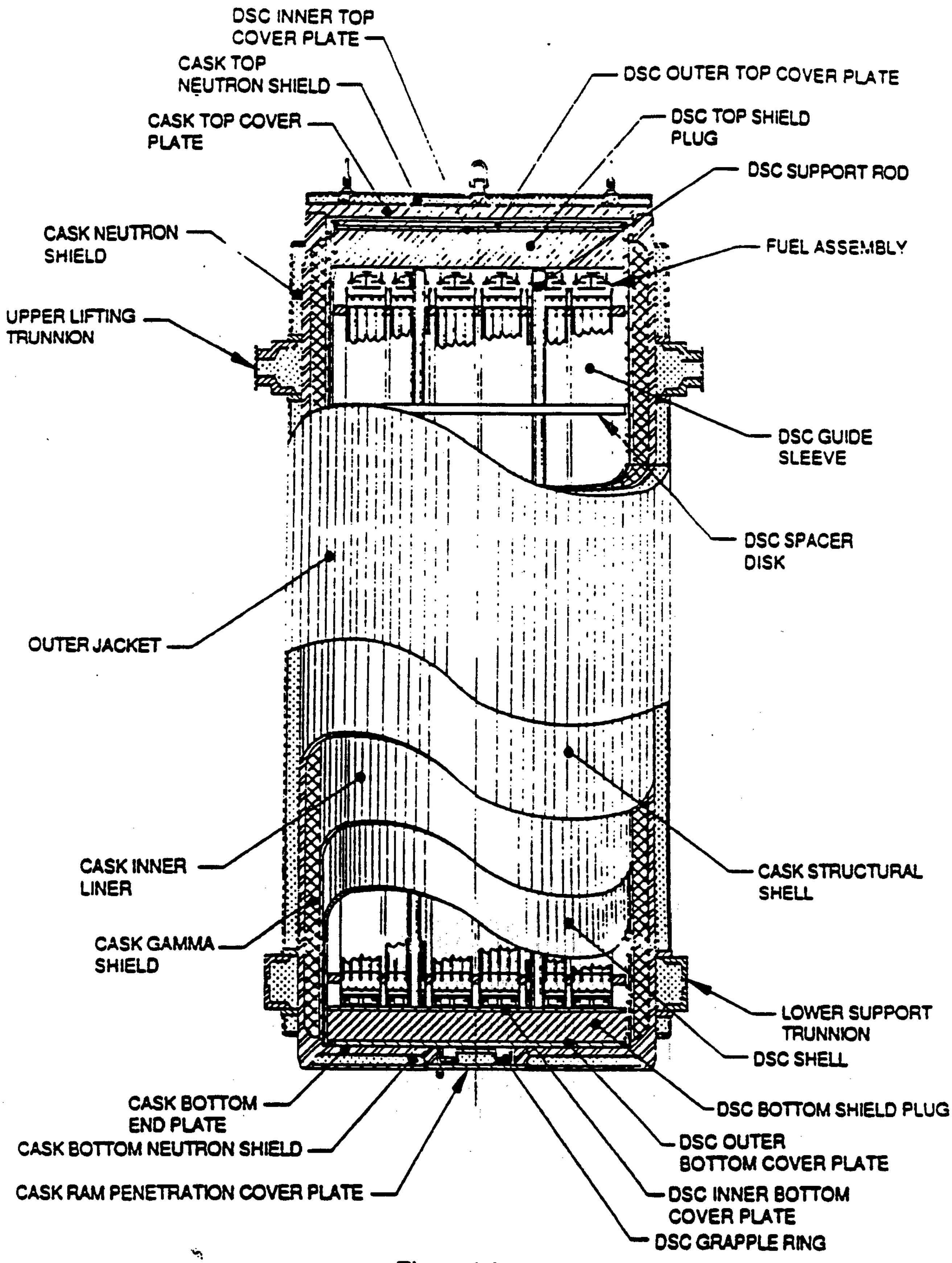

Figure 1.4

Composite View of NUHOMS® Transfer Cask and DSC with Spent PWR Fuel

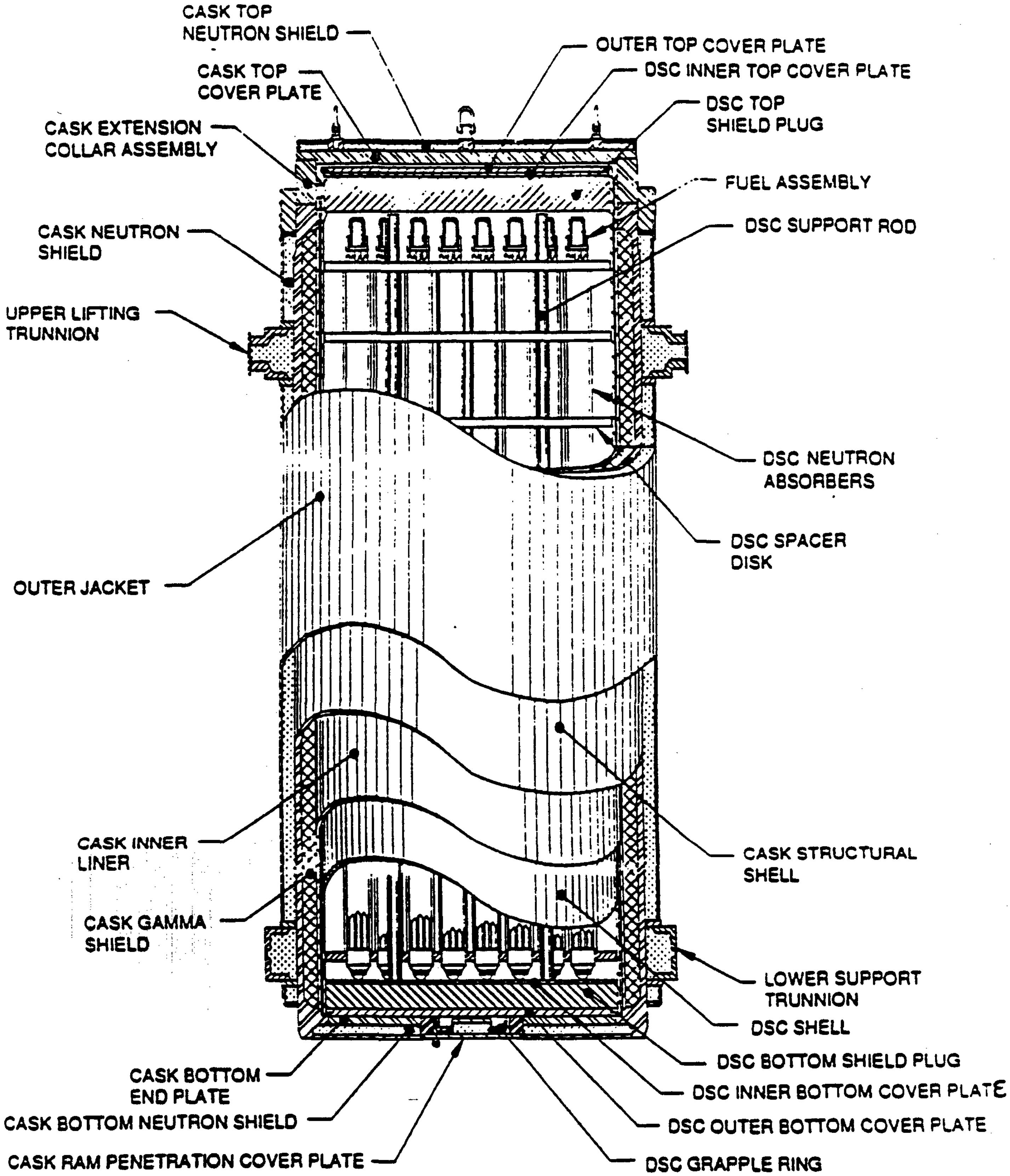

Figure 1.5

Composite View of NUHOMS® Transfer Cask and DSC with Spent BWR Fuel

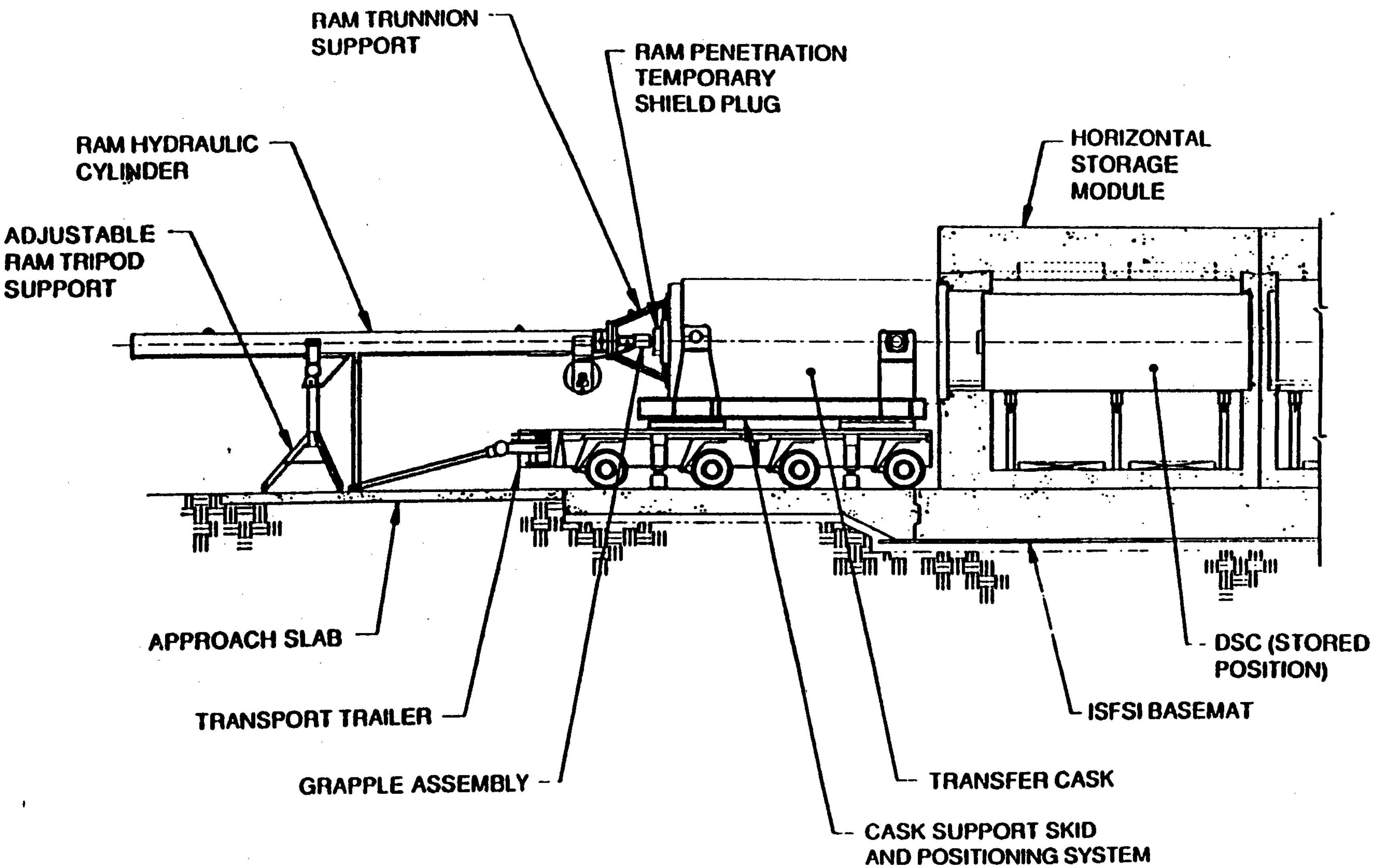

Figure 1.6

NUHOMS® System Components, Structures, and Transfer Equipment Elevation View

2.0 PRINCIPAL DESIGN CRITERIA

2.1 Introduction

10 CFR 72.236(b) requires that design bases and design criteria must be provided for structures, systems and components important to safety. The criteria for the design, fabrication, construction, testing and performance of components important to safety are set forth in the general requirements of 10 CFR 72.236(a) through (i). In addition, this SER addresses the staff's consideration of design criteria in 10 CFR Part 72, Subpart F, "General Design Criteria For Independent Spent Fuel Storage Facilities (ISFSI)."

The following subsections discuss the design criteria applied by the NRC staff to the standardized NUHOMS system and the degree to which the design as described in the SAR is in compliance with these criteria. The subsection headings generally correspond to criteria in 10 CFR 72, Subpart F and 10 CFR 72.236.

Section 3.0 of the SAR contains the design criteria proposed by the standardized NUHOMS system vendor. It also identifies sources for these design criteria. The sources and their acceptability are summarized in Table 2.1 of this report. As shown in the table, the identified sources were determined to be acceptable.

2.2 Fuel to be Stored

10 CFR 72.236(a) requires that a specification for the spent fuel to be stored in the cask be provided, including type of spent fuel (i.e., BWR, PWR, both), maximum allowable enrichment of the fuel before irradiation, burn-up, minimum acceptable cooling time of the spent fuel before storage in the cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel, and inerting atmosphere requirements.

The fuel specified to be stored in the standardized NUHOMS system is intact (not consolidated) PWR or BWR, with physical characteristics presented in Table 12-1a and 12-1b of the SER. The related characteristics and parameters of the fuel to be stored are determined by assumptions used by the vendor to analyze and evaluate the capabilities of the system, such as criticality safety, shielding, heat removal, confinement, and the limitations imposed by these analyses in meeting acceptance criteria. The fuel specification, accepted by the staff, is presented in Section 12.2.1 of this report.

2.3 Quality Standards

The quality standards considered by the staff for the spent fuel storage system are in 10 CFR 72.122(a) and in 10 CFR 72.234(b). 10 CFR 72.122(a) provides that structures, systems and components important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.

10 CFR 72.234(b) requires that the design, fabrication, testing and maintenance of spent fuel storage casks must be conducted under a quality assurance program that meets the requirements of Subpart G of 10 CFR 72.

Quality standards dealing with the design, materials, fabrication techniques, inspection methods, etc., are cited by the vendor in the sections of the SAR where the standards are applicable. Judgments regarding the adequacy of these standards are also presented in the corresponding sections of this report.

The Quality Assurance program proposed by the vendor for the design, fabrication and construction of the standardized NUHOMS system is presented in Section 11.0 of the SAR. The staff's evaluation of the vendor's Quality Assurance program is discussed in Section 10.0 of this report.

2.4 Protection Against Environmental Conditions and Natural Phenomena

10 CFR 72.236(b) requires design bases and design criteria for structures, systems and components important to safety. 10 CFR 72.122(b) provides that structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. It also provides that structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions.

The standardized NUHOMS system is intended to withstand environmental conditions and natural phenomena that may occur under "normal," "off-normal," and "accident" circumstances. Normal operations, off-normal operations, and accidents are described in NRC Regulatory Guide 3.62 (Reference 72). In Regulatory Guide 3.62, normal operations of an ISFSI consists of the set of events that are expected to occur regularly or frequently. Off-normal operations consist of the set of events that, although not occurring regularly, can be expected to occur with moderated frequency or on the order of once during a calendar year. Design criteria call for normal and off-normal conditions to both satisfy allowable limits for routine operations. Accidents are any credible incident that could result in a potential radiation dose of 25 mrem or more beyond the controlled area and situations wherein direct radiation or radioactive materials may be released in such quantities as to endanger personnel within the controlled area. Accidents include two sets of events. The first consists of that set of infrequent events that could reasonable be expected to occur during the lifetime of the ISFSI. The second is concerned with natural phenomena or low probability events that are postulated because their consequences may result in the maximum potential impact on the immediate environs. Their consideration establishes a conservative design basis for certain systems with important to safety.

The principal design criteria used for structural design are listed in Tables 2.2 (normal), 2.3 (off-normal), and 2.4 (accident). Column (5) of Table 2.2, 2.3, and 2.4 includes comments on applicability of the criteria for use of the standardized NUHOMS system at possible sites. Some criteria are acceptable and are not affected by the actual site. Some criteria vary by location, but the values used in the SAR are sufficient to envelope the values that may be appropriate for credible sites in the continental United States. Other criteria should be verified as bounding the proper values for the specific installation location.

The SAR does not present a HSM foundation design for certification. The foundation design shown in the drawing is considered a nominal design for illustration. This foundation design is probably adequate or conservative for many potential sites given appropriate site preparation. A foundation analysis should be performed for verification of the adequacy of the nominal design or to provide the basis for a new design specific to the installation.

The foundation is not relied upon to provide safety functions. There are no structural connections or means to transfer shear between the HSM base unit module and the foundation slab. However, the user must evaluate the foundation in accordance with 10 CFR 72.212(b)2 and (b)(3) to ensure that, in an unlikely event, no gross failures would occur that would cause the DSC to jam during transfer operation, or cause the standardized NUHOMS system to be in an unanalysed situation, and would prevent removal of a DSC from a HSM.

Evaluation of an ISFSI design is accomplished by evaluating the stated criteria and the actual design as separate review stages. Criteria may be acceptable, but if the actual design does not meet the criteria the system may not be acceptable. Similarly, some proposed criteria may not be acceptable, however, because of conservatism in the actual design, it may be determined to satisfy more stringent, yet acceptable, criteria.

2.4.1 Normal Operating Conditions

The staff considers that the design criteria as stated and referenced in Table 2.2 are acceptable for certification with the following exception: in principle, the DSC should be considered a live load rather than treated as a dead load as in the SAR. The weight of the DSC is precisely known; however, any additional loads associated with its transfer are treated as live loads. Based on staff review of the design, factors of safety, and impact of treatment as a dead load, the usage in the SAR is accepted. It has been determined that the factor of safety, if the DSC were treated as a live load, would still be acceptable for the actual design.

Section 8.1.1.1 of the SAR states that the long-term average yearly ambient temperature is assumed to be 21°C (70°F). This temperature bounds most, but not all, reactor sites in the Continental United States. Reactor sites which exceed this temperature are Palo Verde, Turkey Point, and St. Lucie.

The SAR states that the design basis operating temperatures are -40°C to 52°C (-40°F to 125°F). While this temperature range is acceptable for storage, it is not acceptable for on-site transfer or lifting and handling of the DSC. Paragraph 10.3.15 of the SAR states that the minimum ambient temperature for transfer of the loaded DSC inside the TC manufactured from ferritic steel shall be -17.8°C (0°F). Furthermore, if the ambient temperature exceeds 37.8°C (100°F), a solar shield shall be provided to protect the solid neutron shield material contained in an annular volume of the TC. No lifting above 203 cm (80 inches) of the loaded DSC is permissible below -28.9°C (-20°F) inside the spent fuel pool building. If the lift height exceeds 203 cm (80 inches), then the minimum temperature is restricted to -17.8°C (0°F) for lifting inside the spent fuel pool building. The appropriate criteria for impact testing of ferritic steels for the DSC shell or basket is NUREG/CR-1815 (Reference 7).

Similarly, the SAR states that the design basis operating temperatures are -40°C to 52°C (-40°F to 125°F) for the TC. This temperatures range is acceptable for handling the empty TC; however, for lift heights of 203 cm (80 inches) or lower the minimum limiting temperature for handling a TC with a loaded DSC shall be restricted to -28.9°C (-20°F). This limiting minimum temperature shall apply inside the spent fuel pool building. For lift heights above 203 cm (80 inches) inside the spent fuel pool building, the minimum temperature is restricted to -17.8°C (0°F). For all transfer operations outside the spent fuel pool building, the maximum height is limited to 203 cm (80 inches) and the minimum temperature is limited to -17.8°C (0°F). The appropriate criteria for impact testing of ferritic steels for the TC is ANSI N14.6 paragraph 4.2.6 (Reference 8).

2.4.2 Off-Normal Operating Conditions

Table 2.3 lists summary design criteria used for off-normal operating conditions. The staff considers that the design criteria stated and referenced in Table 2.3 are acceptable and appropriate with the following exceptions:

- The jammed condition loading for the DSC support assembly is properly listed as an off-normal condition; however, it is actually used in the load combinations as though it were an "accident" loading. The NRC requires normal and off-normal loads to be evaluated similarly in load combination expressions without use of the increases in stresses permitted for "accident" type loads.

- The DSC support assembly does not have criteria identified for off-normal temperature rise. This is considered not acceptable, however the actual design is determined to satisfy criteria which should have been listed.

- Identical minimum service temperature restrictions apply to the transfer of the loaded DSC and the TC with a loaded DSC, as described in 2.4.1 above.

2.4.3 Accident Conditions

Table 2.4 lists summary design criteria used for accident conditions. These conditions include extreme natural phenomena, accidental drops and impacts, fire, and explosions.

The staff considers that the design criteria as stated are acceptable with the exception that the jammed loading condition was treated as an accident (discussed above). Where other criteria as determined by the staff are considered more appropriate, the criteria are stated in the table and are determined to be conservative and thereby acceptable.

The following accident design criteria should be verified as equal to or exceeding appropriate parameters for the actual installation site.

- Flood parameters, especially the 4.6 m per second (15 foot per second) maximum velocity.
- Seismic maximum horizontal and vertical ground accelerations.
- Maximum ambient temperature.
- Potential for fire or explosion, Section 2.5, below.
- Requirements for lightning protection.
- Extreme low temperature, see Section 3.0, below.
- Maximum lift height of loaded DSC to 203 cm (80 inches) outside the spent fuel pool.

It is recognized that some other environmental condition limits used in the SAR and SER may not envelope all points within the continental United States. These are not included in the above list of criteria to be verified due to their negligible impact on the design and due to the otherwise unsuitability of the exceptional locations for ISFSI.

2.4.4 Load Combinations

Load combinations presented in the SAR for use in verification of design are presented for the HSM in Table 2.5 and for the DSC Support Assembly in Table 2.6. Staff comments are included in the tables on the acceptability and use of the load combinations.

As annotated in the tables, the load combinations used and omitted are considered acceptable with the exception of that used for the "off-normal" case of a jammed DSC loading the DSC support assembly (Table 2.5). The combinations used (numbers 15 and 16) have factored strengths (1.7 for strength or stress in other than shear, 1.4 for shear) which are not appropriate for off-normal loads.

In addition to specifying load combinations to be used for the design of the HSM, the SAR also specified design load combinations for the DSC and the TC. In both cases, parts of the ASME B&PV Code Section III are used (Reference 9). These definitions of normal, off-normal, and accident operations are discussed. Tables 8.1-1 and 8.1-1a in the SAR define types of loads for all components for normal and off-normal conditions respectively.

Normal loads for the DSC shell include deadweight, internal pressure, thermal loads, and normal handling loads. The DSC basket is not subjected to internal pressure loads. Normal loads for the TC include deadweight, thermal, normal handling and live loads. These load combinations correspond to Service Level A in the ASME Code.

Off-normal loads for the DSC shell include deadweight, internal pressure, off-normal temperature loads and off-normal handling loads. The DSC internals are subjected to deadweight and off-normal thermal loads. The TC is subjected to combined loads including deadweight, off-normal thermal and off-normal handling loads. These load combinations correspond to Service Level B in the ASME Code.

Table 8.2-1 of the SAR defines the various load combinations for the accident loads, or Service Levels C and D of the ASME Code. The effects of loads are considered separately. For example, a design basis earthquake at the ISFSI site (.26g), followed by the design basis flood 17m (50 ft) submersion would not be additive. The accidents considered for the DSC include: earthquake, flood, accidental drop, blockage of HSM inlet and outlet vents, and accidental internal pressure. The accidents considered for the TC include: tornado wind and tornado wind driven missiles, earthquake, accidental drop, and loss of cask neutron shield. In evaluating whether the cask design bases envelope site parameters under 72.212(b)(3), the licensee must consider all conditions that are a consequence of the occurrence of the accident conditions under evaluation.

2.5 Protection Against Fire and Explosion

10 CFR 72.172(c) contains criteria for fire protection which the staff has considered. Section 3.3.6 of the SAR addresses the credibility of ISFSI initiated fires and explosions.

As noted in Table 2.4, the SAR states that design criteria for fire or explosion are "enveloped by other design events." This SER evaluation recognizes that the probability of a fire or explosion affecting standardized NUHOMS system nuclear safety varies with potential installation sites, procedures and equipment used for transfer actions, and possible accidents at or in the vicinity of the system (e.g., aircraft and vehicle crashes, railroad, truck, or pipeline fires and explosions).

The SAR did not identify specific criteria for fire and explosion. It stated that such events were bounded by other criteria. Externally initiated explosions are considered in the SAR to be bounded by design basis tornado generated missiles. The DSC can withstand the external pressure of a flood of a head of water equal to 15.2 m (50 feet). For certification of the design, appropriate basic criteria could be based on limits associated with nuclear safety such as:

- Maintenance of acceptable radiation shielding to keep exterior surface dose rates within acceptable limits.

- Maintenance of physical protection of the DSC from other events.

- Limiting the maximum temperature reached by cladding to the acceptable limit.

- Limiting stresses and deformations of the DSC shell due to temperature and/or loads to ensure that rupture of that confinement barrier is not risked.

- Limiting stresses and deformation of the interior spaces, support, and positioning elements with the DSC due to temperature and/or accelerations to ensure acceptable spacing and retrieval of IFAs.

The load limits, expressed in load combinations involving other design loads, should provide adequate criteria to satisfy the above. However, the user must not assume that the temperatures, accelerations, missile impacts, and other loads examined for the certification cannot be exceeded by credible fires and explosions regardless of site location and other circumstances. 10 CFR 72.212(b)(2) requires written evaluations to establish that certificate conditions are met with respect to fire and explosion because a potential exists for all sites in the use of internal combustion engine-powered transport trailer.

Accordingly, verification that loadings resulting from potential fires and explosions do not exceed those used in the SAR for other events and conditions, is required for installation of the standardized NUHOMS systems in accordance with 10 CFR 72.212(b)(2).

2.6 Confinement Barriers and Systems

The staff considered 10 CFR 72.122(h) which provides that confinement barriers and systems shall: "(1) protect the spent fuel cladding 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"; (2) must provide ventilation and off-gas systems "where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions"; (3) "must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken"; (4) "must be packaged in a manner that allows handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of 10 CFR Part 20 limits."

The staff has reviewed the features of the DSC design which provide confinement of radioactive material and, specifically, protection of the spent fuel assemblies. The protection of the spent fuel assemblies depends on two conditions: (1) appropriate fuel clad temperature is not exceeded, and (2) the inert helium atmosphere does not leak out. The first condition is met by thermal hydraulic analyses. See SER Section 4.0. The maintenance of the helium atmosphere is discussed below. The review was directed at two aspects of the design: the integrity of the DSC and the allowable leak rate. Confinement is ensured by a combination of inspection techniques, including radiographic inspection, dye penetrant testing, and helium leak testing.

The SAR takes the position that the inert helium atmosphere in the DSC will not leak out and that the fuel cladding temperature will be held below levels at which damage could occur. The staff determined that this position is acceptable as a criterion. The staff accepts that the helium atmosphere will be maintained during storage. This is based on the specified acceptance leak rate for the primary seal weld of $\leq$ 0.01 kPa-cc/sec (10^{-4} atm-cc/sec), as well as on the integrity of the DSC. The confinement integrity is ensured by the use of stainless steel, thus precluding corrosion of the DSC, and also by the design criteria which include accident cases such as a drop.

10 CFR 72.236(e) requires that the cask must be designed to provide redundant sealing of confinement systems. Although not expressed as a design criterion, the standardized NUHOMS system employs redundant sealing as discussed in Section 5.0 of the report.

10 CFR 236(j) requires that the cask must be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness. The quality standards under which the DSC is fabricated and welded provide the assurance of confinement integrity. In addition, the DSC is pressurized and leak tested after all confinement welds have been completed, in accordance with procedures described in SAR Section 10.3.4.

The criteria for continuous monitoring are issues which have also been evaluated by the NRC staff. To date, under the general license, NRC has accepted continuous pressure monitoring of the inert helium atmosphere as an indicator of acceptable performance of mechanical closure seals for dry spent fuel storage casks. The NRC does not consider such continuous monitoring for the standardized NUHOMS system double weld seals for the DSC to be necessary because: (1) there are no known long-term degradation mechanisms which would cause the seals to fail within the design life of the DSC; (2) the possibility of corrosion has been provided for in the design because the canister is stainless steel; (3) the creep mechanism is not plausible, because the internal storage pressure is approximately atmospheric, (4) cyclic loading has been considered, and it is below the threshold which the ASME B&PV Code Section III has established; (5) the internal atmosphere in the DSC cavity is inert helium gas.

Therefore, an individual continuous monitoring device for each DSC is not necessary. However, the NRC considers that other forms of monitoring storage confinement systems including periodic surveillance, inspection and survey requirements, and application of preexisting radiological environmental monitoring programs of 10 CFR Part 50 licensees during the period of use of the DSC canisters with seal weld closures can adequately satisfy the criteria in 10 CFR 72.122(h)(4).

2.7 Instrumentation and Control Systems

The staff considered 10 CFR 72.126 which provides the provision of (1) protection systems for radiation exposure control; (2) radiological alarm systems; (3) systems for monitoring effluents and direct radiation; and (4) systems to control the release of radioactive materials in effluents. The SAR takes the position (Paragraph 3.3.3.2) that, because of the passive nature of the standardized NUHOMS system, no safety related instrumentation is necessary. Since the DSC was conservatively designed to perform its confinement function during all worst-case conditions, as has been shown by analysis, there is no need to monitor the internal cavity of the DSC for temperature or pressure during normal operations.

The staff also considered 10 CFR 72.122(i) which provides that an ISFSI should have the capability to test and monitor components important to safety. The user of the standardized NUHOMS system will, as provided in Chapter 12, be required to verify by a temperature measurement, the system thermal performance on a daily basis to identify conditions which threaten to approach design temperature criteria. The user will also be required to conduct a daily visual surveillance of the air inlets and outlets as provided in Chapter 12. Therefore, the criteria in of 10 CFR 72.122(i) are satisfied.

While the DSC and HSM are considered components important to safety that comprise the standardized NUHOMS system design, they are not considered operating systems in the same sense as spent fuel pool cooling water systems or ventilation systems which may require other instrumentation and control systems to ensure proper functioning. Hence, due to this passive design, temperature monitoring and surveillance activities are appropriate and sufficient for this design. They ensure adequate protection of the public health and safety and meet the criteria in 10 CFR 72.122(i). Given the passive nature and inherent safety, there is no technical reason to require other instrumentation and control systems for monitoring the standardized NUHOMS system during storage operations.

Non-safety related instrumentation that would be used within a fuel pool facility during loading, unloading, and decontamination is considered by the SAR (Paragraph 3.3.3.1) as being covered by the user's 10 CFR Part 50 license. Instruments used in fuel pool facilities that would be used with DSC loading and unloading operations, and for other operations, include instruments measuring the boron content of the spent fuel pool water and the surface contamination and/or dose rates of the DSC and TC.

Additional instrumentation that may be used in fuel pool facilities that may not already be used in current operations would provide: helium leak detection of the DSC welds, helium pressure in the DSC, and vacuum measurement of the DSC. These instruments may also be used for weld inspection. Formal NRC evaluation of instrument use within fuel pool facilities is in conjunction with 10 CFR Part 50 review of an updated FSAR and associated documentation.

Instrumentation used outside of the fuel pool facility and specifically associated with the ISFSI operations would be as follows:

- Prime mover instruments. The principal concern is that prime mover instruments be operational, support any velocity restrictions and reduce the probability of vehicle malfunction or fire.

- Measurement of the HSM surface dose rates.

- Measurement of air temperature rise through the HSM following loading.

- Measurement of hydraulic pressure for the ram with pressure gauges.

- Alignment of cask and ram with HSM using optical survey equipment.

Use of the instruments is summarily described in Sections 5 and 10 of the SAR (by statements and by inclusions SAR paragraph 10.3.5.2, 10.3.5.6, and 10.3.4.1 by reference).

The staff considers the descriptions of instrumentation usage and commitment to preparation of operating procedures, which should include use of the instruments, to be satisfactory for certification.

2.8 Nuclear Criticality Safety

To address nuclear criticality safety, the staff considered the provisions of 10 CFR 72.124 and 10 CFR 72.236(c). 10 CFR 72.124 provides that the system should be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of the system must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions. The design must also be based on favorable geometry, permanently fixed neutron absorbing materials, or both. 10 CFR 72.236(c) requires that the cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

The proposed criticality safety criteria for the standardized NUHOMS system are discussed in Section 3.0 of the SAR. The standardized NUHOMS system is designed to maintain nuclear criticality safety under normal handling and storage conditions, off-normal handling, and hypothetical accident conditions. According to the SAR, the principal criticality design criteria is that k_{eff}, which includes error contingencies and calculational and modeling biases, remain below 0.95 during both normal operation and accident conditions. The design basis accident is defined as the inadvertent misloading of the DSC with unirradiated fuel of the maximum allowable enrichment.

The NRC staff considers that the proposed criteria satisfy 10 CFR 72.124 and 72.236(c) with the following conditions/observations:

For the Standardized NUHOMS-24P Design

1. The vendor is required to include an additional constraint of limiting the initial enrichment equivalent of stored PWR fuel assemblies to 1.45 wt.% U-235 so that the optimal moderated array reactivity is less than 0.95 (including bias and uncertainties). The initial enrichment equivalent of an irradiated fuel assembly is the U-235 enrichment of unirradiated fuel assemblies which would give the same reactivity as the irradiated fuel array. Although not included as a criterion, this constraint is included in the proposed specification for the fuel to be stored. (See Table 12-1a.)

For the Standardized NUHOMS-52B Design

2. The vendor is required to include a constraint of limiting the initial fuel enrichment of stored BWR fuel assemblies to 4.0 wt.% U-235.

3. The vendor is required to ensure a minimum fixed absorber plate boron content of 0.75 wt.% boron in the fabrication of the DSC. (See Table 12-1b.)

The staff's evaluation of the nuclear criticality safety for the standardized NUHOMS system is included in Section 7.0 of this report.

2.9 Radiological Protection

With respect to on-site protection, Section 20.1201(a) of 10 CFR Part 20 states that the licensee shall control the occupational dose to individual adults to the dose limits specified in 1201(a)(1) and 1201(a)(2). Also, section 20.1101 of 10 CFR Part 20 states that each licensee shall develop, document, and implement a radiation protection program and that the licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

Section 72.126 provides for the provision of: (1) protection systems for radiation exposure control; (2) radiological alarm systems; (3) systems for monitoring effluents and direct radiation; and (4) systems to control the release of radioactive materials in effluents.

Guidance for ALARA considerations is also provided in NRC Regulatory Guides 8.8 and 8.10 (References 10 and 11).

For off-site radiological protection, the staff considered the requirements contained in 10 CFR 72.104(a) for normal operations and anticipated occurrences, and 10 CFR 72.106(b) for design basis accidents. In addition, the staff considered the dose limitations in 10 CFR Part 20 including the requirement that doses to members of the public must be as low as is reasonably achievable.

The radiological protection design features of the standardized NUHOMS system are described in Chapters 3 and 7 of the SAR and are evaluated in Section 8.0 of this SER. These features consist of: (1) radiation shielding provided by the transfer cask, DSC, and HSM; (2) radioactive material containment within the DSC; (3) prevention of external surface contamination; and (4) site access control. Access to the site of the standardized NUHOMS system array, which is a site-specific issue not specifically addressed in the SAR, would be restricted to comply with 10 CFR 72.106 controlled area requirements.

Based on analyses presented in the SAR (discussed in Section 8.0 of this SER), the staff concludes that the standardized NUHOMS system, if properly sited, meets the design criteria for on-site and off-site radiological protection, including the incorporation of ALARA principles.

2.10 Spent Fuel and Radioactive Waste Storage and Handling

The staff considered 10 CFR 72.128(a), which provides that the spent fuel and radioactive waste storage systems should be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with (1) a capability to test and monitor components important to safety; (2) suitable shielding for radiation protection under normal and accident conditions; (3) confinement structures and systems; (4) a heat-removal capability having testability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive waste generated. Section 72.128(b) further states that radioactive waste treatment facilities should be provided for the packing of site-generated low-level wastes in a form suitable for storage on-site awaiting transfer to disposal sites.

Criteria covering items (1) through (4) above have been addressed throughout the preceding sections in this SER in the preceding sections of this Chapter. The SAR does not specifically address the issue of minimization of radioactive waste generation. Solid wastes will likely be limited to small amounts of sampling or decontamination materials such as rags or swabs, while liquid wastes will consist mainly of small amounts of liquid resulting from decontamination activities. Contaminated water from the spent fuel pool and potentially contaminated air and helium from the DSC, which are generated during cask loading operations, will be treated using plant-specific systems and procedures. No radioactive wastes requiring treatment are generated during the storage period during either normal operating or accident conditions.

The staff agrees that the design of the standardized NUHOMS system provides for minimal generation of radioactive wastes, and that any wastes that are generated would be easily accommodated by existing plant-specific treatment or storage facilities.

2.11 Decommissioning/Decontamination

Under 10 CFR 72.236(i), considerations for decommissioning and decontamination must be included in the design of an ISFSI. In this regard the staff has considered 10 CFR 72.130 that provisions should be incorporated to: (1) decontaminate structures and equipment; (2) minimize the quantity of waste and contaminated equipment; and (3) facilitate removal of radioactive waste and contaminated materials at the time of decommissioning.

10 CFR 72.30 defines the need for a decommissioning plan which includes financing. Such a plan, however, is not considered applicable to this review. The cost of decommissioning the ISFSI must be considered in the overall cost of decommissioning the reactor site.

To facilitate decommissioning of the HSM, the design should be such that:

(1) There is no credible chain of events which would result in widespread contamination outside of the DSC; and

(2) Contamination of the external surfaces of the DSC must be maintained below applicable surface contamination limits. The SAR uses the following smearable (non-fixed) surface removable contamination limits as a limiting condition for operation:

Beta-gamma emitters:	36.5 Bq/100 cm^2 (2200 dpm/100 cm^2)
Alpha emitters:	3.65 Bq/100 cm^2 (220 dpm/100 cm^2)

Decommissioning considerations are described in Sections 3.5 and 9.6 of the SAR and are evaluated in Section 9.0 of this report.

The staff acknowledges that decommissioning considerations are sometimes in conflict with other requirements. The reinforced structure of the HSM, for example, will require considerable effort to demolish. Although it is not likely that significant contamination can spread beyond the DSC, demolition of the HSM may generate slightly contaminated dust. However, the staff concurs that primary concern in such cases rests with operational safety considerations, and ease of decommissioning is a secondary consideration. In this regard, the staff concludes that adequate attention has been paid to decommissioning in the design of the standardized NUHOMS system.

2.12 Criteria for Fuel Stability

The staff considered the general design criteria set forth in Section 72.122(h) on "Confinement Barriers and Systems." Paragraph (1) of this section provides that "spent fuel cladding must be protected during storage against degradation that leads to gross rupture" and "that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage." This aspect of the standardized NUHOMS system design is discussed in Section 2.6 of this SER. Paragraphs (2) and (3) in Section 72.122(h) relate to underwater storage of fuel and to ventilation and off-gas systems, respectively, and are therefore not considered in this review. Paragraphs (4) and (5) deal with monitoring and handling and retrievability operations, respectively, and are addressed in Sections 2.6, 2.7 and 2.10 of this document.

2.13 Findings and Conclusions

Tables 2.2, 2.3, and 2.4 summarize the principal design criteria for the standardized NUHOMS system components important to safety. Criteria identified in the SAR for design of the standardized NUHOMS system are acceptable with the exceptions noted below. These findings and conclusions apply to criteria and not the actual design (see "Conclusions/Discussion" paragraph at the end of each Section).

Exceptions related to the HSM and its integral DSC Support Assembly and their resolution are summarized below:

- There are criteria used which may not be acceptable for all potential sites in the continental United States for: earthquake maximum ground accelerations, lightning, flood, and maximum ambient temperature. Uses of the standardized NUHOMS system at individual sites requires verification that the appropriate site parameters and that these parameters are within the acceptable design criteria.

- Fire and explosion loads are assumed to be within maximums for other included loadings. Use of the standardized NUHOMS system at a site requires examinations of potential causes and magnitudes of fires and explosions, and verification that site parameters are bounded by appropriate design criteria evaluated in this SER.

- The loads associated with a jammed DSC are acceptable; however, evaluation of those loads in a load combination expression unintended for "accidents" is not acceptable. Although the usage of the criteria is not acceptable, the staff has determined that the actual design, evaluated with the acceptable load combination expression, is acceptable.

The SAR includes a nominal design for the HSM foundation. The foundation only has nuclear safety implications in the event of gross failure, since it is structurally independent of (although loaded by) the supported HSM. Suitability of the HSM foundation design of the SAR should be verified for the actual site by a foundation analysis, or an alternative foundation design should be used for the site. The user must perform written evaluations before use to establish that cask storage pads have been designed in accordance with 10 CFR 72.212(b)(2) and (b)(3) to ensure that no gross failures occurred that could cause the standarized NUHOMS system to be in an unanalyzed situation.

TABLE 2.1 Design Criteria Sources Cited in the SAR
[*DS-Docketed submittals which modify and/or extend the SAR presentation]

SAR Reference	Source	Use	NRC Comments
3.2.5.1	ANSI/ANS 57.9-1984	Load combinations for HSM Design	Acceptable
3.2.5.2	ASME B&PV Code (1983) Section III, Div. 1, Subsection NB and NF for Class 1 Components and Supports	Subsection NB used for stress analysis and allowable stresses for DSC shell and lids. Subsection NF used for stress analysis and allowable stresses for DSC basket.	Acceptable
3.2.5.2	ASME B&PV Code (1983) Section III, Div. 1, Subsection NC for Class 2 Components	TC stress analysis and allowable stresses excluding the lifting/tilting trunnions.	Acceptable
3.2.5.3	ANSI N14.6-1986	Allowable stresses for lifting trunnions inside fuel building.	Acceptable
Table 3.2-1	ACI-318-83	Construction criteria for concrete HSM.	Acceptable
Table 3.2-1	AISC Code for Structural Steel	DSC Support Assembly Design.	Acceptable for design stresses, but not load combinations.
Table 3.2-1	ASME B&PV Code (1983) Section III, Subsection NC	Allowable stresses for lifting and support trunnions on-site transfer for TC.	Acceptable
3.3.4.1.1.A.	ORNL/NUREG/CSD-2	"SCALE-3" Code used for Criticality Analysis	Acceptable
3.3.4.1.2.A.	STUDSVIK/NR-81/3	"CASMO-2" Code for Fuel Burnup	Acceptable
3.3.4.1.2.A.	DPC-NE-1002A	Duke Power Co. Reload Methodology	Acceptable
3.3.4.1.2.A.	RF-78/6293	STUDSVIK CASMO Benchmark	Acceptable
3.3.4.1.2.A.	STUDSVIK/NR-81/61	CASMO Benchmark	Acceptable
3.6	NUREG/CR-2397	Fuel Assembly Thermal Parameters	Acceptable
3.6	ORNL/TM-7431	Fuel Assembly Thermal Parameters	Acceptable
3.6	ANSI/ANS-5.1-1979	Fuel Assembly Thermal Parameters	Acceptable

Table 2.1 Design Criteria Sources Cited in the SAR (Continued)
[*DS-Docketed submittals which modify and/or extend the SAR presentation]

SAR Reference	Source	Use	NRC Comments
3.6	A.D. Little, Inc., "Tech. Supt for Rad Stds. Hi-Lvl Rad Waste Mgt"	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.3, Tbl 3.1-4a Tbl 3.1-4b	NUREG\CR-2397 NUREG\CR-0200 DOE\RW-0184	Development of radiological characteristics using ORIGEN	Acceptable
3.1.2.1	Reg. Guide 1.60	Seismic Design Response Spectra	Acceptable
3.1.2.1	Reg. Guide 1.61	Seismic Design Damping Values	Acceptable
3.1.2.1	ANSI/ANS-57.9-1984	Operational Handling Loads	Acceptable
3.1.2.1	ANSI/ANS-57.9-1984	Accidental Drop Loads	Acceptable
3.1.2.1	ANSI/ANS-57.9-1984	Thermal and Dead Loads	Acceptable
3.1.2.1	Reg. Guide 1.76	Tornado Wind Loads	Acceptable
3.1.2.1	NUREG-0800	Impact Force Criteria, Tornado Missiles, Recommended Empirical Formula Use	Acceptable
3.2	ANSI/ANS-57.9-1984	Extreme Environmental and Natural Phenomena	Acceptable
3.2.1.2	ANSI A58.1-1982	Tornado Wind MPH to Pressure Conversion	Acceptable
3.2.1.2	Bechtel BC-TOP-9-A	Method for Determining Impact Force for Design of Local Reinforcing	Not an accepted source, but the results of PNFS calculations are acceptable.
3.2.3	10 CFR 72 10 CFR 100, Appendix A Reg. Guide 1.60 Reg. Guide 1.61	Seismic Criteria and Basis for Criteria	Acceptable
3.2.4	ANSI A58.1-1982	Snow and Ice Loads	Acceptable

Table 2.1 Design Criteria Sources Cited in the SAR (Continued)
[*DS-Docketed submittals which modify and/or extend the SAR presentation]

SAR Reference	Source	Use	NRC Comments
3.2.5.1	ACI-349-1985	Reinforced Concrete Design	Acceptable
3.3.4.1.2A	SAND 86-0151	Major neutron absorbers	Reference has not been used in NRC review. The NRC accepts credit for boron in pool water. The use of burnup credit for storage casks is not approved.
3.3.4.1.3A	ANSI/ANS-57.2-1983	Criticality Criteria	Acceptable
3.3.4.1.3A	ANSI/ANS-8.17-1984	Credit for Burnup	Reference not used in NRC review. NRC accepts credit for boron in pool water. Use of burnup credit for storage casks is not approved.
3.3.4.1.3A	PNL-2438	Sources of Negative Reactivity	Reference not used in NRC review. NRC accepts credit for boron in pool water. Use of burnup credit for storage casks is not approved.
3.3.4.1.3C	EPRI - NP-196	Critical Experiment Benchmarks	Reference not used in NRC review.
3.3.4.1.4A	ANSI/ANS 8.17-1984	Double Contingency Principle	Reference not used in NRC review.
3.3.4.2.2	ORNL CCC-548	"KENO5A-PC" Monte Carlo Code	Acceptable
3.3.4.2.3A	NUREG/CR-1784	Criticality Experiments	Reference not used in NRC review.
3.3.4.2.3C	NUREG/CR-0073	Criticality Separation	Reference not used in NRC review.

Table 2.1 Design Criteria Sources Cited in the SAR (Continued)
[*DS-Docketed submittals which modify and/or extend the SAR presentation]

SAR Reference	Source	Use	NRC Comments
3.3.4.2.3C	NUREG/CR-0796	Criticality Experiments	Reference not used in NRC review.
3.3.4.2.3C	BAW-1484-7	Criticality Experiments B&W	Acceptable
3.3.4.2.3C	PNL-6838	Reactivity Measurements	Acceptable
3.3.4.2.3D	ORNL/TM-10902	Physical Characteristics of GE Fuel	Reference not used in NRC review.
3.3.7.1.1	PNL-6189	Fuel Cladding Temperature Limits	Acceptable
3.3.7.1.1	PNL-4835	Fuel Cladding Temperature Limits	Acceptable
3.4.4.1	NUREG/0612	Lifting Devices Criteria	Acceptable
Not cited	NUREG/CR-1815	Brittle Fracture Criteria for Ferritic Steel	Acceptable, used in NRC review.

Table 2.2 Evaluation of Design C[illegible]. for Normal Operating Conditions
[Columns (1) - (5) are extracted from SAR Tables 3.2-1 and 3.2-5]

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
HSM	Dead Load	SAR 8.1.1.5	Dead weight including loaded DSC	ANSI 57.9-1984 ACI 349-85 and ACI 349R-85	Acceptable, site immaterial. Although DSC is actually a "live load," DSC weight is precisely known.
	Load Combination	SAR Table 3.2-5	Strength requirements for specific load combination	ANSI 57.9-1984	Acceptable.
	Design Basis Normal Temperature	SAR 8.1.1.5 SAR 8.1.1.1	DSC with spent fuel rejecting 24.0 kW decay heat for 5 yr cooling time. Ambient air temperature range -40° to +125°F. Average yearly ambient temperature = 70°F	ANSI 57.9-1984	Acceptable for most of Continental US for this system. This temperature bounds most reactor sites.
	Normal Handling Loads	SAR 8.1.1.1	Hydraulic ram load: 20,000 lb.	ANSI 57.9-1984	Acceptable, site immaterial.
	Snow and Ice Loads	SAR 3.2.4	Maximum load: 110 psf	ANSI 57.9-1984	Acceptable for all of continental US.
	Live Loads	SAR 8.1.1.5	Design load: 200 psf	ANSI 57.9-1984	Acceptable for all of continental US for this system.
	Shielding	SAR 7.1.2	Average contact dose rate on HSM exterior surface <400 mrem/hr at 3 feet from HSM surface.	ANSI 57.9-1984	Acceptable.
HSM Foundation	Static Loads	SAR 3.4.3	To be designed for individual site based on site foundation analysis for static loads	10 CFR 72.212(b)(2)(ii)	Acceptable for Certificate.
Dry Shielded Canister	Dead Loads	SAR 8.1.1.2	Weight of loaded DSC: 65,000 lb. nominal, 80,000 lb. enveloping	ANSI 57.9-1984	Acceptable.
	Design Basis Internal Pressure Load	SAR 8.1.1.2	DSC internal pressure 9.6 psig	ANSI 57.9-1984	Acceptable.
	Structural Design	SAR Table 3.2-6	Service Level A and B Stress Allowables	ASME B&PV Code Sec. III, Div. 1, NB, Class I	Acceptable.

Table 2.2 Evaluation of Design Criteria for Normal Operating Conditions (Continued)

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
Dry Shielded Canister (Cont'd)	Design Basis Operating Temperature Loads	SAR 8.1.1.2 SAR 10.3.15 SAR Table 8.1-2	DSC decay heat 24.0 kW for 5yr cooling time. Ambient air temperature -40°F to 125°F. Lifting inside the spent fuel pool building of loaded DSC restricted to -20°F ambient air temperature if lift height is 80 inches or less. If lift height is above 80 inches, minimum temperature is restricted to 0°F. Outside spent fuel pool building, maximum lift height is 80 inches and minimum temperature is 0°F.	ANSI 57.9-1984 ASME B&PV Code Sect. III, Div. 1, NF-2300 NUREG/CR-1815	Acceptable. Acceptable with restrictions on use. Acceptable for impact testing.
	Operational Handling	SAR 8.1.1.2	Hydraulic ram load: 20,000 lb enveloping	ANSI 57.9-1984	Acceptable.
	Criticality	SAR 3.3.4	K_{eff} less than 0.95	ANSI 57.2-1983	Acceptable.
DSC Support Assembly	Load Combinations	SAR Table 3.2-5C	Allowable factored stresses for specific load combinations.	ANSI 57.9-1984	Acceptable, site immaterial.
	Dead Loads	SAR 8.1.1.4	Loaded DSC + self weight	ANSI 57.9-1984	Acceptable, site immaterial.
	Normal Handling	SAR 8.1.1.4	DSC Reaction load with hydraulic ram load: 20,000 lbs.	ANSI 57.9-1984	Acceptable, site immaterial.
	Normal Temperature	SAR Table 3.2-5c	Factored allowable stresses for specific load combinations.	ANSI 57.9-1984	Design acceptable.
	Stress Evaluation	SAR Table 3.2-7	Stress allowables.	AISC Steel Construction Manual	Acceptable.
Transfer Cask	Normal Operating Condition	SAR Table 3.2-8	Service Level A and B Stress allowables	ASME B&PV Code Sec. III, Div. 1 NC-3200	Acceptable.
Structure: Shell, Rings, etc.	Dead Loads	SAR 8.1.1.9	a) Vertical orientation, self weight + loaded DSC + water in cavity: 200,000 lb. enveloping. b) Horizontal orientation, self weight + loaded DSC on transfer skid: 200,000 lb. enveloping.	ANSI 57.9-1984	Acceptable.
	Snow and Ice Loads	SAR 3.2.4	External surface temperature of cask will preclude buildup of snow and ice loads when in use: 0 psf	10 CFR 72.122(b)	Acceptable.

Table 2.2 Evaluation of Design Criteria for Normal Operating Conditions (Continued)

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
Transfer Cask (Cont'd)	Design Basis Operating Temperature Loads Outside Spent Fuel Pool Building	SAR 8.1.1.8 8.1.2.2 SAR 10.3.15	Loaded DSC rejecting 24.0 kW decay heat with 5yr cooling time. Ambient air temperature range -40°F to 125°F with solar shield, -40°F to 100°F w/o solar shield.	ANSI 57.9-1984 ANSI N14.6	Acceptable. Note, the minimum handling temperature of the loaded DSC inside the TC is 0°F (upper trunnions are ferritic).
	Design Basis Operating Temperature Loads Inside Spent Fuel Pool Building	SAR Table 8.1-2 SAR 10.3.15	The minimum handling temperature of a loaded DSC inside a TC is -20°F, for height of 80 inches or less. For lift heights greater than 80 inches the minimum handling temperature is 0°F. Impact testing at -40°F required per SAR.	ASME B&PV Code Section III, Div. 1, NC-2300 ANSI N14.6 ¶ 4.2.6	Acceptable. The ASME B&PV Code is acceptable with the restrictions stated in design parameters (upper trunnions are ferritic). Acceptable for impact testing.
	Shielding	SAR 7.1.2	Average contact dose rate less than 100 mrem/hr.	ANSI 57.9-1984	Acceptable.
TC Upper Trunnions	Operational Handling	SAR 8.1.1.9 SAR App. C	a) Upper lifting trunnions while in Auxiliary Building: i) Stress must be less than yield stress for 6 times critical load/trunnion nominal ii) Stress must be less than ultimate stress for 10 times critical load b) Upper lifting trunnions for onsite transfer: 118,000 lb./trunnion 94,000 lb./shear 29,500 lb./trunnion axial	ANSI N14.6-1978 ASME B&PV Code Sec. III, NC Class 2	Acceptable. Acceptable.
TC Lower Trunnions	Operational Handling	SAR 8.1.1.9	Lower support trunnions weight of loaded cask during downloading and transit to HSM	ASME B&PV Code Sec. III, NC, Class 2	Acceptable.
TC Shell	Operational Handling	SAR 8.1.1.9	Hydraulic ram load due to friction of extracting loaded DSC: 20,000 lb. enveloping	ANSI 57.9-1984	Acceptable.
TC Bolts	Normal Operating	SAR Table 3.2-9	Service Levels A, B, and C Avg. stress less than 2 S_m Max. stress less than 3 S_m	ASME B&PV Code Section III, NC, Class 2, NC-3200 XIII-1180	Acceptable.

Table 2.3 Evaluation of Design Criteria for Off-Normal Operating Conditions [Columns (1) - (5) are extracted from SAR Tables 3.2-1 and 3.2-5]

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
HSM	Off-Normal Temperature	SAR 8.1.1.5 SAR 8.1.1.1	-40°F to +125°F ambient temperature 70°F average yearly ambient	ANSI 57.9-1984	Acceptable for most of Continental US for this system.
	Off-Normal (Jammed Condition) Handling	SAR 8.1.1.4	Hydraulic ram load of 80,000 lb.	ANSI 57.9-1984	Acceptable, site immaterial.
	Load Combination	SAR Table 3.2-5	Strength requirements for specific Load Combinations	ANSI 57.9-1984	Acceptable, site immaterial.
Dry Shielded Canister	Off-Normal Temperature	SAR 8.1.1.2 SAR 8.1.2.2 SAR 10.3.15 SAR Table 8.1-2	-40°F to 125°F ambient temperature Lifting of loaded DSC restricted to -20°F or more ambient air temperature for lifts of less than 80 inches. Outside spent fuel pool building maximum lift height is 80 inches, and minimum temperature is 0°F.	ANSI 57.9-1984 ASME B&PV Code, Sec. III, NF-2300 NUREG/CR-1815	Acceptable for storage. Acceptable for transport. Acceptable for impact testing.
	Off-Normal Pressure	SAR 8.1.1.1 SAR 8.1.1.2	DSC internal pressure less than 9.6 psig	ANSI 57.9-1984	Acceptable.
	Jammed Condition Handling	SAR 8.1.2.1	Hydraulic ram load equal to 80,000 lb. nominal	ANSI 57.9-1984	Acceptable.
	Structural Design Off-Normal Conditions	SAR Table 3.2-6	Service Level C Stress Allowables	ASME B&PV Code Sec. III, Div. 1, NB Class 1	Acceptable.
DSC Support Assembly	Jammed Handling Condition	SAR 8.1.1.4	Hydraulic ram load: 80,000 lb. nominal	ANSI 57.9-1984	Acceptable, site immaterial. However, used in load combination as though an "accident" load. Not acceptable application of criteria.
	Off-Normal Temperatures	SAR Table 3.2-5c	Factored allowable stresses for specific load combinations.	ANSI 57.9-1984	Actual design acceptable.
	Load Combination	SAR Table 3.2-5C	Factored allowable stresses for specific load combination	ANSI 57.9-1984	Acceptable, site immaterial.

Table 2.3 Evaluation of Design Criteria for Off-Normal Operating Conditions (Continued)

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
Transfer Cask	Off-Normal Temperature	SAR 8.1.1.1	100°F ambient temperature w/o solar shield, 125°F with solar shield	ANSI 57.9-1984	Acceptable.
	Brittle fracture of ferritic steel trunnions	SAR 8.1-2	Lower temperature limit is -40°F for use inside spent fuel pool building for any lift without loaded DSC. Lower temperature limit for loaded DSC is -20°F for lifts less than 80 inches.	ASME B&PV Code, Sec. III, NC-2300 ASME B&PV Code, Sec. III, NF-2300 ANSI N14.6 ¶ 4.2.6	Acceptable. Acceptable for impact testing.
	Jammed Condition Handling	SAR 8.1.2.1	Hydraulic ram load: 80,000 lb. nominal	ANSI 57.9-1984	Acceptable.
	Structural Design Off-Normal Conditions	SAR Table 3.2-8	Service Level C Stress Allowables	ASME B&PV Code Sec. III, Div. 1, NC, Class 2	Acceptable.
	Bolts, Off-Normal Conditions	SAR Table 3.2-9	Service Level C Avg. stress less than 2 S_m Max. stress less than 3 S_m	ASME B&PV Code Sec. III, Div. 1, NC, Class 2 NC-3200	Acceptable.

Table 2.4 Evaluation of Design Criteria for Accident Conditions
[Columns (1) - (5) are from SAR Table 3.6-3 and Paragraph 8.2.6]

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
HSM	Design Basis Tornado	SAR 3.2.1	Max. velocity 360 mph Max. wind pressure 397 psf	Regulatory Guide 1.76 ANSI A58.1-1982	Acceptable for US.
	Load Combination	SAR Table 3.2-5	Strength requirements for specific load combinations	ANSI 57.9-1984	Acceptable, site immaterial.
	Design Basis Tornado Missiles	SAR 3.2.1	Max. velocity 126 mph Types: Automobile 3,967 lb. 8 in. diam shell, 276 lb. 1 in. solid sphere	NUREG-0800 Sec. 3.5.1.4	Acceptable for US. Does not include all NUREG-0800 missiles but those used are the most critical for the HSM.
	Flood	SAR 3.2.2	Maximum water height: 50 feet Maximum velocity: 15 fps	10 CFR 72.122(b)	Acceptable for Certification. Verification that design criteria bound site parameters.
	Seismic	SAR 3.2.3	Horizontal ground acceleration 0.25g (both directions) Vertical ground acceleration 0.17g	NRC Regulatory Guides 1.60 and 1.61	Acceptable for Certification. Verification that design criteria bound site parameters.
	Accident Condition Temperatures	SAR 8.2.7.2	DSC with spent fuel rejecting 24.0 kW of decay heat for 5yr cooling time. Ambient air temperature range of -40°F to +125°F with HSM vents blocked for 5 days or less.	ANSI 57.9-1984	Low temperature acceptable for continental US. Verification of maximum temperature for individual sites. Blockage criteria acceptable with appropriate daily surveillance.
	Fire and Explosions	SAR 3.3.6	"Enveloped by other design events," e.g. - design basis tornado - design basis tornado and missiles	10 CFR 72.122(c)	Analysis of potential fires and explosions from any credible sources required for each site. Verification that assertion (Column (4)) bound site parameters.

Table 2.4 Evaluation of Design Criteria for Accident Conditions (Continued)

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
HSM (Continued)	Lightning	SAR 8.2.6	"Lightning protection system requirements are site specific..."		Acceptable for Certificate. NFPA 78, Lightning Protection Code is to be used for evaluation of need and design of lightning protection at site.
HSM Foundation	Static load	SAR 3.4.3	To be designed for individual site based on site foundation analysis for static loads.	10 CFR 72.212(b)(2)(ii)	Acceptable for Certificate. Nominal SAR design or alternative design should be verified for individual site.
DSC	Accident Drop	SAR 8.2.5	Equivalent static deceleration: 75g vertical end drop 75g horizontal side drop 25g corner drop with slap down (corresponds to an 80 inch drop height) Structural damping during drop: 10%	10 CFR 72.122(b) Reg. Guide 1.61	Acceptable. Administrative controls must be imposed to prevent lifting or transporting the loaded DSC outside the spent fuel pool building higher than 80 inches. 10% damping value exceeds R.G. 1.61 guidance. A 7% value has been evaluated by the staff and has been accepted.
	Flood	SAR 3.2.2	Maximum water height: 50 feet	10 CFR 72.122(b)	Acceptable for Certification. Verification required for individual sites.
	Seismic	SAR 3.2.2 SAR 8.2.3.2	Horizontal ground acceleration 0.25g Vertical ground acceleration 0.17g Horizontal acceleration: 1.5g Vertical acceleration: 1.0g 3% critical damping	 NRC Regulatory Guides 1.60 and 1.61	Acceptable.
	Accident Internal Pressure (HSM vents blocked for 5 days)	SAR 8.2.7.2 Table 8.1-4a	DSC internal pressure: 50.3 psig based on 100% fuel clad rupture and fill gas release, and ambient air temp. = 125°F DSC shell temperature: 587°F	10 CFR 72.122(b)	Acceptable.
	Accident Conditions	SAR Table 3.2-6	Service Level D Stress allowables	ASME B&PV Code Sec. III, Div. 1 NB, Class I	Acceptable.
	Fire and Explosions	SAR 3.3.6	"bound by" other events, e.g. - design basis tornado - design basis tornado with missiles - postulated drop - external pressure due to 50 feet head of water	10 CFR 72.122(c)	Verification that assertion (column (4)) bound site parameters for both fire and explosions.

Table 2.4 Evaluation of Design Criteria for Accident Conditions (Continued)

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
DSC Support Assembly	Seismic	SAR 3.2.3	DSC reaction loads with Horizontal ground acceleration: 0.25g Vertical ground acceleration: 0.17g	NRC Reg. Guides 1.60 and 1.61	Acceptable for Certification. Verification of criteria.
	Jammed Condition (Treated as a drop of heavy load accident in load combination)	SAR Table 8.2-12	Hydraulic ram load of 80,000 lbs.	ANSI 57.9-1984	Off-normal loads should be treated as live loads (or other non-accident category) in "normal" load combinations. Design is acceptable.
	Load Combination	SAR Table 3.2-5c	Factored allowable stresses for specific load combinations.	ANSI 57.9-1984	Acceptable, site immaterial.
Transfer Cask	Design Basis Tornado	SAR 3.2.1	Max. wind velocity: 360 mph Max. wind pressure: 397 psf	NRC Reg. Guide 1.76, ANSI 58.1-1982	Acceptable.
	Design Basis Tornado Missiles	SAR 3.2.1	Automobile, 3967 lb. 8 in. diameter shell, 276 lb.	NUREG-0800 Sec. 3.5.1.4	Acceptable. Missiles selected bound effects of other missiles in NUREG.
	Flood	SAR 3.2.2	Flood not included in design basis. Cask use to be restricted by administrative controls.	10 CFR 72.122	Acceptable for Certification. Appropriate administrative controls must be in place at any site where flooding is a possibility.
	Seismic	SAR 3.2.3	Horizontal ground acceleration 0.25g (both directions) Vertical ground acceleration 0.17g, 3% critical damping	NRC Reg. Guides 1.60 and 1.61	Acceptable for Certification. Verification of criteria required for individual sites.
	Accident Drop	SAR 8.2.5	Equivalent static deceleration: 75g vertical end drop 75g horizontal side drop 25g corner drop with slapdown (corresponds to an 80 inch drop height) Structural damping during drop 10%	10 CFR 72.122(b)	Acceptable. Administrative controls must be imposed to prevent lifting or transporting TC with loaded DSC outside of the spent fuel building higher than 80 inches. 10% damping exceeds R.G. 1.61 guidance, however, 7% has been evaluated by staff and accepted.
	Bolts, Accident Drop	SAR Table 3.2-9	Service Level D Stress allowables	ASME B&PV Code Sec. III, Div. 1 NC, Class 2, NC-3200	Acceptable.

Table 2.4 Evaluation of Design Criteria for Accident Conditions (Continued)

Component (1)	Design Load Type (2)	Reference (3)	Design Parameters (4)	Applicable Codes (5)	NRC Staff Comments (6)
Transfer Cask (Continued)	Structural Design, Accident	SAR Table 3.2-8	Service Level D	ASME B&PV Code Sec. III, Div. 1 NC, Class 2 NC-3200	Acceptable.
	Internal Pressure	SAR Table 3.2-1	Not applicable because DSC provides pressure boundary.	10 CFR 72.122(b)	Acceptable.
	Lightning	Not Addressed	[NRC Staff: Should not permit damage to DSC or affect DSC retrievability]		Acceptable, based on separate staff analysis of hazard while on transit.
	Fire and Explosions	SAR 3.3.6	"Enveloped by other design basis events," e.g. - design basis tornado generated missile loads	10 CFR 72.122(c)	Verification that assertion (column (4)) bound site parameters for both fire and explosions.

Table 2.5 Load Combinations Used for HSM Reinforced Concrete

Load Comb.	Load Combination Description	Correlation to Standards	NRC Staff Comments
1,2	1.4 D + 1.7 L	ANSI 57.9, Paragraph 6.17.3.1(a)	Acceptable.
3,4	0.75 (1.4 D + 1.7L + 1.7 H + 1.7 T + 1.7 W)	ANSI 57.9, Paragraph 6.17.3.1(c)	[Note: Uses W_t for W] Acceptable. Conservative relative to ACI 349, Paragraph 9.2.1(5) which is also acceptable.
5	D + L + H + T + E	ANSI 57.9, Paragraph 6.17.3.1(e)	Acceptable.
6	D + L + H + T + F	ANSI 57.9, Paragraph 6.17.3.1(f)	Acceptable.
7	D + L + H + T_a	ANSI 57.9, Paragraph 6.17.3.1(g)	Acceptable.
WHERE:			
D =	Dead Weight *1.05	ANSI 57.9, Paragraph 6.17.1.1	Acceptable.
L =	Live Load (varied between 0-100% for worst case)	ANSI 57.9, Paragraph 6.17.1.1	Acceptable.
H =	Lateral Soil Pressure Loads (H taken as = 0)	ANSI 57.9, Paragraph 6.17.1.1	Acceptable.
W =	Tornado Wind Loads	NRC Reg. Guide 1.76 and ANSI A58.1	Acceptable.
T =	Normal Condition Thermal Load	ANSI 57.9, Paragraph 6.17.1.1	Acceptable.
T_a =	Off-Normal or Accident Thermal Loads	ANSI 57.9, Paragraph 6.17.1.3	Acceptable.
E =	Earthquake Load	ANSI 57.9, Paragraph 6.17.1.2	Acceptable.
F =	Flood Load	ANSI 57.9, Paragraph 6.17.1.3	Acceptable.
A =	Accident (e.g., drop accident)	None	

Table 2.5 Load Combinations Used for HSM Reinforced Concrete (Continued)

Load Combination Description	Correlation to Standards	NRC Staff Comments
Omitted Load Combinations of ANSI 57.9		
1.4 D + 1.7 L + 1.7 H (L.C. #2)	ANSI 57.9, Paragraph 6.17.3.1(b)	Omission acceptable [with H=O same as L.C. #1].
0.75(1.4 D + 1.7 L + 1.7 H + 1.7 T) (L.C #4)	ANSI 57.9, Paragraph 6.17.3.1(d)	Omission acceptable [with H=O encompassed by L.C. #3]
D + L + H + T + A DSC Support Structure (Structural Steel), See Table 2-6	ANSI 57.9, Paragraph 6.17.3.1(f)	Omission would not be acceptable except that tornado missile loadings are acceptably analyzed, and that potential consequences of accidental drop of HSM access door is not considered a nuclear safety situation.

Table 2.6 Load Combinations Used for DSC Support Assembly

<table>
<tr><th>Load Combination Description</th><th>Correlation to Standards</th><th>NRC Staff Comments</th></tr>
<tr><td>Equation 1 S > DL + HLf</td><td>ANSI 57.9, Paragraph 6.17.3.2.1(a)</td><td>Acceptable.</td></tr>
<tr><td>Equation 2 1.5S > DL + HLf + T</td><td>ANSI 57.9, Paragraph 6.17.3.2.1(d)</td><td>Acceptable.</td></tr>
<tr><td>Equation 3 1.6S > DL + HLf + T + E</td><td>ANSI 57.9, Paragraph 6.17.3.2.1(e)</td><td>Acceptable. [H = O]</td></tr>
<tr><td>Equation 4 1.7S > DL + Ta</td><td>ANSI 57.9, Paragraph 6.17.3.2.1(g)</td><td>Acceptable. [H, L = O]</td></tr>
<tr><td>Equation 5 1.7S > HLj</td><td>ANSI 57.9, Paragraph 6.17.3.2.1(f)</td><td>Acceptable. [L, H, T = O, and dead load of support assembly is negligible]</td></tr>
<tr><td colspan="3"><u>Where:</u>

DL = Dead Load Support Assembly including DSC weight
HLf = Normal Handling (transfer) Loads due to friction
T = Normal Thermal Load
E = Seismic Load
Ta = Accident Thermal Load
HLj = Off-Normal Handling Loads due to a jammed DC including weight of DSC (accident condition)
H = Lateral Earth Pressure = O
W = Wind or tornado missiles = O (Support assembly is shielded by HSM)
L = Live load not applicable when HSM is closed</td></tr>
<tr><td><u>Omitted Load Combinations of ANSI 57.9</u>

S > D + L + H
1.33S > D + L + H + W</td><td>

ANSI 57.9, Paragraph 6.17.3.2.1(b)
ANSI 57.9, Paragraph 6.17.3.2.1(c)</td><td>

Omission acceptable [H = O]
Omission acceptable [H, W = O]</td></tr>
</table>

3.0 STRUCTURAL EVALUATION

Introduction

This section evaluates the structural designs of the HSM, DSC, and TC. The designs are evaluated against design criteria as presented in the SAR, or otherwise determined to be acceptable (discussed in Section 2.1). Although 10 CFR Part 72 is the basis for review, it does not specify the criteria that must be used. The staff summary and conclusions are therefore presented in terms of: (1) criteria suitability and any restricting conditions that might apply, and (2) whether or not the standardized NUHOMS system design satisfies the criteria and any restricting conditions.

The structural and mechanical systems of the standardized NUHOMS system important to safety are the TC, the DSC, and the HSM including the DSC Support Assembly. Loading conditions for the individual components in the system result from all phases of normal operating conditions, exposure to natural phenomena, and accident conditions. The NRC staff evaluated all analyses for all components submitted in or with the SAR (See Reference 1). All calculations were reviewed by the NRC staff. The review included spot checks, parallel calculations, and validations of sources or expressions used. Assumed loads, material properties, and ASME, ACI, AISC, or ANSI code allowable stress limits were checked.

Applicable Parts of 10 CFR Part 72

The SAR was submitted as part of the application for a Certificate of Compliance under 10 CFR Part 72, Subpart L. Applicable design requirements are therefore stated in 10 CFR 72.236. This SER evaluation also used 10 CFR Part 72, Subpart F, for review of design bases and criteria. The guidance of Regulatory Guide 3.48 has been used for review of the comprehensiveness of the material presented in the SAR and supplementing and modifying docketed documentation.

The review was performed in stages. The stages addressed: the sources of requirements and the criteria stated as constituting the basis for the design (SER Section 2.0), the structural evaluation of the actual design against the stated and other appropriate criteria (Section 3.0), and other evaluations (Sections 3 through 13).

Materials

The materials used for fabrication of HSM (and DSC Support Assembly), DSC, and TC are identified in the corresponding fabrication specifications and/or drawings submitted in supplement to the SAR. The mechanical properties of the materials used for the design and the sources of those properties are shown in SAR Table 8.1-2.

The sources identified in SAR Table 8.1-2 for properties of steel are the ASME Boiler and Pressure Vessel Code, Section III-1 (Reference 9), Appendices, Code Case N-171-14 ASTM, and Handbook of Concrete Engineering by Fintel (Reference 12). The ASME Code is an acceptable standard and is in compliance with the quality standards in 10 CFR Part 72, Subpart F. The source identified in SAR Table 8.1-2 for the mechanical properties of concrete and reinforcing steel is the Handbook of Concrete Engineering (Reference 12), a document that is not considered to meet the quality standards of 10 CFR 72.122. However, the staff has compared the data in Table 8.1-2 with ASTM specifications for steel and the pertinent American Concrete Institute specifications for concrete which do meet Subpart F standards. The staff concurs with the data in SAR Table 8.1-2.

The source identified in SAR Table 8.1-2 for the structural properties of lead (Reference 13) is not considered a recognized standard that is consistent with the quality standards of 10 CFR 72.122(a). However, the material strength properties for lead were used conservatively. The staff concludes that the way the data were used meets the intent of the quality standards of 10 CFR 72.122(a) for material properties.

The supplemental material provided with the SAR includes supporting design calculation packages, construction drawings, and fabrication specifications. The SER review is based on supporting design calculation packages, and summary data included in Chapters 4 and 8 of the SAR. The construction drawings and fabrication specifications were used to verify that there is a one-to-one correspondence of dimensional and material property data between the drawings and the calculation packages.

10 CFR 72.3 defines structures, systems, and components important to safety which have features that: "(1) maintain the conditions required to store spent fuel or high-level radioactive waste, (2) prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage, or (3) provide reasonable assurance that spent fuel or high-level radioactive waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public."

The HSM is considered as important to safety because it provides radiation shielding and protects DSCs from damage (features 1 and 2). The DSC is important to safety since it forms the secondary confinement boundary and prevents and controls criticality (feature 1). The TC is important to safety since it provides radiation shielding during transport and prevents radioactive releases (features 1, 2, and 3). The DSC and TC are also "safety related" equipment in conjunction with their use in fuel pool facilities, per 10 CFR Part 50.

Evaluation of Ferritic Steels Against Brittle Fracture

The standardized NUHOMS system uses ferritic steels in portions of the DSC and the TC. Because ferritic steels are subject to brittle fracture at low temperatures when movement of the component may involve an impact, the use at low temperature must be evaluated. The two components are subject to slightly different criteria due to the following reasons.

Brittle Fracture Considerations for the DSC

In the case of the DSC, the brittle fracture question has two aspects. The first aspect concerns maintenance of the confinement boundary during an impact (drop accident) at low operating temperature. Because the DSC confinement boundary is manufactured entirely of SA 240 Type 304 steel, brittle fracture is not an issue. However, because the basket materials are manufactured entirely of ferritic steels, the concern is maintenance of favorable basket geometry required to ensure subcriticality. The NRC staff considers this factor to be equally important to maintenance of confinement. Hence, the staff accepts NUREG/CR-1815 (Reference 7) as appropriate for brittle fracture test methods for the DSC.

As described in the SAR, the basket components are designed according to the ASME B&PV Code, Section III, Subsection NF for component supports. The basket materials shall, according to the SAR, be impact tested in accordance with the requirements of NF-2300 at -28.9°C (-20°F). However, the NRC staff notes that this requirement is not equivalent to NUREG/CR-1815, and therefore the staff imposes limiting conditions of operation on the use of the DSC as follows.

1. No lifts or handling of the DSC at any height are permissible at basket temperatures below -28.9°C (-20°F) inside the spent fuel pool building.

2. The maximum lift height of the DSC shall be 203 cm (80 inches) if the basket temperature is below -17.8°C (0°F) but higher than -28.9°C (-20°F) inside the spent fuel pool building.

3. No lift height restriction is imposed if the basket temperature is higher than -17.8°C (0°F) inside the spent fuel pool building.

4. The maximum lift height and handling height for all transfer operations outside the spent fuel pool building shall be 203 cm (80 inches) and the basket temperature may not be lower than -17.8°C (0°F).

Brittle Fracture Considerations for the TC

In the case of the TC, which serves only as a lifting and transfer device, and not as a confinement structure, the brittle fracture question only deals with the possibility of dropping the TC/DSC, and consequences of the DSC or DSC basket brittle fracture. The staff accepts ANSI N14.6 and NUREG-0612 (References 8 and 14) as appropriate for brittle fracture test methods for the TC.

As described in the SAR, for all operations except lifting, the TC is designed and tested in accordance with the ASME B&PV Code, Section III, Subsection NC for Class 2 Components. For critical lifts, ANSI N14.6 has been used. However, paragraph 4.2.6 in

ANSI N14.6, which specifies impact testing, was not used by PNFS for the trunnions or the shell, which are ferritic steel. The SAR specifies that impact testing of ferritic steels is required in accordance with ASME requirements of Table NC-2332.1-1, and that tests shall be made at -40°C (-40°F). The guidance in ANSI N14.6 for impact testing ferritic steels is more conservative than the ASME code, i.e., the nil ductility transition temperature (NDT) shall be 4.4°C (40°F) lower than the lowest service temperature. The impact test procedure used by PNFS will, in fact, never determine the NDT. Therefore, in order to maintain the 4.4°C (40°F) margin, use of the loaded TC will be limited to a minimum temperature of -17.8°C (0°F) outside the spent fuel pool building.

In previous NRC SERs, on-site ferritic transfer casks have had limiting conditions of operation with regard to lift height and temperature (References 15 and 16). The staff imposes limiting conditions of operation on the use of the TC/DSC as follows.

1. No lifts or handling of the TC/DSC at any height are permissible at DSC basket temperatures below -28.9°C (-20°F) inside the spent fuel pool building. (The DSC basket is limiting.)

2. The maximum lift height of the TC/DSC shall be 203 cm (80 inches) if the basket temperature is below -17.8°C (0°F) but higher than -28.9°C (-20°F) inside the spent fuel pool building. (The DSC basket is limiting.)

3. No lift height restriction is imposed on the TC/DSC if the basket temperature is higher than -17.8°C (0°F) inside the spent fuel pool building.

4. The maximum lift height and handling height for all transfer operations outside the spent fuel pool building shall be 203 cm (80 inches) and the basket temperature may not be lower than -17.8°C (0°F).

It should be noted that the DSC is designed to maintain the confinement boundary for drop heights of 203 cm (80 inches) or less. Thus, even if the TC trunnion were to fail due to brittle fracture, the DSC would not release any radioactive material. The only situation which might involve lift heights above 203 cm (80 inches) would be inside the spent fuel pool building, where the -17.8°C (0°F) minimum temperature shall apply and handling of the DSC is controlled by 10 CFR Part 50 requirements.

Discussion of Concrete Constituents and Temperature Suitability

The SAR indicates that HSM concrete temperatures might exceed ACI 349 (Reference 17) limits, i.e., the 65.6°C (150°F) limit for bulk concrete, the 93.3°C (200°F) limit for local areas for normal operation or any long term period, and 177°C (350°F) for accident or other short term period. The above limits are imposed by ACI 349 for concrete in the absence of tests to evaluate the reduction in strength and to show that the concrete will not deteriorate with or without load (ACI 349, Section A.4).

The NRC staff accepts the ACI 349 criteria and, based on separate research and analysis, also accepts the following as alternative criteria in lieu of the ACI 349 temperature requirements for ISFSIs only:

1. If concrete temperatures of general or local areas do not exceed 93.3°C (200°F) in normal or off-normal conditions/occurrences, no tests or reduction of concrete strength are required.

2. If concrete temperatures of general or local areas exceed 93.3°C (200°F) but would not exceed 149°C (300°F), no tests or reduction of concrete strength are required if Type II cement is used and aggregates are selected which are acceptable for concrete in this temperature range. The staff has accepted the following criteria for aggregates (fine and coarse) which are considered suitable:

    a. Satisfy ASTM C33 requirements and other requirements as referenced in ACI 349 for aggregates.

    b. Have demonstrated a coefficient of thermal expansion (tangent in temperature range of 21°C to 37.8°C (70°F to 100°F)) no greater than $1x10^{-5}$cm/cm°C ($6x10^{-6}$ in/in/°F) or be one of the following minerals: limestone, dolomite, marble, basalt, granite, gabbro or rhyolite.

The above criteria in lieu of the ACI 349 requirements (for ISFSI only) do not extend above 149°C (300°F) for normal or off-normal temperatures for general or local areas and do not modify the ACI requirements for accident situations. For an ISFSI, use of any Portland cement concrete, where normal or off-normal temperatures of general or local areas may exceed 149°C (300°F), or where "accident" temperatures may exceed 177°C (350°F), require tests on the exact concrete mix (cement type, additives, water-cement ratio, aggregates, proportions) which is to be used. The tests are to acceptably demonstrate the level of strength reduction which needs to be applied, and to show that the increased temperatures do not cause deterioration of the concrete either with or without load.

The NRC staff considered an exception to the second criteria above for the requirements for fine aggregates only. It should be noted that the HSM roof temperature is calculated to be 121°C (250°F) on a 52°C (125°F) ambient day, for off-normal conditions, and therefore does not qualify for the following exception. This exception should not be construed as general acceptance for ISFSI usage for any normal temperatures exceeding 93.3°C (200°F) or any off-normal temperatures exceeding 107°C (225°F).

1. Fine aggregates composed of quartz sand, sandstone sands, or any sands of the following minerals: limestone, dolomite, marble, basalt, granite, or rhyolite; or any mixture of these may be used without further documentation as to the coefficient of thermal expansion.

2. Fine aggregates must satisfy requirements of ASTM C33 and ACI 349, and of the documents incorporated in those by reference.

Design Descriptions

A description of the standardized NUHOMS system is included in Section 1 of this SER. More detailed descriptions are given in this section, where appropriate, to provide the context of the evaluation. The formal description is given by the SAR and subsequent docketed documentation provided (Reference 1). The SER is based on the formal description in the SAR and not on the descriptions as summarized or extracted in the SER.

3.1 Horizontal Storage Module

3.1.1 Design Description of HSM

A general description of the HSM is included at Section 1.5.1 of this SER. Each HSM is essentially a monolithic reinforced concrete structure with a separate, bolted-on roof slab. The wall and roof thicknesses are dictated by radiation shielding considerations. The reinforcing steel must satisfy requirements for minimum steel as well as the strength requirements for all load combinations. Embedments must provide for attachment of the roof slab, DSC support assembly, door, TC, shield walls, and screens covering gaps between HSMs and between HSMs and shield walls.

The front wall of the HSM contains a round port for DSC access which is closed by a round, shielded steel and concrete door welded in place when the DSC is in place. The roof and the front wall of the individual HSM are of sufficient strength to resist tornado missiles. The HSM is unique in the way in which the modules can be configured. They may be located singly, in single rows, or in back-to-back configurations. Shielding requirements for adjacent modules are provided by the adjacent module itself. For the end modules, the 0.46 m (1 ft.-6 in.) wall thickness is not sufficient to provide the required shielding alone, and an additional 0.60 m (2 ft.) thick end module shield wall is attached to the side of the HSM. If the modules are located in a back-to-back configuration, the rear walls are protected by the abutting HSM. If the modules are located singly or in single rows, the 0.3 m (1 ft.) thick rear wall is not sufficient to provide the required shielding, and an additional 0.46 m (1 ft.-6 in.) thick rear shield wall is attached to the rear of the HSM. As the passive air cooling uses vents at the sides of the base unit at floor and roof levels, a 15 cm (6-in.) gap is left between adjacent HSMs and end shield walls.

The shield walls have been designed to the same standards as the HSM and have been analyzed for the loadings of dead weight, live load, thermal loads, and accident loads of tornado winds/missiles, earthquakes, and floods. The resulting stresses for the end module shield wall are summarized in Table 3.1.2-1 and shown to be acceptable. Significant effects resulting from the tornado missile load are discussed in Section 3.1.2.2.A. The rear shield walls, which abut the HSM, were analyzed and shown in Table 3.1.2-1 to have lower stresses than the end module shield walls. The tornado missile load was calculated to be within the allowable limits even while using very conservative analytical assumptions.

Located within and attached to the concrete structure, the DSC support structure is a welded steel assembly which supports and restrains the DSC. It is designed to satisfy the structural loads of dead weight, seismic forces, thermally induced loads, and handling loads.

3.1.2 Design Evaluation

The SAR was reviewed in conjunction with the calculation package NUH 004.0200 (Reference 18). The computer runs which were made to simulate the load conditions for the controlling PWR or BWR DSC design were also included in the review.

3.1.2.1 Normal and Off-Normal Operations

A. Dead Weight and Live Load Analysis

Tables 8.1-3a and 8.1-3b of the SAR provide the dead weights of both 24 PWR Spent Fuel Assemblies and 52 BWR Spent Fuel Assemblies respectively. The vendor has chosen to use a design weight of approximately 36,290 kg (80,000 lbs.), somewhat higher than the total dry DSC loaded weight of the heaviest assembly, for the analyses of the HSM and DSC support assemblies. Because the weight of the DSC is known, the vendor has chosen to treat it as a dead load. The weight of the concrete HSM is included as dead load. The weight of the steel DSC support structure is trivial. The vendor has also chosen to increase the dead load by five percent for all load combinations. The dead and live loads were applied to the finite element model depicted in SAR Figure 8.1-10a.

B. Concrete Creep and Shrinkage Analysis

The vendor has chosen to neglect concrete creep and shrinkage effects based on the summary *analysis* that thermal expansive forces would mitigate rather than aggravate the creep and shrinkage forces. This is acceptable for the HSM design as a conservative simplification. The HSM design satisfies minimum steel requirements of ACI 349-85 (Reference 17), which are partly based on creep and shrinkage considerations and which are more restrictive than the requirements for shrinkage and temperature reinforcement of ACI-318 (Reference 19).

C. Thermal Loads

The results of thermal analyses performed by the vendor are given in SAR Table 8.1-9b and in Figure 8.1-3a. They are derived from calculations documented in NUH004.0416 (Reference 20). For the normal operations case the thermal gradients, calculated with a long time ambient air temperature of 37.8°C (100°F), were applied to the finite element model depicted in the figure on NUH 004.0200, Rev. 5, page 10b. The thermal loads are the greatest inputs to the normal load combinations and result in the lowest margin of safety for both the concrete HSM and the steel DSC support structure. In accordance with ACI 349-85 Appendix A (Reference 17), the ratio of cracked section modulus to gross section modulus is applied to the stresses obtained from the thermal analyses. The NRC staff accepts this approach.

D. Radiation Effects on HSM Concrete

The vendor calculated the neutron and gamma energy flux deposited in the concrete and determined these levels to have negligible effect on the concrete properties. The NRC staff accepts this determination.

E. HSM Design Analysis

The vendor analyzed the HSM with its DSC support structure using the ANSYS finite element analysis computer program (Reference 21) and documented the results in reference 18 [NUH004.0200]. The staff reviewed these computations included in the original SAR and in supplemental and modifying docketed material submitted subsequently and considered as part of the SAR. The final design analysis calculations were determined to be acceptable. The analysis resulted in no load cases where the margin of safety for any structural component was less than 0.1. Margin of safety is defined as the allowable load divided by the calculated load minus 1.

3.1.2.2 Accident Analysis

A. Tornado Winds/Tornado Missiles

Tornado forces used in the SAR treat the tornado forces as normal or off-normal wind loads. This is considered very conservative. ANSI 57.9 (Reference 22) does not identify tornado loads. Such loads may be considered as "accident" loads and are so treated in ACI-349 (Reference 17) load combination expressions. The vendor chose to use the most severe tornado wind loadings specified by NUREG-0800 and NRC Regulatory Guide 1.76 (References 23 and 24) as the design basis for the standardized NUHOMS design. An analysis was also performed to determine whether the HSM would overturn or slide due to the tornado wind.

To demonstrate the adequacy of the HSM design for tornado missiles, a bounding analysis of the end and rear modules in an array was performed. The end module shield walls were evaluated for the direct impact of a 1,799 kg (3,967 lb.) automobile having a 1.86 m^2 (20 sq. ft.) frontal area and traveling at 56.3 m/sec (184.8 ft./sec). Upon impact, the three spacer plates at the top of the shield wall collapse, and the shield wall is expected to form a yield line along the length of the shield wall at mid-height. No damage will occur to the HSM; however, the damaged end module shield wall will require replacement. The rear shield walls which abut the HSM were also evaluated for the same impact load; no damage is expected. Both end and rear walls have been designed to meet the impulsive and impactive requirements of ACI 349-85. The HSM was shown to meet the minimum acceptable barrier thickness requirements for local damage against tornado generated missiles as specified in NUREG-0800.

B. Earthquake

The standardized NUHOMS HSM was analyzed for a peak horizontal ground acceleration of 0.25 g and a vertical acceleration of 0.17 g in accordance with NRC Regulatory Guide 1.60 (Reference 25) and a 7% damping coefficient in accordance with NRC Regulatory Guide 1.61 (Reference 26). These ground accelerations are in agreement with 10 CFR 72.102(a)(2) for sites which are underlaid by rock east of the Rocky Mountain Front except in areas of known seismic activity. Frequency analyses and response spectrum analyses were performed. The modal responses were combined in accordance with Regulatory Guide 1.92 (Reference 27) and the directional responses were then combined by the square root of the sum of the squares method. The vendor determined that the HSM would neither slide nor overturn due to the seismic input. The NRC finds this approach acceptable and concurs with the findings.

C. Flood

The HSM was analyzed for a 15.2 m (50 ft.) static head of water and a maximum flow velocity of 4.6 m (15 ft/sec). For this condition the vendor showed that the maximum flood induced moment is considerably less than the ultimate moment capacity of the HSM. Further calculations showed that the HSM would neither slide nor overturn under the design load condition specified. Based on the docketed material, the NRC staff finds the results acceptable.

D. Lightning

Lightning protection system requirements are site specific and depend upon the frequency of occurrences of lightning storms in the proposed location and the degree of protection offered by other grounded structures in the vicinity. NFPA 78 Lightning Protection Code (Reference 28) is to be used for evaluation of need and design of lightning protection at the site.

E. Blockage of Air Inlet and Outlet Openings

The vendor defined the design basis accident thermal event as one in which the inlet and outlet vents are blocked for 5 days with an extreme ambient temperature of 52°C (125°F) and maximum solar heat load. The HSM was analyzed for this condition referred to in NUH004.0200 as Accident Thermal (T_a). The results of thermal analyses performed by the vendor are given in SAR Table 8.1-9b. They are derived from calculations documented in NUH004.0418 and NUH004.0419 (References 29 and 30). For the accident case the thermal gradients were applied to the finite element model depicted in NUH004.0200, Rev. 5, page 10b. The thermal loads are the greatest inputs to the accident load combination and result in the lowest margin of safety for both the concrete HSM and the steel DSC support structure. In accordance with ACI 349-85 Appendix A, the ratio of cracked section modulus to gross section modulus is applied to the stresses obtained from the thermal analyses. The NRC staff accepts this approach.

F. Load Combinations

The HSM is designed and evaluated for satisfaction of load combination criteria, as identified in Table 2.5, derived from SAR Table 3.2-5. These load combinations are as stated in Regulatory Guide 3.60 (Reference 31) and ANSI 57.9 (Reference 22, paragraph 6.17.3.1) which are incorporated into the Regulatory Guide by reference. The load combinations incorporating tornado forces used in the SAR treat the tornado forces as normal or off-normal wind loads. This is considered very conservative.

Load combinations identified in the SAR for the DSC support structure are shown in Table 2.6, derived from SAR Tables 3.2-5c and 8.2-11. These load combinations are acceptable with the exception that the off-normal jammed DSC handling load was treated as an "accident" load rather than in an expression for normal (and off-normal) loads. The actual design was checked by the staff by separate calculation. It was determined that if the loads were used in the acceptable expression, the factor of safety would still be acceptable. Therefore the design is considered acceptable despite inappropriate use of the load combination.

3.1.3 Discussion and Conclusions

The maximum loads on the five major concrete structural components of the HSM (floor slab; side, front, and rear walls; and roof slab) are listed in SAR Tables 8.1-10 and 8.2-3 and in supplemental and modifying docketed material. These data were checked by the staff and found to be acceptable.

Allowable loads for bending and shear are included in Table 3.1.2-1. These are derived from the structural design and analysis package, NUH004.0200 (Reference 18), which is part of the docketed material and which has been verified by the NRC staff.

The staff review included independent development of load combinations acceptable to the NRC. The forces, computed by the vendor and the staff, as well as the resulting margins of safety computed by the staff for the concrete components of the HSM are included in Table 3.1.2-1 and found to be acceptable.

Table 3.1.2-2 presents the results of examination of the DSC support structure stresses and load combinations. The submitted data are extracted or derived from the structural design and analysis package which is part of the docketed material (Reference 1). Table 3.1.2-2 shows analyses for the load combination for the support rails, cross beams, support columns, and lateral tie beams of the DSC support assembly. The allowable stresses shown in the tables are those developed by the NRC staff based on the submitted calculations. The calculated maximum combined load stresses shown in the table are below the allowable stresses. The axial and bending stresses, divided by their respective allowable stresses, are further combined in order to obtain an interaction margin of safety. This combination, per the AISC Specification for Structural Steel, June 1, 1989, Paragraph H1 (Reference 32) must have a value not greater than 1.0. Review of Table 3.1.2-2 shows that the selection of the steel sections used for DSC columns, cross beams, rails, and tie beams was found to be acceptable.

The rail-transverse member interconnection assembly, web stiffeners installed in the W8x35 members, and other miscellaneous HSM steel were checked and determined to be satisfactory. This included door and supports, collars, brackets, TC restraint assembly, heat shield, seismic restraints, and end stops.

The overall result of the review of the HSM and DSC support assembly structural design criteria, load combination, and final design is that the HSM and DSC support, as represented in the current docketed material (Reference 1), are considered to be structurally acceptable and meet the requirements of 10 CFR Part 72.

3.2 Dry Shielded Canister

3.2.1 Design Description of Dry Shielded Canister and Internals

There are two DSCs for the standardized PWR and BWR NUHOMS systems. The DSC is the secondary confinement barrier for the spent fuel. The primary confinement barrier is considered to be the fuel cladding. Each DSC will accommodate 24 PWR irradiated spent fuel assemblies or 52 BWR irradiated spent fuel assemblies. The DSC fits inside the transfer cask for handling and transfer operations, and is moved out of the TC and into the HSM with the hydraulic ram.

The main structural parts of both versions of the DSC consist of the following stainless steel items: a 1.6 cm (5/8-inch) thick shell, a thick outer bottom cover, a thick outer top cover plate, a thin inner top plate and a thin inner bottom plate. The BWR DSC has a total of nine 3.8 cm (1.5-inch) thick spacer discs made from SA-516 ferritic steel, whereas the PWR DSC has eight 5.1 cm (2-inch) thick spacer discs. Each DSC has four 7.6 cm (3-inch) diameter spacer support rods. The PWR DSC has twenty-four square fuel guide sleeves. The BWR DSC has slots cut in the spacer discs to accept borated stainless steel poison plates. Square shaped holes accommodate fifty-two BWR assemblies. In addition to the above structural items, there are two steel shield plates, and numerous small items associated with a grapple, vent and siphon system, and lifting lugs.

The SAR was reviewed in conjunction with the calculation package NUH 004.0202 (Reference 33) plus all of the computer runs which were made to simulate all the load conditions for both PWR and BWR DSC designs.

With one exception, the DSCs are designed as pressure vessels in accordance with the ASME B&PV Code Division 1 Section III Subsection NB-3000-1985 (Reference 9). Material qualifications are in accordance with Subsection NB-2000. Fabrication and inspection are to be done in accordance with Subsections NB-4000 and NB-5000, respectively. Proof pressure tests are to be carried out according to NB-6000. The exception is the weld design and inspection at the top and bottom of the DSC.

The double seal welds at the top and bottom of the DSC do not comply with all the requirements for the ASME B&PV Code, Section III, Subsection NB. The inspection procedures outlined in the SAR do not comply with the code; however, the NRC staff has determined that an exception to Code requirements for volumetric weld inspection is permissible due to the following reasons:

1. The closure to the confinement boundary is a double-weld design, i.e., two weld joints provide confinement.

2. The gauge pressure (for normal operation) inside the DSC is on the order 1 psig. Therefore, pressure stresses are very low.

3. The test method of ensuring a gas tight seal for the inner top seal weld is helium leak detection which is very sensitive. Also dye penetrant testing will be performed at two levels including the weld root pass and cover pass on the outer seal weld to ensure no weld surface imperfections. The test method of ensuring a gas tight seal for the bottom welds consists of a helium leak test by the fabricator for the inner seal weld in accordance with ASTM E499, in addition to two levels of dye penetrant testing for this weld. For the outer seal weld a multi-level dye penetrant test is specified.

3.2.2 Design Evaluation for DSC

3.2.2.1 DSC Normal Operating Conditions

The dry shielded canister was analyzed for: (1) dead weight loads, (2) design basis operating temperature loads, (3) internal pressure loads and (4) normal handling loads. Table 3.2.2-1 of this SER summarizes all the stress analysis results for normal operating conditions. The summary table shows stresses for each DSC component for each load condition analyzed by PNFS and the corresponding stress as verified by the NRC staff. Each stress intensity value was compared to the allowable stress for the particular material at the stated temperature as defined by the ASME Code for Service Levels A and B conditions. All calculated stresses are below allowable levels.

A. Dead Weight Loads for DSC

The dead load analysis for the DSC is presented in Section 8.1.1.1.A of the SAR. Both vertical and horizontal orientations of the DSC were considered. For the horizontal orientation the DSC inside the HSM, resting on the support rails, as well as the DSC inside the TC were modeled. The weights are shown in Tables 8.1-3a and 3b of the SAR, and stresses are shown in Table 8.1-7a and 7b of the SAR. The NRC staff reviewed these stress levels and reports them in Table 3.2.2-1 of this SER. Basically, all stresses are lower than the ASME B&PV Code allowable stresses by a substantial margin.

B. Design Basis Internal Pressure

The design basis normal internal pressure for the DSC is 47.6 kPag (6.9 psig), however, the analyzed pressure is 69 kPag (10 psig). This provides some conservatism in the analysis. Tables 8.1-4a and 8.1-4b of the SAR show five cases for operating and accident pressures. The ANSYS (Reference 21) finite element code was used to model the internal pressure load for the top and bottom portions of the DSC. PNFS used 345 kPag (50 psig) for the internal pressure and then multiplied the stress results by a factor corresponding to the particular load case per SAR Table 8.1-4a and 4b. Additionally these cases bound the internal pressure load of 55.2 kPa (8 psi) which exists during the helium leak test of the bottom inner seal weld during fabrication.

Thus, there are two seal welds for the pressure boundary at the top and the bottom of the DSC; i.e., the weld for the outer top cover plate, and an inner weld applied to the inner top plate. The outer top cover plate is the primary structural component, and the weld at that joint is much more substantial than the weld at the inner cover plate. The pressure stresses in the weld of the top inner and outer cover plates were evaluated for normal and accident cases and found to be below the allowable limits. The same type of analysis used to evaluate the top portion was used to evaluate the bottom position of the DSC. Shell stresses were evaluated for the remainder of the DSC by an ANSYS model. The computer model used 345 kPag (50 psig) as an internal pressure load. Because of a linear response of stress to the internal pressure load, the normal, off-normal and accident pressure cases could be evaluated simply by using factors of 0.2 and 1.01, respectively. All stress intensities were evaluated and were below allowable levels for pressure stress.

Tables 8.1-7, 8.1-7a, 8.1-7b and 8.1-7c of the SAR report the DSC pressure stresses for normal pressure of 69 kPag (10 psig). The staff reviewed these pressure stresses and concurs with them. The results of the SER are shown in Table 3.2.2-1 of this SER.

C. Design Basis Operating Temperature

PNFS has provided for axial thermal expansion of the basket assembly and the inner surfaces of the top and bottom end plates. Thus, no thermal stresses are induced due to restriction of expansion of internal parts. Similarly, PNFS has sized the spacer disc smaller than the inside diameter of the DSC shell to preclude induced thermal stresses. PNFS performed four different finite element analyses to determine thermal stresses for differential expansion of the shell, the spacer disc, and the shell/end cover interface. The axial thermal gradient as well as the circumferential thermal gradient for the shell were modeled using thermal input from separate temperature evaluation (NUH004.0407 Rev. 0, Reference 34). These analyses were performed at all ambient conditions ranging from -40°C to 52°C (-40°F to 125°F), with and without solar loads, both horizontal and vertical orientation, and with and without air gaps. The parametric study was performed for both the PWR and BWR designs. The maximum temperature calculated determines the material allowable stresses, and the NRC determined that 260°C (500°F) bounds all cases for the DSC shell, disc, ends, and rods. Tables 8.1-13, 8.1-13a, in the SAR report the results of the DSC temperature distribution.

The thermal stresses are always defined as "secondary stresses" by the ASME B&PV Code. This means that higher allowable stresses are permitted and only Service Level A (for normal operations) and Service Level B (for off-normal operations) need be considered.

For normal operations at an ambient temperature of -40°C (-40°F), the maximum primary plus secondary stress for all thermal cases considered is 86,877 kPa (12.6 ksi) for the DSC shell. The allowable stress is 386,810 kPa (56.1 ksi). The BWR spacer disc has a thermal stress of 265,460 kPa (38.5 ksi), and, the allowable stress for the disc material is 448,860 kPa (65.1 ksi), so this is acceptable. The staff has reviewed all the documentation provided with the SAR and concurs that thermal stresses for the DSC for normal operations meet ASME B&PV Code requirements. They are shown in Table 3.2.2-1 of the SER.

D. DSC Handling Stress

The DSC handling load cases were divided into three groups, each requiring different analytical techniques. The design basis handling load is 50% of the DSC dry weight applied axially as it would be during normal operations when the loading ram is used to insert or extract the DSC from the HSM. The 50% factor is based on actual data obtained during the operation of a similar design at the ISFSI for the Oconee nuclear plant. Other normal cases are dead loads applied ± 1 g vertically, ± 1 g horizontally, and ± 1 g axially, and ± 1/2 g acting in all three orthogonal directions simultaneously. These could occur during transfer in the TC. The off-normal case is a jammed condition occurring inside the TC or HSM. All stresses in all components were evaluated and found to be below the ASME B&PV Code allowable. In addition to the confinement boundary, the grapple and lifting lugs were analyzed for the design basis loads, both normal and off-normal. Both components were evaluated against ASME allowables and found to be satisfactory. The resulting stresses are much lower than allowable stresses, as shown in Table 3.2.2-1 of the SER.

3.2.2.2 DSC Off-Normal Events

Three off-normal events were evaluated by PNFS for the DSC. They were off-normal pressure, jammed DSC during transfer and off-normal temperature. The off-normal temperature of -40°C (-40°F) ambient and the jammed DSC bound the range of loads.

A. Jammed DSC During Transfer

The basis for the postulated off-normal event, involving jamming of the DSC during transfer into the HSM, is the axial misalignment of the DSC. Should this occur, the hydraulic ram could exert an axial force equal to the weight of the loaded dry DSC, before a relief valve would prevent further load. A detailed finite element model including the actual load path through the grapple ring was performed to estimate this loading. The weight chosen by PNFS was 36,290 kg (80,000 pounds), a figure which exceeds the actual dry loaded weight, thereby affording additional conservatism. The bending stress in the bottom cover plate of the DSC is the highest stress anywhere in the DSC and is smaller than the allowable. Also, the bending stress in the DSC shell is well below the allowable stress. These results are shown in Table 3.2.2-2 of this report.

B. DSC Off-Normal Thermal/Pressure Analysis

The off-normal temperature range was taken as -40°C to 52°C (-40°F to 125°F) for the DSC inside the HSM (and inside the TC). The off-normal ambient temperature of -40°C (-40°F) is the basis for the high thermal gradient for the spacer disc, and the top and bottom corners of the shell. These high thermal gradients result in high thermal stresses which are shown to be lower than the allowable stresses for secondary stress.

C. DSC Off-Normal Pressure

The design basis off-normal internal pressure acting in the DSC is 38.6 kPag (5.6 psig), however the value used in the analysis is 69 kPag (10.0 psig). Both inner and outer DSC pressure boundaries were analyzed for the off-normal pressure case. Because the applied value of 69 kPag (10 psig) is the same for the off-normal and the normal, the stress results shown in Table 3.2.2-2 are the same as shown in Table 3.2.2-1.

D. DSC Load combination for Normal and Off-Normal Conditions

Table 3.2-5a of the PNFS SAR outlines the different load combinations considered for normal and off-normal conditions and accident. These conditions correspond to Service Levels A, B, C, and D of the ASME B&PV Code. Altogether, Table 3.2-5a of the SAR shows 17 combinations for all service levels. However, due to the fact that PNFS combined several combinations because normal and off-normal pressure cases are actually identical, and all thermal cases are bounded by one temperature providing the highest thermal gradient, only nine unique combinations are shown in Tables 3.2.2-3, -5, and -7 of this SER. The staff summarized the combinations as described and finds that all stresses are below the allowables for Service Levels A and B. These are given in Table 3.2.2-3 of the SER. PNFS references are Tables 8.2-9a and 9b.

3.2.2.3 DSC Accident Conditions

Section 8.2 of the SAR defines the accident conditions associated with the standardized NUHOMS system. The accident conditions which were examined for the DSC are: (1) earthquake, (2) flood, (3) accident pressure, (4) accident thermal, and (5) accidental drop of the TC with DSC inside. Of these accidents, the drop case is by far the most severe. The SAR classifies the thermal accidents, the pressure accident, and the drop accidents as Service level D conditions, and the remaining accidents including seismic and flood as Service Level C conditions. The NRC staff concurs with this classification.

A consequence of classifying the thermal accidents as Service Level C or D is that the ASME B&PV Code does not require any stress analysis because of the ASME definition of thermal stresses as "secondary" stresses or "self-relieving" stresses. The only required consideration of the accident thermal cases was in a reduction of material properties at the higher temperature, which was properly accounted for.

A. DSC Seismic Analysis

The standardized NUHOMS system is designed to withstand seismic events which have a maximum horizontal ground acceleration of 0.25 g and a maximum vertical component of 0.17 g. These ground acceleration values are in agreement with 10 CFR 72.102(a)(2) for sites which are underlaid by rock east of the Rocky Mountain Front, except in areas of known seismic activity. NRC Regulatory Guide 1.60 (Reference 25) was used to determine dynamic load amplification factors for the horizontal and vertical directions. NRC Regulatory Guide 1.61 (Reference 26) was used to estimate the critical damping value for the DSC and the HSM. The DSC was conservatively correlated with large diameter piping and therefore has a damping value of 3%.

The DSC was evaluated for two distinct modes of vibration to establish fundamental frequencies, which in turn was used with Figures 1 and 2 in Regulatory Guide 1.60 to estimate the amplification. The shell cross-sectional ovaling mode turned out to be the only mode of interest since it is 13.8 Hz. The beam bending mode is too high to cause a dynamic amplification factor at 62.8 Hz. The resulting spectral accelerations for the DSC shell ovaling mode are 1.0 g and 0.68 g for horizontal and vertical directions, respectively. PNFS applied a factor of 1.5 to these accelerations to account for a multi-mode excitation. The applicant used the results of 75 g vertical drop analyses factored by (1.5 x 0.68/75) to obtain stresses for the DSC. For the horizontal orientation, PNSF used the results of the horizontal drop analysis factored by (1.5x1x2/75) to obtain the DSC stresses. DSC shell stresses obtained from vertical and horizontal analyses are summed absolutely. These are recorded in Table 3.2.2-4.

The DSC was also evaluated for roll-out of the support rails. Horizontal and vertical accelerations of 0.37 g and 0.17 g were applied to the center of the DSC. The resulting factor of safety against roll-out was 1.23 according to an NRC staff evaluation. This corresponds to 1.30 as calculated by PNFS (Reference 1).

B. DSC Flood Condition

The design basis flood is specified in the SAR as 15.2 m (50 feet) of water with a maximum flow velocity of 4.6 m/s (15 feet per second). The flood condition is postulated to occur only when the DSC is housed inside the HSM. The consequences of the water flow will not affect the DSC inside the HSM, and the consequences to the HSM are reported in another section of the SER. Therefore, the DSC is only affected by the static head.

The DSC shell and outer cover plates and inner cover plates were modeled with a finite element analysis. PNFS modeled both inner and outer cover plates coupling the nodes of both plates to allow transmission of forces perpendicular to the plates. A more conservative approach would have been to assume no inner plates. However, the resulting stresses due to the 149.6 kPa (21.7 psi) external pressure are so small (~6,895 kPa (1 ksi)) that even if this conservative approach had been used, the resulting stresses would still be lower than the allowables. See Table 3.2.2-4.

C. DSC Accident Pressure

The bounding DSC internal accident pressure is 379.7 kPag (50.3 psig) according to Section 8.2.9 of the SAR. The maximum ambient temperature of 52°C (125°F) is assumed. This accident is postulated for a DSC inside the HSM, which has all inlet and outlet vents blocked, i.e., the adiabatic heat-up case. Assumptions are that the cladding of all fuel rods failed and that 100% of the fill gas and 30% of the fission gas are released inside the DSC. Under these conditions, the internal pressure could reach 379.7 kPag (50.3 psig). Table 3.2.2-4 of this SER shows the stress results of this case. All stress intensities are lower than the allowables.

The heat-up time period which is postulated by PNFS is five days. At that time the fuel cladding temperature is still below the cladding limit of 570°C (1058°F) for accident conditions. The adiabatic accident case bounds all thermal accidents and shows the need for daily inspection of air inlets and outlets. A more complete discussion of thermal performance may be found in Section 4 of the SER.

It should be noted that PNFS stated that 204.4°C (400°F) is the appropriate temperature to select the allowable stresses for the materials in the DSC (NUH004.0202 p.161). Table 8.2-9e of the SAR stated that 260°C (500°F) was the correct temperature. However, Table 8.1-

13 of the SAR indicates that the DSC shell reaches a maximum temperature of 303.9°C (579°F) for this accident; therefore, the NRC staff used lower material allowable stresses for this case.

D. DSC Load Combination (Thermal Accident) for Service Level C Accident Conditions

Tables 8.2-9c and -9d of the SAR show the results of two load combinations. These are the enveloping load combinations defined in Table 3.2-5a of the SAR. Table 3.2.2-5 in the SER shows the results of the three unique load combinations of C1, C2, and C7. There is a slight discrepancy between the allowable stress as reported in the SAR and as reported in this SER. The discrepancy arises because the maximum DSC temperature, as reported in the SAR is 304.4°C (580°F) for the accident pressure case; whereas, the allowables used in the SAR were based on 260°C (500°F). The NRC staff used the allowables associated with 304.4°C (580°F). The staff has recorded the results in Table 3.2.2-5 of the SER. The conclusion is that the design for the DSC is adequate.

E. Accidental Drop of TC with DSC

Because the cask drop accidents postulated in the SAR cause the highest stresses in the both the DSC and the transfer cask, it is appropriate to discuss the basis for selecting some of the parameters and assumptions for this case. All drop situations that were postulated in the SAR involve dropping the TC, with the DSC inside, at a maximum height of 203 cm (80 inches). The NRC staff considers these assumptions reasonable, because the loaded DSC will always be in the TC or inside the HSM whenever it is outside of the spent fuel pool building. The centerline of the HSM is located at 259 cm (102 inches) above the base pad; and therefore, the maximum drop height would be about 173 cm (68 inches) for the DSC, should it fall off of the transport trailer during loading or during transport between the spent fuel pool building and the ISFSI site. Thus, 203 cm (80 inch) drop is conservative.

<u>Discussion of PNFS Design Methodology</u>

One of the major cornerstones of the PNFS justification for the deceleration levels associated with the postulated cask drop accident is a research report published by EPRI (Reference 35). This work has attempted to correlate average deceleration values acting on the cask as a function of several parameters, including drop orientation, drop height, and concrete target hardness. The latter is a non-dimensional variable which includes the following parameters: concrete elastic modulus, concrete ultimate strength, soil elastic modulus, soil ultimate strength, steel reinforcement ratio and footprint of cask. The cask itself is considered to be infinitely rigid, so that from an absorbed energy standpoint, all kinetic energy would be absorbed by the target. The resulting cask deceleration values would represent an upper bound compared to an assumption which permitted the cask to absorb any elastic or plastic energy as a result of the impact. The EPRI report NP-4830 (Reference 35) was supplemented by a second report NP-7551 (Reference 36) which correlated a small sample of existing experimental evidence of cask drops to the analytical presentation made in the NP-4830 report. The magnitude of the deceleration for each drop case was selected as the design criteria in Section 3 of the SAR as 75 g for either vertical or horizontal drop orientations and 25 g for the corner drop. The SAR values are based on an EPRI report (References 35 and 36). The target chosen for this scenario is a 91 cm (36-inch) thick under-reinforced concrete slab.

PNFS argues that drop accidents which might occur while the DSC, inside the TC, is enroute to the HSM, would be less severe than a drop accident of the DSC/TC on the reinforced concrete pad/apron adjacent to the HSM. Their argument is based on the fact that the road which would be used as the route between the spent fuel pool building and the HSM location would typically be 30 cm (12 inches) or less of concrete or asphalt on a compacted gravel bed. The "target" or impact surface would thus be significantly "softer" than the loading/unloading approach slab near the HSM. The thickness of this slab is not specified by PNFS, but would be no thicker than 91 cm (36 inches) and would be designed in accordance with ACI-318-83 (Reference 19).

Because references 35 and 36 do not document the deceleration time history, it was necessary to establish damping coefficients and the representative time histories for the three orientations, in order to predict appropriate dynamic load factors (DLF). The SAR provided additional material in Appendix C that included references to drop test data for a 81.6 t (90-ton) rail cask (Reference 37). The time histories from this reference were used to determine the DLFs for the different drop orientations.

Based on the documentation provided and the references cited, NRC staff concludes that the DLFs for the vertical, horizontal, and corner drops are 1.50, 1.75, and 1.25, respectively. These factors, when multiplied by the unfactored deceleration levels obtained from reference 36, produced values of 73.5 g, 66.5 g, and 25.0 g for the three drop orientations, which compare favorably with the deceleration values of 75 g, 75 g, and 25 g selected by PNFS in their design criteria. The staff determined that a damping value of 7% is conservative. This was based on sources in the open literature as well as the information provided by PNFS.

<u>Discussion of NRC Staff Evaluation of Accidental Drop</u>

The vendor's use of the EPRI report methodology (References 35 and 36) to determine design deceleration loads is not currently endorsed by the NRC. Therefore, the staff independently calculated elastic and plastic strains associated with the absorption of the kinetic energy resulting from dropping a fully loaded DSC through 203 cm (80 inches). The height of 203 cm (80 inches) is conservative because the DSC is not raised more than 173 cm (68 inches) during all transfer operations outside the spent fuel pool building.

The total strain rating on the DSC shell was calculated to be 1.28% compared to the minimum 40% elongation (strain) required for the material by the ASME B&PV Code, Section II, Part A. Thus, the strain due to a vertical drop is a very small fraction of the total strain capacity.

The general membrane stress in the DSC shell was calculated from the strains and the moduli of elasticity and found to be equal to 142,730 kPa (20.7 ksi). This value is well below the ASME Code allowable of 289,590 kPa (42 ksi) for Service Level D conditions. Therefore, the factor of safety, as determined by the energy method is slightly in excess of 2 for the general membrane stress.

Based on these independent calculations, the NRC staff confirmed that the design of the DSC will provide ample margin of safety during a drop accident. In addition to the independent analysis described above, NRC staff evaluated all the design calculations submitted by PNFS and reported the results in Table 3.2.2-6. These design calculations were performed using an NRC-approved finite element computer program and are described in the next section of this SER.

The staff concluded that a drop of the loaded DSC from a height greater than 38 cm (15 inches) may cause damage to the DSC and the stored fuel. Because the ASME Code, Section III for Service Level D permits plastic deformation, portions of the DSC shell and basket may sustain damage, without compromising the confinement boundary or geometry of the spent fuel array. However, such potential damage is cause for limiting conditions of operation and surveillance.

a. The loaded DSC/TC shall not be handled at a height greater than 203 cm (80 inches) outside the spent fuel pool building.

b. In the event of a drop of a loaded DSC/TC from a height greater than 38 cm (15 inches) (a) fuel in the DSC shall be returned to the reactor spent fuel pool; (b) the DSC shall be removed from service and evaluated for further use; and (c) the TC shall be inspected for damage. The affected fuel may be subsequently transferred to dry storage if it meets the requirements for storage. The DSC may be returned to service or disposed of depending on the results of the evaluation. The TC may also be returned to service if the sustained damage is repairable.

These conditions are reflected in the conditions for system use, Section 12.2.10.

F. Discussion of Finite Element Models for Cask Drop

Reference 33 is a calculation package which presents all the structural analyses for the DSC. Together with 33 separate ANSYS computer runs, the calculation substantiates the design of the PWR and BWR DSC. NRC staff evaluated the entire package. In all cases, the SAR uses the ANSYS (Reference 21) finite element code to model the DSC and TC cask components. PNFS ran nineteen models for all drop cases. These cases included both PWR and BWR DSCs, in vertical and horizontal orientations. Each part of each DSC modeled. For the vertical drop, an axisymmetric load and an axisymmetric geometry were modeled, using an equivalent 75 g static load. For the horizontal drop case, an axisymmetric structure with non-axisymmetric loading was modeled. The asymmetrical loading was approximated with a Fourier series technique in conjunction with an ANSYS element type designed to facilitate the use of the Fourier (harmonic) series. PNFS did not model the corner drop because they stated that the 75 g vertical and 75 g horizontal drop orientations are bounding for the 25 g corner case.

PNFS modeled thirteen cases for the horizontal drop orientation and seven cases for the vertical drop orientation. The shell, top cover plate and inner top plate were modeled using axisymetric geometry for top end drops. The shell, bottom cover plate and inner bottom plate were also modeled using axisymetric geometry for bottom end drops. The PWR loaded basket mass is greater than the BWR basket mass, therefore, the loads and the stresses are greater for the PWR DSC than for the BWR DSC.

The shell, top and bottom cover plates, and top and bottom inner plates were modeled for 3 horizontal orientations, i.e., 0°, 18.5°; and 90° azimuth oriented upward. These orientations correspond with possible drops at 0° and 90° azimuth for the DSC inside the TC falling off the transfer trailer, and 18.5° for a TC/DSC falling directly onto one of the cask rails on the support trailer.

Because the spacer discs for the PWR and BWR DSCs are completely different parts, it was necessary for PNFS to analyze both types of discs and both variations of plate thickness. For the 75 g vertical drop cases, the loads acting on the PWR spacer discs include the support rods, guide sleeves and oversleeves. For the 75 g vertical drop cases, the loads acting on the BWR spacer discs include the support rods, poison plates and poison plate support bars. An elastic plastic analysis using classical bilinear material properties with a conservative tangent modulus of 5% was used by PNFS for the top spacer disc for the 52-B DSC. The stresses were evaluated according to the ASME B&PV Code requirements and found to be acceptable. The calculation package reported the results of each computer run and summarized the results by listing the highest stresses in a particular component for the given drop orientation.

Each type of spacer disc was also modeled separately for two side drop orientations. The two orientations used for the 75 g 24-P basket were 0° and 90° azimuth upward. Elastic-plastic analysis was used to account for local yielding. Three azimuth orientations were used to model the 52-B spacer disc in the horizontal drop case. Elastic-plastic material properties were specified to predict stresses and displacements more accurately than using simple elastic properties. Again, all of the stresses which were calculated for each of the separate runs were reported by PNFS and evaluated by the NRC staff. Summary tables showing only the maximum stresses were provided in the calculation package.

Buckling analyses were performed on the spacer disc and support rods for vertical drop orientations, and for the 52-B poison plates in the side orientation. There is a factor of safety of 2.2 for the 52-B spacer disc out-of-plane, 1.8 for the 24-P spacer disc out-of-plane, 8.36 for the 24-P support rod (vertical drop), and 1.37 for the vertical drop case for the 52-B poison plates.

The support rods were analyzed for compliance with stress levels below allowable stress levels for the side drop and vertical end drop orientations. The top end drop resulted in a much higher stress level than the horizontal drop; however, the stress is below the allowable stress level.

Table 3.2.2-6 of the SER presents a summary of all the results of the 20 ANSYS computer analyses which were outlined above. Both vertical and horizontal results are given. The table shows that all calculated stress levels are below the ASME B&PV Code allowables for Service Level D. The table lists results that PNFS reported in the calculation package and the results obtained by NRC staff evaluation of the calculation package including all of the computer runs. The reader of Table 3.2.2-6 will note some differences between results of PNFS and NRC staff. These differences are attributed primarily to location of stresses in the actual models. NRC staff has consistently been more conservative than PNFS, and still the stress levels are lower than allowable levels.

The weld stresses for the critical secondary confinement joints between the top outer cover plate and the shell and the bottom outer cover plate and the shell were also evaluated by NRC staff. A joint efficiency factor of 60% per ASME B&PV Code ND-4245a(3) was used for a Class C, Type 3 weld, which is non-volumetrically examined. Table 3.2.2-6 shows the results of the individual load cases. All of the calculated stresses for the welds are below the allowable stress levels.

G. DSC Load Combination for Service Level D Accident Conditions

The SAR uses Service Level D for accident case allowable stresses. While NRC staff concurs with this decision, it must be coupled with the operating controls and limits as proposed in Section 10 of the SAR. Following a cask drop of 38 cm (15 inches) or greater, the DSC must be retrieved, and the DSC and the internals must be inspected for damage. NRC staff sets this operational control because it is in keeping with the high allowable stress of the Service Level D, i.e., permanent deformations of the DSC confinement boundary and the DSC internals are permitted under Service Level D conditions. Additionally, given the predicted failure of the weld between the guide sleeve and spacer disc at a deceleration below the 75 g level predicted, there is justification for inspection of the 24-P DSC and internals following any cask drop of 38 cm (15 inches) or greater. Note that for the 52-B DSC basket, the poison plates are designed to remain in place during the postulated drop accidents. Therefore, there is no possibility of an unanalyzed geometry in the poison plates with regard to subcriticality.

In both of the load combination cases D2 and D4, the term Pa, or accident pressure, is used. The maximum temperature associated with the maximum internal pressure of the DSC shell is given as 304°C (579°F) for the adiabatic heatup inside the HSM. However, this temperature is not reached when the DSC is inside the TC, which is the only scenario considered for drop. The maximum DSC shell temperature for the DSC inside the TC appears to be 231°C (447°F) from Figure 8.1-3b. This temperature is higher than 211°C (411°F) given in Table 8.1-13 for the DSC inside the TC. Therefore for the load combination cases of D2 and D4 which include accident pressure, the material properties for a DSC temperature of 260°C (500°F) is used to be conservative. Also conservative is the fact that the maximum pressure stress is used in cases D2 and D4; however, this pressure occurs under adiabatic HSM heatup conditions which are not possible with the DCS inside the TC.

Table 3.2.2-7 summarizes the results of the bounding two load combinations, D2 and D4. As noted in the table, the stress intensities are conservatively combined irrespective of location in the DSC, unless otherwise noted. For the case of the DSC shell, this conservative procedure was not used. However, the ASME B&PV Code requires that the stress intensities at any point for all load combinations shall be lower than an allowable stress; i.e., it is not required to combine stresses irrespective of location. As may be seen from Table 3.2.2-7, these conditions are met.

3.2.2.4 DSC Fatigue Evaluation

Section NB-3222.4a of Section III of the ASME B&PV Code (Reference 9) requires that components be qualified for cyclic operation under Service Level A limits unless the specified service loadings of the components meet all six conditions defined by NB-3222.4d. Although it is superficially clear that the DSC is inherently not subjected to high cycles of pressure, temperature, temperature difference, or mechanical loads, the Topical Report (TR) (Reference 38) previously evaluated each of the six conditions defined by the ASME B&PV Code in the submittal of the TR. NRC staff evaluated the previous analysis and concurs with the finding that the service loading of the DSC meets all conditions (Reference 39). Therefore a separate analysis is not required for cyclic service. NRC staff does not find any basis for finding that the service loading will deviate from the conditions assumed in the SAR, consequently no fatigue analysis is required.

3.2.2.5 DSC Corrosion

The suitability of stainless steel casks for containment of spent fuel was reported in "Laboratory Experiments Designed to Provide Limits on the Radionuclide Source Term for the Nevade Nuclear Waste Storage Investigateion's Project" by V. M. Overby and R. D. McCright. (SAND 85-0380, Reference 71). Stainless steel performs well in oxidizing conditions. Average oxidation rates for stainless steel are: 0.15 micrometer/yr submersed in water at temperatures in the 50°C to 100°C; 0.16 micrometer/yr in 100°C saturated steam, and 0.001 to 0.08 micrometers/yr in 150°C unsaturated steam. The oxidation environment of the DSC in the standardized NUHOMS system is expected to be less than the above conditions. It is therefore concluded that even under worst oxidation condition, the corrosion depth of the DSC for a 50 year design life is insgnigicant and will not affect the DSC from performing its intended safety functions.

3.2.3 Discussion and Conclusions for DSC

Tables 3.2.2-1 through 3.2.2-7 have summarized the structural evaluation of the DSC for Service Levels A, B, C and D. All of the results show that the DSC complies with all of the requirements for ASME B&PV Code, Section III, Subsections NB and NF (Shell and basket respectively). This evaluation includes many conservatisms as discussed in Section 3.2.2.

3.3 Transfer Cask

3.3.1 Design Description of Transfer Cask

The TC is used to house the DSC inside of the spent fuel pool building and during transport operations between the spent fuel pool building and the HSM. It is designed to provide radiological shielding during all operations when the DSC has spent fuel in it. It is also designed to provide protection to the DSC against potential natural and operational hazards during transport of the DSC to the HSM.

The main structural parts of the TC consist of the following items: a 3.8 cm (1.5-inch) thick shell, top and bottom machined rings which join the shell to a 5.1 cm (2-inch) thick bottom cover plate, and a 7.6 cm (3-inch) thick top cover plate. Some of these items are stainless steel and other are ferritic steel. For lifting and transporting purposes, two ferritic steel upper trunnions are welded to the structural shell. For tilting and transport purposes only, two stainless lower trunnions are welded above the centerline of the structural shell. The shell itself may be either stainless steel or ferritic steel, depending on fabricator's option. The top cover plate is fixed to the top structural ring with sixteen 1.75-8 UNC bolts.

The payload of the TC is 40,826 kg (90,000 pounds) and the total gross weight with fuel and water but no top lid is 89,566 kg (197,500 pounds) enveloping 84,186 kg (185,600 pounds) with fuel, top lid but no water. These values are for the 52-B DSC assembly, which are slightly larger than the TC for the 24-P design. However, the load used for the analyses is 90,700 kg (200,000 pounds), thus providing a small conservatism.

The TC is classified as "important to safety" and has been designed to meet several criteria depending on the function. The primary function of transporting the DSC inside the TC is covered by the ASME B&PV Code, Section III, Subsection NC for Class 2 components. Load combinations have been extracted primarily from the ASME B&PV Code. The lifting and tilting trunnions have been designed to meet ANSI N14.6-1978. Table 3.2-1 of the SAR provides a complete summary of the design criteria. Material qualifications are in accordance with Subsection NC-2000. Fabrication and inspection are to be done in accordance with Subsection NC-4000 and NC-5000, respectively.

The review of the structural integrity of the TC is presented according to function, i.e., either transfer function, or lifting/and tilting function. The ASME Code governs for transfer, whereas ANSI N14.6 governs for the lift and tilt trunnions.

The version of the TC which is specific for the PWR fuel has previously been evaluated by the NRC staff (Reference 16). However because the standardized NUHOMS system design heat load for 5-year-old PWR fuel is 24 kW instead of 15.8 kW, PNFS performed new thermal stress analysis which is evaluated in this SER. Additionally, because the BWR fuel is typically longer than the PWR fuel, PNFS provides a collar which is to be bolted on the top of the PWR TC. The stainless steel collar is designed in accordance with the ASME B&PV Code Section III, Subsection NC.

Material Considerations

This SER previously discussed the possible brittle fracture of ferritic steels. Paragraph 4.2.6 of ANSI N14.6 establishes low temperature criteria which are acceptable to the NRC staff for the TC design. This criteria was not used by PNFS. Instead the vendor used a test temperature of -40°C (-40°F) and an impact test procedure according to the ASME B&PV Code, Section III, Subsection NC. As a consequence of not using ANSI N14.6, the use of the TC shall be restricted. Inside the spent fuel pool building, for DSC/TC lift heights above 203 cm (80 inches), the minimum temperature shall be -17.8°C (0°F) or higher. For DSC/TC lift heights of 203 cm (80 inches) or lower, inside the spent fuel pool building, the minimum temperature shall be -28.9°C (-20°F) which corresponds to the minimum DSC basket use temperature. For all use outside the spent fuel pool building, the minimum temperature shall be -17.8°C (0°F). This corresponds with ANSI N14.6 paragraph 4.2.6, as well as the SAR paragraph 10.3.15. (Note all TCs have ferritic steel trunnions which are part of the lifting device.)

3.3.2 Design Evaluation of the Transfer Cask

3.3.2.1 TC Normal Operating Conditions

The calculation packages concerning the stress analysis for the TC are contained in several PNFS submittals because the Standardized TC derives from a design that was first evaluated by the NRC staff in the NUHOMS 24P Topical Report. (References 3, 4 and 40). There have been some additional analyses performed in conjunction with the TC for a site-specific application (Reference 41, calculation package BGE 001.0202 Revision 4), and the current submittal has two calculation packages NUH 004.0205 and NUH 004.0206 (References 42 and 43) which deal with the BWR collar and a new thermal stress analysis. All of these packages have been evaluated, and the summary tables in this SER reflect portions from all above analyses. It should also be noted that, compared with previously reviewed submittals, portions of drawings for the Standardized TC differ dimensionally and with respect to material options. NRC staff has taken the various combinations into consideration in this SER.

The TC was designed for: (1) dead weight loads, (2) design basis thermal loads, and (3) handling and transfer loads. Table 3.3.2-1 of this SER summarizes all the stress analysis results for normal operating conditions. The summary table shows stresses for each TC component for the three load conditions analyzed by PNFS and the corresponding stress as verified by NRC staff. Each stress intensity was compared to the allowable stress for the particular material at the operating temperature as required by the ASME Code for Service Levels A and B conditions. For TC parts which allow more than one material specification, the lower allowables are listed in Table 3.3.2-1 and all subsequent SER tables.

A. Deadweight Loads for the TC

The TC is evaluated for two dead weight loads, e.g., a fully loaded cask hanging vertically from its two lifting trunnions and a fully loaded cask supported horizontally from its trunnions at top and bottom ends of the TC on the transport skid.

Review of calculation package NUH 004.0206 (Reference 43) indicates that the dead weight loads are trivial when compared to the stress allowables. The results are broken out by orientation, i.e., vertical, horizontal or corner in Table B-1 of NUH 004.0206. All the calculations were made using ANSYS runs for the three orientations (75 g vertical, 75 g horizontal and 25 g corner drops) and then factoring the drop accelerations to obtain 1 g for dead weight.

B. Thermal loads for the TC

Calculation Package NUH 004.0206 presents the thermal stress analysis associated with the TC. The normal temperature range is considered to be 17.8°C to 37.8°C (0°F to 100°F) and the off-normal excursions go to -40°C (-40°F) and 52°C (125°F) for the 5-year-old PWR and BWR fuel assemblies. The TC has been analyzed for the combined effects of the worst case radial, axial, and circumferential thermal gradients. Tables 1, 2 and 3 in the calculation package show that the 5-year-old PWR fuel on a 21.1°C (70°F) ambient day causes the highest gradient for radial and axial surfaces, whereas the 10-year-old PWR and BWR fuel cause the highest circumferential gradient for 21.1°C (70°F) ambient day. For ambient temperatures above 37.8°C (100°F), use of a solar shade for any operations involving the TC is required, because the 37.8°C (100°F) ambient temperature with solar load is outside of the design envelope for the neutron shielding of the TC.

The effects of dissimilar materials has been accounted for in the analyses by modeling the material properties of all four structural and non-structural (shielding) materials.

Table 5 of the calculation package summarized the results of three ANSYS runs, the top end axial, bottom end axial, and circumferential thermal stresses. PNFS conservatively summed the thermal stress intensities for axial and circumferential orientation for the shell regardless of actual location in the shell. The staff notes that Table 8.1-10a in the SAR does not reflect the results of the latest thermal analyses as provided in the calculation package NUH 004.0206 (Reference 43). However, the results as given in the SER in the Table 3.3.2-1 and 3.3.2-2 do incorporate the appropriate material. Also the tables in the SER incorporate different material allowables to account for the least strong material which could be used according to the PNFS drawings.

C. Operational handling loads for TC

As described in the dead weight load section above, there are two normal operation handling cases for the TC: vertically supported by the crane, and horizontally supported by the skid. The former is governed by ANSI N14.6 rules (Reference 8) and the latter is governed by the ASME B&PV Code (Reference 9).

The ANSI code is concerned with critical lift loads and consequently only addresses the lifting trunnion design and the TC shell in the vicinity of the lifting trunnion. The evaluation of this aspect of normal handling is discussed in a subsequent section of 3.3.2 of this SER. The actual transportation and transfer handling cases which are considered are 1 g vertical, 1 g horizontal, 1 g axial, and $\pm 1/2$ g applied simultaneously in all three directions. Table 3.3.2-1 of this SER summarizes the results of stress analysis for the TC shell and top and bottom rings and cover plates. All results for the normal handling case are satisfactory for Service Level A. Loads acting on the upper and lower trunnions are discussed in a subsequent portion of this SER.

D. TC Load Combinations for Normal and Off-normal Conditions

Table 3.2-5b of the SAR defines the different load combination for normal and off-normal events. These conditions correspond to Service Levels A and B of the ASME B&PV Code. There are five Level A conditions and two Level B conditions. Table 3.3.2-2 of this SER has combined the conditions as follows. Load combination 1 combines the worst case of load cases A1, A2, A3, A4 and B1, and load case 2 combines the worst case of A1, A2, A3, A5, and B2. Note that thermal stresses are the same for all cases, i.e., 21.1°C (70°F) ambient day for 5-year-old PWR fuel for radial and axial gradients, and 10-year-old PWR fuel for circumferential gradients. Cases A1, A2, and A3 are all exceptionally low. Case A4 and B1 correspond to TC transport outside the fuel pool building, and A5 and B2 correspond to transfer of the DSC into/out of the HSM.

Calculation Package NUH 004.0205 (Reference 42)describes the structural evaluation of the BWR cask collar. PNFS evaluated transfer loads as well as accidental drop loads for the collar, bolts and welds. Of these loads, only the transfer loads relate to Service Levels A and B. In order to estimate the dead load and thermal load, PNFS argued that loads would be similar to those of the TC without the collar in the vicinity of the top structural ring. The NRC accepts this position based on similar geometry. In all cases, the allowable stresses were evaluated for a material temperature of 204°C (400°F), a conservative value. As shown in Table 3.2.2-2 all the stresses are lower than the allowables.

3.3.2.2 TC Accident Conditions

Section 8.2 of the SAR defines the accident conditions that affect the transfer cask. These conditions are: (1) earthquake, (2) accidental drop of the TC with the DSC inside, and (3) tornado wind loads. Tornado generated missiles, although not discussed in the SAR, was the subject of an NRC staff concern. It is addressed by PNFS and evaluated in this SER.

A. TC Seismic Condition

The design basis earthquake for the standardized NUHOMS system is 0.25 g peak horizontal ground acceleration and 0.17 g peak vertical ground acceleration. The SAR evaluates the effects of a seismic event on a loaded DSC inside the TC for two conditions. The first case postulated was for the TC in a vertical orientation in the decontamination area during closure of the DSC. For this case, the SAR shows that the loaded TC would not overturn during an earthquake, provided the loaded TC weighs 453.5 kg (190 kips) and experiences a horizontal acceleration not greater than 0.40 g. N.B. This 0.4 g horizontal acceleration is not ground acceleration, which is limited to 0.25 g; rather, it is the peak acceleration at some elevation above ground level, and could result from a ground acceleration of 0.25 g multiplied by an amplification factor.

The second case postulated in the SAR is for a seismic event occurring during the normal transport of the TC loaded on the trailer. The SAR stated that this case is enveloped by the handling case of ± 0.5 g acting in the vertical, axial, and transverse directions simultaneously. In Section 8.2.3 of the SAR the statement is made that the seismic stress intensities are to be taken as the normal transport stress intensities, because the accelerations for seismic are the same as those assumed for transport. NRC staff has included these stresses in Table 3.3.2-3. The individual stress intensities as well as the three load combination stress intensities are below ASME B&PV Code allowables.

B. Design Basis Tornado Wind Loads Acting on TC

The SAR shows that if the height to the top of the cask is 371 cm (146 inches), and the half wheel base of the transport vehicle is 168 cm (66 inches), there is safety factor of 1.5 against overturning when the TC is subjected to Design Basis Tornado (DBT) winds. Shell stresses were also evaluated and found to be 26,201 kPa (3.8 ksi), well below the 232,362 kPa (33.7 ksi) allowable for Service Level C. NRC staff concurs with the results for the DBT winds, provided the site-specific equipment, i.e., the trailer and the skid, correspond dimensionally with the example in the SAR.

C. Cask Drop Accident

This SER presents a detailed discussion of the cask drop accidents postulated by the SAR. This includes the basis for the selection of the parameters and the assumptions used for the ANSYS finite element models. All drop scenarios assume that the DSC is inside the TC. Thus all previous discussions about the drop apply equally to the DSC and the TC.

The ANSYS models predict that the stresses will exceed the yield stress for all major structural TC components except the top cover. The results in the columns entitled "NRC" in Table 3.3.2-4 are somewhat higher than those of the PNFS columns. This is due to the NRC staff conservatively selecting locations in the TC which may be more localized than locations selected by PNFS. However it is important to note that, in spite of this conservative process, the NRC staff results are still lower than allowable stress intensities. As discussed in the structural analysis of the DSC, any drop height higher than 38 cm (15 inches) shall require the retrieval and inspection of the DSC and its internals, in keeping with the guidelines of the ASME B&PV Code when using Service Level D allowables. Because the TC is also designed to ASME B&PV Code requirements, it will be necessary to inspect the TC as well, should it be subjected to a drop height higher than 38 cm (15 inches). Results are given in Table 3.3.2-4 of this SER. In all cases, the stresses are below the code allowables.

D. Tornado Generated Missiles

In docketed responses to NRC staff's questions for the NUHOMS-24P TR, PNFS presented results of an accident condition, namely design basis tornado (DBT) generated missiles. The two missiles considered are those suggested in NUREG-0800 (Reference 23), a 1,677 kg (3,697 pound) automobile, and a 125 kg (276 pound) 20 cm (eight-inch) diameter shell. TC stability, penetration resistance, and shell and end plate stresses were calculated and shown to be below the allowable stresses for Service Level D stresses. Although the SAR for the standardized NUHOMS system did not include specific reference to these loads, the NRC staff has included them. The NRC staff believes that there is no need to recalculate stresses for this accident case because identical shell, bottom plate material and thickness were used, and the identical tornado missiles were postulated for both versions of the TC. The top plate material for the standardized NUHOMS system is a higher strength material as noted in Table 3.3.2-5. The results from the TR are shown for completeness in Table 3.3.2-5.

E. TC Load Combination for Service Level D Accident Conditions

Table B-2 of the design calculation NUH 004.0206 (Reference 43) summarizes the load combination for the three accident cases postulated in the SAR. Three drop cases were considered (1) vertical drop, (2) corner drop, and (3) horizontal drop. In each drop case the dead weight loads were combined with the drop loads. Table 3.3.2-6 of this SER shows the results and the material allowables at 204°C (400°F) for the materials specified in the drawings. These allowables are somewhat lower than given in the SAR, but they represent the values for the specified materials for 204°C (400°F). In all cases the actual stress intensities are lower than the allowables. Thus the TC meets the ASME B&PV Code for Service Level D conditions.

3.3.2.3 TC Fatigue Evaluation

Section C.4.2 of the SAR for the standardized NUHOMS system presents an evaluation of the loading cycles of the TC to show that the six criteria associated with NC-3219.2 of the ASME B&PV Code are met. NRC staff evaluated Section C.4.2 and concurs that all six criteria are met.

3.3.2.4 TC Trunnion Loads and Stresses

The relevant design criteria for lifting a "critical load," i.e., the spent fuel loaded in the DSC inside the TC while in the fuel building, are covered by ANSI N14.6, 1987 (Reference 8) and NUREG-0612 (Reference 14). Critical loads, defined by N14.6, are loads "whose uncontrolled movement or release could adversely affect any safety-related system or could result in potential off-site exposures comparable to the guideline exposures outlined in 10 CFR Part 100." In the case of the transfer cask, the cask lifting trunnions shall be considered as special lifting devices for the DSC. Because its design does not provide a dual-load path, the design criteria require that load bearing members shall be designed with a safety factor of two times the normal stress design factor for handling the critical load. Thus, the load bearing members must be sized so that yield stresses are no more than one-sixth minimum tensile yield strength of the material or no more than one-tenth the minimum ultimate tensile strength of the material. An additional allowance for crane hoist motion loads is recommended by NUREG-0612. Although Reference 14 does not quantify the magnitude of this dynamic load, ANSI NOG-1-1983 (Reference 44) does specify 15%, which was used in the SAR. Therefore the allowance for dynamic loads is appropriate. Because the titling trunnions are used in tilting and horizontal transfer and transfer modes instead of lifting, the lower tilting trunnions are designed to ASME III Class 2 criteria.

Table 3.3.2-7 summarizes the results for the lifting trunnion assemblies, weld regions, and cask shell. This table presents summary results for the lifting and supporting trunnions that are designed in accordance with: (1) ANSI N14.6 for critical lift loads, and (2) ASME for horizontal support loads. The local stresses in the TC at the intersection of the trunnion sleeve and the shell stiffener insert are calculated by using finite element analyses which appear in reference 18. The summary Table 3.3.2-7 shows that all stresses are less than the allowable stresses for both the ANSI N14.6 critical lift load conditions and the ASME B&PV Code for on-site transfer.

Table 3.3.2-7 also shows the results for the lower tilting trunnion assembly. Comparisons between the PNFS values and NRC staff values are given. All stresses are below the ASME III allowable stress intensities for Class 2 components. This table has included the consequences of using various material options which are noted in the drawings for the standardized TC. For instance, the shell and trunnion sleeve materials have two options from which the fabricator may choose.

Table 3.1.2-1 HSM LOAD COMBINATION RESULTS

Section:	Floor Slab			Side Wall			Front Wall			Rear Wall		
Force:	Shear	Trans. Moment	Long. Moment	Shear	Trans. Moment	Long. Moment	Shear	Trans. Moment	Long. Moment	Shear	Trans. Moment	Long. Moment
Load Comb. 1	0.2	2.9	2.3	1.6	11.4	8.5	1.1	10.7	13.4	1.9	6.4	5.9
Load Comb. 3	7.5	154.0	117.0	20.7	231.0	185.0	6.4	494.0	293.0	12.3	118.0	127.0
Load Comb. 5	5.3	111.0	73.9	11.2	167.0	140.0	6.8	435.0	283.0	3.1	102.0	86.7
Load Comb. 6	5.1	110.0	67.0	13.7	152.0	127.0	3.0	308.0	193.0	4.4	105.0	78.7
Allowable	14.6	217.0	229.0	23.9	724.0	762.0	40.4	910.0	910.0	15.3	477.0	313.0
MOS	0.9	0.4	1.0	0.2	2.1	3.1	4.9	0.8	2.1	0.2	3.0	1.5
Load Comb. 7	3.8	146.0	155.0	8.4	340.0	340.0	5.3	691.0	574.0	1.5	366.0	142.0
Allowable	13.5	185.0	195.0	22.7	650.0	618.0	38.3	774.0	774.0	14.5	470.0	267.0
MOS	2.6	0.3	0.3	1.7	0.9	0.8	6.2	0.1	0.3	8.7	0.3	0.9

Section:	Roof Slab			End Shield Wall		Rear Shield Wall	
Force:	Shear	Trans. Moment	Long. Moment	Trans. Moment	Long. Moment	Trans. Moment	Long. Moment
Load Comb. 1	12.8	392.0	399.0	0	0	0	0
Load Comb. 3	27.1	755.0	928.0	114	60.4	172	172
Load Comb. 5	13.0	390.0	533.0	23	12.2	32	32
Load Comb. 6	15.3	445.0	596.0	92.5	49	167	167
Allowable	49.8	2,151.0	2,087.0	516	1,593	423	1,178
MOS	0.8	1.8	1.2	3.5	25.4	1.5	5.8
Load Comb. 7	20.0	618.0	1,035.0	0	0	0	0
Allowable	47.2	1,780.0	1,830.0				
MOS	1.4	1.9	0.8	n.a.	n.a.	n.a.	n.a.

shear in units of kips/ft.; transverse and longitudinal moments in units of in.-kips/ft.
MOS = Margin of Safety = (allow/calc)-1
n.a. not applicable

Load Comb 1 = 1.4 DL + 1.7 LL
Load Comb 3 = 0.75 (1.4 DL + 1.7 LL + 1.7 T + 1.7 W)
Load Comb 5 = DL + LL + T + E
Load Comb 6 = DL + LL + T + F
Load Comb 7 = DL + LL + Ta

Table 3.1.2-2 DSC Support Assembly Load Combination Results
Actual and Allowable Stress Values

COMPONENT Load Combination[a]	fa	Fa	fbx	Fbx	fby	Fby	AISC Sect 1.6[b]	fv	fv/Fv	Comments[c]
COLUMN										
Equation 1	2.43	16.63	1.65	20.99	0.38	20.99	.24	.10	.01	A
Equation 2	3.21	16.63	3.51	20.99	2.43	20.99	.33	.29	.02	A
Equation 3	5.88	16.63	6.04	20.99	3.75	20.99	.54	.38	.02	A
Equation 4	4.62	14.07	9.94	15.96	3.22	15.96	.72	.52	.03	A
Equation 5	4.43	18.66	1.13	23.76	3.68	23.76	.27	.19	.01	A
CROSS BEAM										
Equation 1	0.88	18.36	2.77	19.08	0.08	19.08	.20	1.95	.15	A
Equation 2	1.19	18.36	1.35	19.08	1.2	19.08	.13	5.33	.30	A
Equation 3	2.02	18.36	6.63	19.08	1.55	19.08	.34	4.28	.24	A
Equation 4	1.69	15.4	6.08	15.96	3.09	15.96	.40	3.36	.23	A
Equation 5	0.47	20.74	4.31	21.6	11.47	21.6	.44	3.89	.19	A
RAIL										
Equation 1	0.57	15.98	4.63	20.99	1.04	23.85	.30	1.07	.08	A
Equation 2	0.81	15.98	3.15	20.99	1.87	23.85	.19	3.25	.18	A
Equation 3	1.48	15.98	10.98	20.99	2.94	23.85	.46	2.15	.12	A
Equation 4	0.51	13.57	9.09	17.56	3.38	19.95	.43	2.17	.15	A
Equation 5	3.93	17.89	12.2	23.76	8.38	27	.61	2.1	.10	A

Table 3.1.2-2 DSC Support Assembly Load Combination Results (Continued) Actual and Allowable Stress Values

COMPONENT Load Combination[a]	fa	Fa	fbx	Fbx	fby	Fby	AISC Sect 1.6[b]	fv	fv/Fv	Comments[c]
TIE BEAM										
Equation 1	0.07	18.15	2.53	20.99	0.62	20.99	.15	.25	.02	A
Equation 2	1.27	18.15	4.47	20.99	5.32	20.99	.36	1.08	.06	A
Equation 3	4.92	18.15	13.63	20.99	11.11	20.99	.91	6.09	.34	A
Equation 4	3.52	15.24	12.58	27.56	8.04	17.56	.83	1.22	.08	A
Equation 5	1.23	20.5	0.82	23.76	3.47	23.76	.14	.09	.01	A

(a) Refer to Table 2.6 for definition of Equations 1 through 5
(b) Combined axial and bending stresses as a fraction of the allowable stress per AISC 8th ed., Section 1.6.1
(c) A = Allowable, U = Unallowable

Table 3.2.2-1 DSC Stress Analysis Results For Normal Loads
Service Level A

Stress (ksi)

DSC Component	Stress Type	Dead Weight		10.0 psig Int. Pressure		100°F Thermal		Normal Handling		Allowable* Level A & B
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	
DSC Shell	Pri Memb	1.1	0.6P	0.5	0.5	N/A	N/A	3.3	3.3	17.5
	Memb + Bend	4.7	6.2	4.5	5.4	N/A	N/A	9.6	9.6	26.3
	Pri + Second	11.0	18.4	7.7	7.8	12.6	17.3	15.0	15.0	52.5
Outer Top Cover Plate	Pri Memb	0.1	0.1	0.4	0.9	N/A	N/A	0.3	0.3	17.5
	Memb + Bend	0.1	0.1	4.2	5.6	N/A	N/A	0.3	0.3	26.3
	Pri + Second	0.1	0.1	13.8	14.0	0.2	0.2	0.3	0.3	52.5
Inner Top Plate	Pri Memb	0.1	0.1	0.4	0.7	N/A	N/A	0.3	0.3	17.5
	Memb + Bend	0.1	0.1	2.5	2.9	N/A	N/A	0.3	0.3	26.3
	Pri + Second	0.1	0.1	13.1	13.2	0.2	3.2	0.3	0.3	52.5
Outer Bottom Cover Plate	Pri Memb	0.1	0.1	0.2	1.2	N/A	N/A	2.0	2.3	17.5
	Memb + Bend	0.1	0.1	3.3	4.2	N/A	N/A	10.3	10.3	26.3
	Pri + Second	0.1	0.1	4.4	4.4	0.3	0.6	13.4	13.4	52.5
24-P Spacer Disc	Pri Memb	0.9	0.1	N/A	N/A	N/A	N/A	2.7	2.7	20.5
	Memb + Bend	1.5	1.6			N/A	N/A	4.5	4.5	30.8
	Pri + Second	2.2	2.2			26.5	32.7	6.6	6.6	61.5
52-B Spacer Disc	Pri Memb	1.0	1.0	N/A	N/A	N/A	N/A	1.5	3.0	20.5
	Memb + Bend	1.7	1.8			N/A	N/A	3.1	5.1	30.8
	Pri + Second	3.1	3.1			38.5	42.9	5.6	9.3	61.5
Inner Bottom Plate	Pri Memb	0.1	0.1	0.2	0.3	N/A	N/A	0.3	0.3	17.5
	Memb + Bend	0.1	0.1	0.5	0.5	N/A	N/A	0.3	0.3	26.3
	Pri + Second	0.2	0.2	0.5	0.5	2.3	7.6	0.7	0.6	52.5
Support Rods	Pri Memb	0.4	0.4	N/A	N/A	N/A	N/A	0.4	0.4	19.3
	Memb + Bend	0.8	0.76			N/A	N/A	0.9	0.9	29.0
	Pri + Second	0.8	0.76			~0	~0	0.9	0.9	57.9

* Allowable stress for Service Levels A and B

Primary Membrane	Sm = 17.5 ksi	Sm = 20.5 ksi	Sm = 19.3 ksi
Primary Memb + Bend	1.5 x Sm = 26.3	1.5 x Sm = 30.8	1.5 x Sm = 29.0
Primary + Secondary	3 x Sm = 52.5	3 x Sm = 61.5	3 x Sm = 57.9
Shell, Disc and end plates for 500°F	SA 240 Type 304	SA 516	SA 36

** Allowable stress for 300°F for spacer disc

3.0 Sm = 60.0

Table 3.2.2-2 DSC Stress Analysis Results For Off-Normal Loads Service Level B

Stress (ksi)

DSC Component	Stress Type	Internal Pressure 10.0 psig		Thermal 100°F		Off-Normal Handling		Allowable*
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	
DSC Shell	Pri Memb	0.5	0.5	N/A	N/A	0.6	3.2	17.5
	Memb + Bend	4.5	5.4	N/A	N/A	9.6	13.2	26.3
	Pri + Second	7.7	7.8	12.6	17.3	30.	16.3	52.5
Outer Top Cover Plate	Pri Memb	0.4	0.9	N/A	N/A	0.1	0.1	17.5
	Memb + Bend	4.2	5.6	N/A	N/A	0.1	0.1	26.3
	Pri + Second	13.8	14.0	0.2	0.2	0.6	0.6	52.5
Inner Top Plate	Pri Memb	0.4	0.7	N/A	N/A	0.	0.	17.5
	Memb + Bend	2.5	2.9	N/A	N/A	0.	0.	26.3
	Pri + Second	13.1	13.2	0.2	3.2	0.8	0.8	52.5
Outer Bottom Cover Plate	Pri Memb	0.2	1.2	N/A	N/A	4.0	4.7	17.5
	Memb + Bend	3.3	4.2	N/A	N/A	20.6	20.6	26.3
	Pri + Second	4.4	4.4	0.3	0.6	26.8	26.8	52.5
24-P Spacer Disc	Pri Memb	N/A	N/A	N/A	N/A	0	—	20.5
	Memb + Bend			N/A	N/A	0	—	30.8
	Pri + Second			26.5	32.7	0	—	61.5
52-B Spacer Disc	Primary Memb	N/A	N/A	N/A	N/A	0	—	20.5
	Memb + Bend			N/A	N/A	0	—	30.8
	Pri + Second			38.5	42.9	0	—	61.5
Inner Bottom Plate	Pri Memb	0.2	0.3	N/A	N/A	0.	0.	17.5
	Memb + Bend	0.5	0.5	N/A	N/A	0.3	0.3	26.3
	Pri + Second	0.5	0.5	2.3	7.6	1.3	1.3	52.5
Support Rods	Pri Memb	N/A	N/A	N/A	N/A	0	—	19.3
	Memb + Bend			N/A	N/A	0	—	29.0
	Pri + Second			~0	~0	0	—	57.9

* Allowable stress is taken for Service Level B for SA 240 Type 304, SA 516 and SA 36 material at 500°F.

Table 3.2.2-3 DSC Load Combinations For Normal and Off-Normal Operating Conditions Service Levels A and B

Stress (ksi)

DSC Component	Stress Type	Case A2		Case[1] A3/A4		Case[2] B2		Allowable* Level A & B
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	
DSC Shell	Pri Memb	1.0	1.1	4.3	4.4	1.6	4.4	17.5
	Memb + Bend	9.2	11.6	18.8	21.2	18.8	24.8	26.3
	Pri + Second	31.3	43.5	46.3	51.1[3]	50.5	52.4[3]	52.5
Outer Top Cover Plate	Pri Memb	0.5	1.0	0.8	1.3	0.6	1.1	17.5
	Memb + Bend	4.3	5.7	4.6	6.0	4.4	5.8	26.3
	Pri + Second	14.4	14.3	14.4	14.6	14.7	14.9	52.5
Inner Top Cover Plate	Pri Memb	0.5	0.8	0.8	1.1	0.5	0.8	17.5
	Memb + Bend	2.6	3.0	2.9	3.3	2.6	3.0	26.3
	Pri + Second	13.4	16.5	13.8	16.8	14.2	17.3	52.5
Outer Bottom Cover Plate	Pri Memb	0.3	1.3	2.3	3.6	4.3	6.0	17.5
	Memb + Bend	3.4	4.3	13.7	14.6	24.0	24.9	26.3
	Pri + Second	4.8	5.1	18.2	18.5	31.6	31.9	52.5
Inner Bottom Plate	Pri Memb	0.3	0.4	0.6	0.7	0.3	0.4	17.5
	Memb + Bend	0.6	0.6	0.9	0.9	0.9	0.9	26.3
	Pri + Second	2.9	8.3	3.6	8.9	4.2	9.6	52.5
24-P Spacer Disc	Pri Memb	0.9	0.9	3.6	3.6	0.9	0.9	20.5
	Memb + Bend	1.5	1.6	6.0	6.1	1.5	1.6	30.8
	Pri + Second	28.7	34.9	35.3	41.5	28.7	34.9	61.5
Support Rods	Pri Memb	0.4	0.4	0.8	0.8	0.4	0.4	19.3
	Memb + Bend	0.8	0.8	1.7	1.7	0.8	0.8	29.0
	Pri + Second	0.8	0.8	1.7	1.7	0.8	0.8	57.9
52-B Spacer Disc	Pri Memb	1.0	1.0	2.5	4.0	1.0	1.0	20.5
	Memb + Bend	1.7	1.8	4.8	6.9	1.7	1.8	30.8
	Pri + Second	41.6	46.0	47.2	55.3	41.6	46.0	61.5

Stress intensities conservatively combined irrespective of location unless otherwise noted.

*Allowable stress is taken for Service Levels A and B for SA 240 Type 304 Material at 500°F.

1. Load cases A3 and A4 are combined into one case because the stresses for the normal and off-normal pressure cases are the same.

2. Load cases B1, B2, B3, and B4 are combined into one case because the pressure for normal and off-normal conditions are the same and all thermal loads are bounded by the 100°F case.

3. The maximum stress intensity in the DSC shell for this load combination was found to be near the bottom of the shell, however the max S.I. of 18.4 ksi for the dead weight case is in the shell near a spacer disc. Consequently a lower DW stress value near the bottom of the shell was used for the combined stress.

Table 3.2.2-4 DSC Stress Analysis Results For Accident Conditions Service Level C[1]

Stress (ksi)

DSC Component	Stress Type	Seismic		Accident[2] Pressure (50.3 psig)		Flood (21.7 psi)		Accident Handling		Allowable*	
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	@500°F	@580°F
DSC Shell	Pri Memb	1.8	1.7	2.7	2.7	1.2	1.1	0.6	3.2	21.	19.9
	Memb + Bend	18.2	14.3	22.6	16.8	1.2	1.1	9.6	18.6	29.1	27.6
Outer Top Cover Plate	Pri Memb	0.4	0.5	1.9	4.4	0.1	18.0	0.1	0.1	21.	19.9
	Memb + Bend	0.4	0.5	20.9	26.9	0.2	18.7	0.1	0.1	29.1	27.6
	Weld	—	—	—	—	—	—	—	0.5	—	—
Inner Top Plate	Pri Memb	0.4	0.5	2.1	3.5	0.2	0	0.	—	21.	19.9
	Memb + Bend	0.4	0.5	12.5	14.2	0.2	0	0.	—	29.1	27.6
Outer Bottom Cover Plate	Pri Memb	0.4	0.5	0.9	6.2	0.1	9.6	4.0	4.7	21.	19.9
	Memb + Bend	0.4	0.5	16.5	20.8	0.4	9.6	20.6	23.6	29.1	27.6
	Weld	—	—	—	—	—	—	—	12.4	—	—
24-P Spacer Disc	Pri Memb	3.2	2.9	N/A	N/A	0	0	0	—	30.7	28.6
	Memb + Bend	5.2	5.3	N/A	N/A	0	0	0	—	36.9	34.3
52B Spacer Disc	Pri Memb	4.0	3.8	N/A	N/A	0	0	0	—	30.7	28.6
	Memb + Bend	5.6	6.7	N/A	N/A	0	0	0	—	36.9	34.3
Inner Bottom Plate	Pri Memb	0.4	0.5	1.0	1.0	0.1	0	0	—	21.	19.9
	Memb + Bend	0.4	0.5	2.5	2.5	0.4	0	0.3	0.3	29.1	27.6
Support Rods	Pri Memb	0	0	N/A	N/A	0	0	0	—	23.2	21.6
	Memb + Bend	0.2	0.2	N/A	N/A	0	0	0	—	34.7	32.4

* Allowable stress for Service Level C @ 500°F for all cases except accident pressure.

P_m larger of 1.2 Sm or Sy = 21.0 for SA 240 Type 304
$P_L + P_B$ = smaller of 1.8 Sm or 1.5 Sy = 29.1 for SA 240 Type 304

1. No secondary stress needs to be evaluated according to ASME Code for Service Level C. This includes thermal as well as secondary bending stresses for pressure cases.

2. Accident pressure applied to inner cover plates, also allowable stresses for this condition should be based on 580°F.

Table 3.2.2-5 DSC Load Combinations for Accident Service Level C Cases[1]

Stress (ksi)

DSC Component	Stress Type	Case[2] C1		Case[3] C2		Case[4] C3/C4/C5/C6/C7		Allowable[5]	
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	@500°F	@580°F
DSC Shell	Pri Memb Memb + Bend	4.4 24.9	5.0 23.2[6]	4.4 22.7	2.2 12.7	3.8 22.0	6.5 25.8[7]	21. 29.1	19.9 27.6
Outer Top Cover Plate	Pri Memb Memb + Bend	2.4 21.4	5.0 27.5	2.1 21.2	19.0 24.4	2.1 21.1	4.6 27.1	21. 29.1	19.9 27.6
Inner Top Plate	Pri Memb Memb + Bend	2.6 13.0	4.1 14.8	2.4 12.8	0.8 3.0	2.2 12.6	3.6 14.3	21. 29.1	19.9 27.6
Outer Bottom Cover Plate	Pri Memb Memb + Bend	0.5 0.9	6.8 21.4	0.2 0.9	10.9 13.9	4.1 21.1	11.0 24.3[8]	21. 29.1	19.9 27.6
24-P Spacer Disc	Pri Memb Memb + Bend	4.1 6.7	3.8 6.9	0.9 1.5	0.9 1.6	0.9 1.5	0.9 1.6	30.7 36.9	28.6 34.3
52-B Spacer Disc	Pri Memb Memb + Bend	5.0 7.3	4.8 8.5	1.0 1.7	1.0 1.8	1.0 1.7	1.0 1.8	30.7 36.9	28.6 34.3
Inner Bottom Plate	Pri Memb Memb + Bend	1.5 3.0	1.6 6.1	1.5 3.0	0.4 0.5	1.1 2.9	1.1 2.9	21. 29.1	19.9 27.6
Support Rods	Pri Memb Memb + Bend	0.4 1.0	0.4 1.0	0.4 0.8	0.4 0.8	0.4 0.8	0.4 0.8	23.2 34.7	21.6 32.4

Stress Intensities conservatively combined irrespective of location unless otherwise noted.

1. Secondary stresses are not required for Service Level C.
2. Seismic stresses are considered "mechanical loads" and must be combined with dead weight and accident pressure for C1.

Table 3.2.2-5 DSC Load Combinations for Accident Service Level C Cases (Continued)

3. Case C2 is dead weight, normal pressure and flooding.
4. Because thermal stresses need not be evaluated for Service Level C, cases C3 through C6 are bounded by C7 and consist of dead weight, accident pressure, and accident handling.
5. Allowables are based on a maximum DSC temperature of 500°F, except for the accident pressure condition for C3-C7 which has a maximum DSC temperature of 579°F.
6. The maximum stress intensity in the DSC shell for this load combination was found to be near a spacer disc, where the pressure stress is only 2.7 ksi. Thus the 26.9 ksi stress due to pressure shown in Table 3.2.2-4 is not used because it occurs near the end of the shell.
7. The maximum stress intensity in the DSC shell for this load condition is at the bottom end of the shell and is due primarily to the accident loading case. The dead weight stress = 6.2 ksi, and the accident pressure stress = 1.0 at node 515.
8. The maximum stress intensity in the outer bottom plate for this load condition is near the grapple connection and is due primarily to the accident handling case. The accident pressure at this location is 0.6 ksi.

Table 3.2.2-6 DSC Drop and Internal Pressure Accident Loads Service Level D

Stress (ksi)

DSC Component	Stress Type	Vertical (75g)		Horizontal[2] (75g)		Accident[3] Pressure (50.3 psig)		Allowable[1]	
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	@500°F	@580°F
DSC Shell	Pri Memb Memb + Bend	10.2 27.1	12.7 27.1	26.1 40.4	32.9 50.7	2.7 22.6	2.7 26.9	42.0 63.0	39.8 59.8
Inner Top Plate	Pri Memb Memb + Bend	6.2 10.2	10.2 10.2	10.3 10.3	10.3 11.3	2.1 12.5	3.5 14.2	42.0 63.0	39.8 59.8
Outer Top Cover Plate	Pri Memb Memb + Bend	1.7 3.2	3.2 3.8	10.3 10.3	10.3 11.3	1.9 20.9	4.4 26.9	42.0 63.0	39.8 59.8
Outer Bottom Cover Plate	Pri Memb Memb + Bend	3.9 4.5	3.9 4.6	10.3 10.3	10.3 11.3	0.9 16.5	6.2 20.8	42.0 63.0	39.8 59.8
Inner Bottom Plate	Pri Memb Memb + Bend	3.3 13.5	3.6 17.9	10.3 10.3	10.3 11.3	1.0 2.5	1.0 2.5	42.0 63.0	39.8 59.8
52-B (bound) Spacer Disc	Pri Memb Memb + Bend	32.5 60.5	32.5 60.5	48. 65.	48. 67.5	N/A N/A	N/A N/A	49.0 70.0	49.0 70.0
Support Rods (24-P bound)	Primary Memb + Bend	32.4 57.3	33.2 57.27	0.3 7.2	3.0 7.0	N/A	N/A	40.6 58.0	40.6 58.0
Top End Struct. Weld*	Primary Pri + Bend	— 4.7	— 6.6	— —	— 11.3	— 19.7	— 25.5	25.2 37.8	23.9 35.9
Bottom End Struct. Weld*	Primary Pri + Bend	— 6.3	— 7.5	— —	— 11.3	— 10.0	— 12.6	25.2 37.8	23.9 35.9

1. Allowables taken at worst cast temperature, i.e., for Case D1, T=500°F shell temperature, except accident pressure.
2. These columns for stresses for shell, top covers and bottom cover are taken from NUH004.0202 and the SAR for the standardized NUHOMS.
* Efficiency factor for Class C, Type 3 non-volumetric inspected welds = 0.6.
3. Accident pressure load applied to outer cover plates, also allowable stress for this condition should be based on 580°F.

Table 3.2.2-7 DSC Enveloping Load Combination Results for Accident Loads Service Level D

Stress (ksi)

DSC Component	Stress Type	Case D2 DW + To + Pa + FD		Case D4 DW + To + Pa + Lo		Allowable*
		PNFSI	NRC	PNFSI	NRC	500°F
DSC Shell	Pri Memb Memb + Bend	29.3 47.8	36.2 59.6[1]	3.8 36.9	6.5 41.6	42.0 63.0
Outer Top Cover Plate	Pri Memb Memb + Bend	12.3 31.3	14.8 38.3	2.1 21.1	4.6 27.1	42.0 63.0
Inner Top Plate	Pri Memb Memb + Bend	12.5 22.9	13.9 25.6	2.2 12.6	3.6 14.3	42.0 63.0
Outer Bottom Cover Plate	Pri Memb Memb + Bend	11.3 19.5	16.6 32.2	4.1 10.1	10.9 44.5	42.0 63.0
Inner Bottom Plate	Primary Memb + Bend	11.4 12.9	11.4 13.9	1.1 2.9	1.1 2.9	42.0 63.0
52-B (bound) Spacer Disc	Pri Memb Memb + Bend	49.0 66.7	49.0 69.3	1.0 1.7	1.0 1.8	49.0 70.0
Support Rods	Primary Memb + Bend	32.8 58.0	33.6 58.0	0.4 0.8	0.4 0.8	40.6 58.0
Top End** Struct. Weld	Primary Pri + Bend	--- 30.4	--- 36.9	--- 24.9	--- 26.1	25.2 37.8
Bottom End** Struct. Weld	Primary Pri + Bend	--- 19.7	--- 24.0	--- 23.8	--- 25.1	25.2 37.8

Stress intensities conservatively combined irrespective of location unless otherwise noted.

* Allowables are based on a maximum DSC temperature of 500°F.

** Efficiency factor for Class C, Type 3 non-volumetric inspected welds = 0.6.

1. The maximum stress intensity in the DSC shell for this load combination was found in the shell near the support rail for the 18.5° drop case. The accident pressure stress at this location is 2.7 ksi.

Table 3.3.2-1 Transfer Cask Stress Analysis Results for Normal Loads Service Levels A and B Allowables

Stress (ksi)

Cask Component	Stress Type	Dead Weight PNFSI	Dead Weight NRC	Thermal** PNFSI	Thermal** NRC	Normal[1] Handling PNFSI	Normal[1] Handling NRC	Allowable*
Cask Shell	Pri Memb	0.7	0.7	N/A	N/A	0.5	0.5	18.7
	Memb + Bend	0.9	0.8	N/A	N/A	4.1	4.1	28.1
	Pri + Second	0.9	0.8	20.3	46.2 (top) 19.9 (bottom)	42.6	4.1	56.1
Top Cover Plate	Pri Memb	0.2	0.9	N/A	N/A	-	-	21.7
	Memb + Bend	0.6	1.0	N/A	N/A	6.3	6.3	32.6
	Pri + Second	0.6	1.0	11.7	11.7	6.3	6.3	65.1
Bottom Cover Plate	Pri Memb	0.2	0.8	N/A	N/A	-	-	18.7
	Memb + Bend	1.3	0.8	N/A	N/A	14.2	14.2	28.1
	Pri + Second	1.4	1.4	5.3	19.4	14.2	14.2	56.1
Top Ring	Pri Memb	0.2	0.8	N/A	N/A	-	1.5	20.3
	Memb + Bend	0.2	0.8	N/A	N/A	-	1.5	30.5
	Pri + Second	0.5	0.8	8.4	23.6	-	6.4	61.0
Bottom Ring	Pri Memb	0.4	0.4	N/A	N/A	-	4.6	20.3
	Memb + Bend	0.4	0.6	N/A	N/A	-	4.6	30.5
	Pri + Second	0.6	0.6	17.4	22.8	-	26.9	61.0
Transfer Cask Collar for BWR DSC	Pri Memb	0.2	0.8	N/A	N/A	0	1.5	20.3
	Memb + Bend	0.2	0.8	N/A	N/A	0	1.5	30.5
	Pri + Second	0.5	0.8	11.8	23.6	0	6.4	61.0

* Allowables taken at 400°F
** Thermal stresses are considered secondary stresses only

1. The PNFSI shell stress reported for handling is located at the trunnion, whereas the NRC stress is located in the middle of the shell where bending would be maximum.

Table 3.3.2-2 Transfer Cask Load Combinations for Normal Operating Conditions Service Levels A and B

Stress (ksi)

Cask Component	Stress Type	Load Comb 1 A1-A4, B1 PNFSI	Load Comb 1 A1-A4, B1 NRC	Load Comb 2 A1-A3, A5, B2 PNFSI	Load Comb 2 A1-A3, A5, B2 NRC	Allowable*
Cask Shell	Pri Memb	1.2	0.7	0.6	1.2	18.7
	Memb + Bend	1.4	0.8	4.2	4.9	28.1
	Pri + Second	63.4[1]	47.1	62.7[1]	51.1	56.1
Top Cover Plate	Pri Memb	0.2	0.9	0.2	0.9	21.7
	Memb + Bend	0.6	0.9	6.9	7.3	32.6
	Pri + Second	18.6	19.0	11.9	12.9	65.1
Bottom Cover Plate	Pri Memb	0.2	0.8	0.2	0.8	18.7
	Memb + Bend	9.9	1.3	8.8	15.0	28.1
	Pri + Second	14.4	21.0	13.2	35.0	56.1
Top Ring	Pri Memb	0.2	0.2	0.2	2.3	20.3
	Memb + Bend	0.2	0.2	0.2	2.3	30.5
	Pri + Second	8.6	24.6	8.6	30.8	61.0
Bottom Ring	Pri Memb	0.4	0.4	0.4	5.0	20.3
	Memb + Bend	0.4	0.4	0.4	5.2	30.5
	Pri + Second	18.0	23.6	17.6	50.3	61.0
Transfer Cask Collar for BWR DSC	Pri Memb	-	2.3	-	-	20.3
	Memb + Bend	-	2.3	-	-	30.5
	Pri + Second	-	30.8	-	-	61.0

* Allowables taken at 400°F

1. This stress is reported by PNFSI in Table B-2 of NUH 004.0206. In that table an allowable stress for primary plus secondary stress was taken as 70 ksi. However, the NRC staff has taken lower allowable stresses based on worst case materials which may be used according to drawing NUH-03-8001. The NRC staff has consequently summed stresses at a point, rather than the more conservative approach of summing stresses regardless of location.

Table 3.3.2-3 Transfer Cask Stress Analysis Results for Accident Loads Service Level C** Allowables

Stress (ksi)

Cask Component	Stress Type	Handling		Seismic		DBT Wind	Load Combination C2***		Allowables*
		PNFSI	NRC	PNFSI	NRC		PNFSI	NRC	
Cask Shell	Pri Memb	0.5	0.5	0.5	0.5	-	1.7	1.7	22.4
	Memb + Bend	4.1	4.1	4.1	4.1	3.8	5.4	9.0	33.7
Top Cover Plate	Pri Memb	-	-	-	-	-	0.2	-	26.0
	Memb + Bend	6.3	3.2	6.3	3.2	0.5	13.2	7.4	39.1
Bottom Cover Plate	Pri Memb	-	-	-	-	-	0.1	0.8	22.4
	Memb + Bend	14.2	14.4	14.2	14.4	0.5	28.6	29.2	33.7
Top Ring	Pri Memb	-	1.5	-	1.5	-	0.1	3.8	24.4
	Memb + Bend	-	1.5	-	1.5	-	0.1	3.8	36.5
Bottom Ring	Pri Memb	-	4.6	-	4.6	-	0.1	9.6	24.4
	Memb + Bend	-	4.6	-	4.6	-	0.1	9.6	36.5
Transfer Cask Collar for BWR DSC	Pri Memb	-	1.5	-	1.5	-	-	3.8	24.4
	Memb & Bend	-	1.5	-	1.5	-	-	3.8	36.5

* Allowables taken at 400°F

** No secondary stresses need to be evaluated according to the ASME Code for Service Level C.

*** The C2 load combination includes deadweight, seismic, and handling loads.

Table 3.3.2-4 Transfer Cask Drop Accident Loads Service Level D Allowables

Stress (ksi)

Cask Component	Stress Type	Vertical Top Drop		Vertical Bottom Drop		Horizontal Drop with DW		Corner Top		Corner Bottom		Allowables*
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	
Cask Shell	Pri Memb	9.6	30.1	8.7	8.7	3.8	21.9	8.3	7.6	4.6	8.5	44.9
	Memb + Bend	10.2	33.6	8.7	8.7	15.5	22.7	13.9	7.6	13.9	11.3	64.4
Top Ring	Pri Memb	25.2	24.2	-	-	12.2	17.3	2.1	7.5	-	-	48.7
	Memb + Bend	25.2	46.4	-	-	12.2	22.6	2.9	12.6	-	-	73.1
Top 3" Cover	Pri Memb	24.2	24.2	-	-	5.8	7.7	2.7	11.7	-	-	49.0
	Memb + Bend	24.2	24.2	3.7	3.7	5.8	8.0	14.1	14.1	-	-	70.0
Bottom 2" Cover	Pri Memb	-	-	22.9	22.9	5.8	6.4	-	-	-	33.1	44.9
	Memb + Bend	14.4	14.4	22.9	22.9	5.8	11.6	-	-	33.1	33.1	64.4
Bottom Ring	Pri Memb	-	-	14.0	26.7	12.2	12.2	-	-	9.7	10.7	48.7
	Memb + Bend	-	-	14.0	26.7	12.2	25.9	-	-	9.7	33.9	73.1
Transfer Cask Collar for BWR DSC	Pri Memb	13.0	13.0	-	-	12.2	17.3	-	-	-	-	48.7
	Memb + Bend	25.2	46.4	-	-	12.2	22.6	-	-	-	-	73.1
Bolts for Top Cover	Ave. Tension	-	-	-	-	-	-	27.1	29.7	-	-	77.0
Bolts for Collar	Ave. Tension	0	0	0	0	-	-	-	-	74.3	74.3	153.0
	Shear	0	0	0	0	56.9	56.9	-	-	39.6	39.6	64.3

* Allowables taken at 400°F

Table 3.3.2-5 Transfer Cask Stress Results for Tornado Driven Missile Impact
Stress (ksi)

Cask Component	Stress Type	Massive Missile	Pen. Resist Missile	Allowable*
Cask Shell	Pri Memb	6.4	4.9	44.9
	Pri + Bend	20.5	30.3	64.4
Top Cover	Pri Memb	0	0	49.0
	Pri + Bend	19.7	13.2	70.0
Bottom Cover	Pri Memb	0	0	44.9
	Pri + Bend	17.5	22.2	64.4

* Allowable stresses based on Service Level D Allowables at 400°F

Table 3.3.2-6 Transfer Cask Load Combinations for Accident Conditions Service Level D

Stress (ksi)

Cask Component	Stress Type	Case D1 (Vert)		Case D2 (Corner)		Case D3 (Horiz)		Allowable*
		PNFSI	NRC	PNFSI	NRC	PNFSI	NRC	
Cask Shell	Pri Memb	9.7	30.8	4.7	9.2	3.9	22.6	44.9
	Memb + Bend	10.3	34.4	14.3	12.1	15.6	23.5	64.4
Top Ring	Pri Memb	25.4	25.0	2.2	8.3	12.3	18.1	48.7
	Memb + Bend	25.4	47.2	3.0	13.4	12.3	23.4	73.1
Top Cover	Pri Memb	24.4	25.1	2.9	12.6	5.9	8.6	49.0
	Memb + Bend	24.4	25.2	14.7	15.1	6.0	9.0	70.0
Bottom Ring	Pri Memb	14.1	27.1	10.1	11.1	12.3	12.6	48.7
	Memb + Bend	14.1	27.3	10.1	34.5	12.3	26.5	73.1
Bottom Cover	Pri Memb	23.1	23.7	0	33.9	5.9	7.2	44.9
	Memb + Bend	23.1	23.7	34.4	33.9	6.0	12.4	64.4

* Service Level D Allowables at 400°F

Table 3.3.2-7 Summary of Stress Analyses for Upper Lifting Trunnions and Lower Resting Trunnions, Weld Regions and Cask Shell

Component Location	Critical Handling Loads (per ANSI N14.6) Stress Intensity (ksi)	Critical Handling Loads (per ANSI N14.6) Allowable (ksi)	On-Site Transportation Loads (per ASME III Class 2) Stress Intensity (ksi)	On-Site Transportation Loads (per ASME III Class 2) Allowable (ksi)*
	Section			
Upper Trunnion (lift pin)	A-A 5.9	13.1	N/A	N/A
(support pin)	B-B 10.0	13.1	N/A	N/A
Upper Trunnion Sleeve (ferritic material)	C-C 6.3	9.0	5.9	45.0
Shell at Sleeve	5.0	5.4	27.7	45.0
Weld Sleeve/Trunnion (Upper Trunnion)	Plane		Plane	
	1 6.9	9.0	1 6.4	45.0
	2 7.7	9.0	2 8.3	45.0
Weld Sleeve/Insert (Upper Trunnion)	Plane		Plane	
	1 5.0	9.0	1 6.0	45.0
	2 5.5	7.0	2 5.4	7.0
	3 4.4	5.4	3 4.2	32.6
Lower Trunnion	N/A	N/A	4.1	28.1
Lower Trunnion Sleeve (304 material)	N/A	N/A	5.6	28.1
Shell at Sleeve			26.9	28.1
Weld Sleeve/Trunnion (Lower Trunnion)	N/A	N/A	Plane	
			1 5.6	28.1
			2 7.3	28.1
			3 5.7	28.1
Weld Sleeve/Cask (Lower Trunnion)	N/A	N/A	Plane	
			2 6.0	28.1

* Material allowables are taken at 400°F stresses and materials have been conservatively combined so that worst case for material options is recorded.

4.0 THERMAL EVALUATION

Introduction

This section evaluates the thermal hydraulic aspects of the designs for the HSM, DSC, and TC. The designs are evaluated against design criteria as presented in the SAR or otherwise determined to be acceptable. Below is a description of the thermal hydraulic review which was made followed by the actual evaluation.

Description of Review

Two similar standarized NUHOMS system designs have previously been reviewed, the 7-P design (Reference 45) and the 24-P design (Reference 40). Safety evaluation reports have been issued for both of these designs (Reference 46 and 39, respectively). The standardized NUHOMS design, which is the subject of this SER, differs from the 24-P thermal design in several respects.

The standardized NUHOMS system design has increased maximum heat load capacity (1 kilowatt per PWR assembly) compared to the 24-P design (0.66 kilowatt per PWR assembly). This increased heat load capacity was achieved by redesign of the air flow passages (larger flow area, but less height difference from the bottom of the DSC to the air outlet), increased fuel temperature limits for normal operation, and more realistic thermal calculations. The design has also been extended to include the capability to store 52 BWR spent fuel assemblies, referred to as the 52-B design.

In light of the two previous reviews of similar designs, this review focused on the differences from the previously approved designs. The review addressed the capability of the standardized NUHOMS system design to maintain fuel cladding temperatures and concrete temperatures within the acceptance criteria during normal, off-normal, and accident conditions, and also on the correctness of thermal gradients determined for use in the structural analysis.

Limiting temperature gradients used for structural and confinement integrity evaluations have been reviewed and found to be determined in an acceptable manner.

In view of the differences in thermal hydraulic characteristics between the standardized NUHOMS system design and the previously approved designs, the staff considered it prudent to require a thermal performance verification for the first standardized NUHOMS system to be used. The discussion for the thermal performance verification is included in Section 12.1.7.

Applicable Parts of 10 CFR Part 72

10 CFR 72.236(f) requires the cask design to have adequate heat removal capacity without active cooling systems. The staff considered Subpart F criteria as well. Section 72.122(h) provides that the fuel cladding should be protected against degradation and gross rupture. Section 72.122(b) states that structures, systems, and components important to safety should be designed to accommodate the effects of, and be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing; and to withstand postulated accidents. Section 72.126(a) provides that radioactive waste storage and handling systems should be designed with a heat-removal capability having testability and reliability consistent with their importance to safety. Also, Section 72.122(f) states that systems and components that are important to safety should be designed to permit inspection, maintenance, and testing.

4.1 Review Procedure

The material reviewed consisted of the SAR, including several sets of responses to staff comments and concerns, submitted by the applicant. A number of design calculations provided by the applicant were also utilized in the review. This material was evaluated to establish compliance with the applicable requirements of 10 CFR Part 72. The review addressed the adequacy of natural convection cooling to maintain fuel cladding and HSM concrete temperatures within acceptable limits during normal, off-normal, and accident conditions. Thermal hydraulic aspects of the transfer cask design were also evaluated.

4.1.1 Design Description

The standardized NUHOMS system provides for the horizontal storage of irradiated fuel in a dry shielded canister which is placed in a concrete horizontal storage module. Decay heat is removed from the fuel by conduction and radiation within the DSC and by convection and radiation from the surface of the DSC. Natural circulation flow of air through the HSM and conduction of heat through concrete provide the mechanisms of heat removal from the HSM.

Spent fuel assemblies are loaded into the DSC while it is inside a transfer cask in the fuel pool at the reactor site. The transfer cask containing the loaded DSC is removed from the pool, dried, purged, backfilled with helium, and sealed. The DSC is then placed in a transfer cask and moved to the HSM. The DSC is pushed into the HSM by a horizontal hydraulic ram. The dry, shielded canister is constructed from stainless steel plate with an outside diameter of 107.8 cm (67.25 inches), a wall thickness of 1.6 cm (0.625 inches), and a length of 473 cm (186.25 inches). Within the DSC, there is a basket consisting of either 24 square cells in the PWR design or 52 cells for the BWR design. An intact spent fuel assembly is loaded into each cell yielding a capacity of either 24 PWR or 52 BWR assemblies per DSC. Spacer disks are used for structural support. The DSC has double seal welds at each end and rests on two steel rails when placed in the HSM.

The HSM is constructed from reinforced concrete, carbon steel, and stainless steel. Passageways for air flow through the HSM are designed to minimize the escape of radiation from the HSM but at the same time to permit adequate cooling air flow. Decay heat from the spent fuel assemblies within the canister is removed from the DSC by natural draft convection and radiation. Air enters along the bottom of each side of the HSM, flows around the canister, and exits through flow channels along the top sides of the module. Heat is also radiated from the DSC to the inner surface of the HSM walls where, again, natural convection air flow removes the heat. Some heat is also removed by conduction through the concrete.

The standardized NUHOMS system utilizes a transfer cask, transporter, skid, and horizontal hydraulic ram. The transporter, skid, and horizontal hydraulic ram are not affected by the thermal analysis. During transport and vacuum drying of the fuel in the DSC, heat is removed by conduction through the transfer cask.

4.1.2 Acceptance Criteria

Peak fuel cladding temperature for normal operation, calculated according to the methodology of PNL-6189, Reference 47, is the acceptance criterion relative to the fuel. The staff has reviewed this methodology and found it to be acceptable. Resulting peak fuel cladding initial storage temperature limits are 384°C (724°F) for PWR fuel and 421°C (790°F) for BWR fuel based on the long term average ambient temperature not exceeding 21°C (70°F). For accident events, the staff has accepted a peak fuel cladding temperature limit of 570°C (1058°F) based on Reference 48. Meeting these criteria for storage with an inert cover gas ensures that the criteria in of 10 CFR 72.122(h) are satisfied.

In Table 3.2-1 of the SAR, the applicant cites ACI-349-85 and ACI-349R-85 as the applicable criteria for concrete design. These criteria are acceptable to the staff with the exception that calculated concrete temperatures for both normal operation and accident conditions could exceed those of the ACI criteria. Use of a concrete mix and aggregate specification for higher temperatures is therefore required in lieu of the ACI 349 criteria. (See the materials discussion in Section 3.0 of this SER.)

This review of the thermal analysis also addresses the correctness of the calculated maximum temperatures and of temperature gradients used for input to structural evaluations. The manner of calculating maximum temperatures and thermal gradients for the structural analyses has been found to be acceptable.

4.1.3 Review Method

The thermal analysis was reviewed for completeness, applicability of the methods used, adequacy of the key assumptions, and correct application of the methods. Thermal analysis was performed primarily with the HEATING-6 (Reference 49) computer program. HEATING-6 is a part of the Oak Ridge National Laboratory SCALE package and is an industry standard code for thermal analysis. Representative input and output was reviewed to establish that the code use was appropriate and that the results were reasonable. Independent calculations were performed to check other portions of the analysis which did not use the HEATING-6 code. This includes the natural convection cooling calculation which determines the magnitude of the air flow through the HSM. Since the heat flux through the DSC surface is significantly increased for the standardized NUHOMS system design compared to the previous 24-P design (Reference 40), the ability to remove heat by air cooling is particularly important. An independent determination of the form losses and friction pressure drop, together with a balancing of the buoyancy and flow loss, confirmed the adequacy of the analysis.

Review effort was directed toward establishing the validity of the analyses and their applicability to the design. The analyses were reviewed for completeness, validity of input, reasonableness of results, and applicability of results to support conclusions regarding the design. In general, independent analyses were not performed. However, in some cases energy balances and simplified calculations were performed as a check.

4.1.4 Key Design Information and Assumptions

The key assumptions made in the thermal analysis are listed below.

1. The total heat generation rate for each fuel assembly is less than or equal to one kilowatt for each PWR assembly and equal to or less than 0.37 kilowatts for each BWR assembly. All heat is assumed to be generated in the fuel region.

2. Each dry storage canister contains 24 intact PWR assemblies or 52 intact BWR fuel assemblies.

3. A factor of 1.08 to account for uneven heat generation along the length of the fuel was assumed for thermal analysis inside of the DSC.

4. Design long term average ambient temperature of the external environment is taken as 21°C (70°F) with normal solar heat load. Limiting normal conditions of -17.8°C (0°F) and 37.8°C (100°F) ambient are also considered.

5. Off-normal temperatures of -40°C (-40°F) and 52°C (125°F) ambient temperature are considered. The 52°C (125°F) case assumed maximum solar heat load for the HSM, but use of solar shades and hence no solar heat load for the transfer cask is permissible above 37.8°C (100°F).

6. Accident condition is assumed to be total blockage of all inlets and outlets for five days with either -40°C (-40°F) or 52°C (125°F) and maximum solar heat flux ambient conditions.

7. A helium atmosphere is assumed to be maintained within the DSC over the entire storage life of the standardized NUHOMS system.

4.2 Horizontal Storage Module (HSM)

4.2.1 Design Evaluation

The SAR was reviewed in conjunction with three calculation packages, References 50, 51 and 52, and responses to several rounds of staff questions.

4.2.1.1 Normal Operation

A total of three cases were considered for normal operating conditions based on the temperature of the air at the inlet of the module. These are: (1) entering air at -17.8°C (0°F) representing "minimum normal conditions," (2) entering air at 21°C (70°F) representing "normal conditions," and (3) entering air at 37.8°C (100°F) representing "maximum normal conditions." The method of calculating concrete temperatures is acceptable. Concrete temperatures on the inside surface of the HSM reach 100°C (212°F) for the 24-P design, and 89.4°C (193°F) for the 52-B design, when the ambient temperature is 37.8°C (100°F). As long as the air temperature at the outlet remains within 37.8°C (100°F) of the ambient (for a heat load of 24 Kw), and maximum long term ambient temperature limits are not exceeded, the conservative design calculations show that the HSM concrete temperature limits and fuel cladding temperature limits will not be exceeded.

The applicant determined that the HSM wall temperature gradients for the 37.8°C (100°F) ambient temperature case are bounding among the normal and off-normal cases. These thermal gradients are either best estimate or conservative and are suitable for use in the structural loading analysis.

4.2.1.2 Off-Normal Conditions

The off normal conditions considered were an inlet temperature of -40°C (-40°F) representing extreme winter minimum and 52°C (125°F) representing extreme summer maximum. Solar heat flux of 1,397 kJ/hr-m^2 (123 Btu/hr-ft^2) was included for the extreme summer case. The concrete temperature on the inside surface of the HSM reaches a maximum of 121°C (250°F) for the 24-P design and 110°C (230°F) for the 52-B design for the extreme condition of 52°C (125°F) ambient temperature.

4.2.1.3 Accident Conditions

The total blockage of all air inlets and exits was analyzed as the accident case (Reference 50). Adiabatic heatup of the various components was assumed, with the HSM providing the slowest heatup rate. Adiabatic heating starting at the 52°C (125°F) inlet temperature condition is the limiting case for maximum concrete and fuel cladding temperatures. A heatup period of five days was assumed. The resulting concrete temperatures 249°C (480°F) on the floor and 231°C (448°F) on the roof are significantly above the acceptance criteria of 177°C (350°F) for accident conditions.

4.2.2 Discussion and Conclusions

Since concrete temperatures may exceed 93.3°C (200°F) during limiting normal conditions, a concrete mix and aggregate specification must be included for the elevated temperatures expected. These are considered acceptable as an alternative to satisfaction of the testing requirements of ACI 349, Section A.4.3. Satisfaction of the limiting condition for operation of a 37.8°C (100°F) maximum air temperature rise on exit from the HSM gives a reasonable degree of assurance that adequate cooling is achieved.

Based on the plot of HSM inside roof temperature response shown on page 25 of Reference 50, the HSM concrete temperature may exceed 177°C (350°F) sometime after 40 hours of flow blockage. Concrete temperature over 177°C (350°F) in accidents (without the presence of water or steam) is not acceptable due to reduction in strength and durability. In view of the facts that a PNFS proposed 4-day inspection frequency of the air inlets and outlets result in: (1) exceeding the ACI 349 and NRC staff recommendations for maximum concrete temperature limits, and (2) approaching the fuel clad temperature limits recommended by PNL-6189 (Reference 47), the NRC staff requires a daily inspection for the air inlets and outlets.

The applicant used thermal load inputs for the HSM stress analysis from the 37.8°C (100°F) ambient temperature case. Thermal gradients are best estimate or conservative and are suitable for use in the structural loading analysis.

See Table 4.2 for a summary of some component temperatures as a function of ambient temperatures.

4.3 DSC

4.3.1 Design Evaluation

The SAR was reviewed in conjunction with five calculation packages, References 53 through 57, and responses to several rounds of staff questions.

4.3.1.1 Normal Operating Conditions

Fuel cladding temperature limits based upon the methodology of PNL-6189 (Reference 47) have been proposed by the applicant. These limits are acceptable to the staff. The licensee must demonstrate that all fuel to be stored meets the criteria of PNL-6189, which are the accepted limits. The applicant has provided analyses demonstrating that these limits can be satisfied for normal and off-normal conditions provided that the fuel meets the acceptance criteria for storage.

The normal operating condition at 21°C (70°F) ambient air inlet temperature and the high temperature limiting normal case at 37.8°C (100°F) ambient air inlet temperature were analyzed for the DSC and internals. HEATING-6 input and output for the 21°C (70°F) case and the corresponding HSM run were reviewed. No errors were detected. Trends and magnitude of the resulting temperature distributions are reasonable. The HEATING-6 computer program is an industry standard code which is widley used for nuclear power plant thermal design analyses and has been in use for about twenty years. Application of the code for thermal design analysis of the standarized NUHOMS spent fuel storage system was performed in a conservative manner where input parameters were chosen so that conservatively high fuel cladding and HSM concrete temperatures were calculated. For the 21°C (70°F) ambient case, maximum cladding temperatures of 366°C (691°F) for PWR fuel and 417°C (782°F) for BWR fuel are below the acceptance criteria of 384°C (724°F) for PWR fuel and 421°C (790°F) for BWR fuel. For the limiting normal case of 37.8°C (100°F) ambient, the cladding temperatures are 371°C (699°F) for PWR and 420°C (788°F) for BWR fuel. These cladding temperatures are also below the acceptance criteria of 384°C (724°F) and 421°C (790°F), respectively, for PWR and BWR fuel. The temperature distribution within a spacer disk was determined from HEATING-6 calculations for the 37.8°C (100°F) ambient temperature case. Results of calculations with both helium and steel in the space between the guide sleeves and the DSC shell were used to determine the temperature distribution. The method used is appropriate for determining a temperature distribution for use in structural loading evaluations.

4.3.1.2 Off-Normal Conditions

The off-normal condition considered is the 52°C (125°F) ambient inlet air temperature. HEATING-6 calculations were performed which yielded a maximum cladding temperature of 423°C (793°F) for BWR fuel and 374°C (705°F) for PWR fuel compared to the acceptance criterion of 570°C (1058°F).

4.3.1.3 Accident Conditions

The applicant has analyzed the complete blockage of all air inlets and outlets for a 5-day period. This adiabatic heatup is addressed in References 50, 51 and 52. The fuel temperatures were calculated for this 5-day heatup period. A steady-state temperature distribution was assumed within the DSC, since its heatup rate is faster than that of the HSM. The resulting temperature distribution is acceptable for use in determining thermal loads. At the end of this time, BWR fuel has reached a temperature of 495°C (923°F), while PWR fuel has reached 447°C (836°F). These temperatures are below the acceptance criteria of 570°C (1058°F) for accident conditions.

4.3.2 Discussion and Conclusion

For normal operating temperatures the maximum fuel cladding temperature is below the acceptance criteria for both the 52-B and the 24-P designs. Therefore the fuel cladding is expected to be protected against degradation leading to gross rupture during long-term storage and the proposed maximum heat loads are acceptable.

Maximum temperatures for both PWR and BWR fuel remain below the acceptance criteria of 570°C (1058°F) off-normal conditions and for accident conditions following five days of adiabatic heatup. With a daily inspection frequency, as required by the NRC, the concrete temperature of the HSM does not exceed the 177°C (350°F) accident limit.

4.4 TC

The SAR was reviewed in conjunction with two calculation packages, References 58 and 59.

4.4.1 Design Evaluation

During loading, evacuation, and transport to the HSM, the DSC is located within a transfer cask. In this case the steady-state temperature distribution through the cask was determined by modeling the cask as a series of cylindrical annular regions to determine the radial distribution and as a series of heat slabs to determine the axial distribution. Both vertical and horizontal orientations of the transfer cask were considered.

4.4.1.1 Normal Operating Conditions

Ambient temperatures of -17.8°C (0°F), 21°C (70°F) and 37.8°C (100°F) were considered for normal operation. The surface temperature at the top and bottom of a horizontal DSC was determined for thermal stress evaluation. Axial temperature distribution was also determined for each of the three ambient temperatures. Maximum DSC surface temperature of 113°C (235°F) occurred at the top of the horizontal cask for the 37.8°C (100°F) case with the 24-P design heat load.

4.4.1.2 Off-Normal Operating Conditions

Extreme ambient conditions of -40°C (-40°F) and 52°C (125°F) were considered as off-normal events. Axial and radial temperature distributions were determined using the same methods as for normal operating conditions. A solar shade will be used whenever temperatures exceed 37.8°C (100°F). Therefore the 37.8°C (100°F) case becomes the limiting case for thermal gradient determination, yielding a 16.1°C (61°F) through wall temperature gradient.

Vacuum drying of the DSC before backfill with helium will result in increased fuel temperatures relative to normal conditions due to the decreased heat transfer within the DSC. Methods similar to that used for the normal operation case were used to determine the temperature gradient through the transfer cask wall, except that in this case the cask was oriented vertically. With an internal vacuum in the DSC, the maximum fuel cladding temperature was calculated to be 531°C (988°F) for BWR fuel and 487°C (909°F) for PWR fuel, both below the short term or accident temperature limit of 570°C (1058°F). These fuel clad temperatures for PWR and BWR fuels are calculated as steady state temperatures. It should be noted that the actual time required for vacuum drying of the DSC is small compared to the time necessary for the fuel cladding temperature to reach the calculated maximum value.

4.4.1.3 Accident Conditions

While references 58 and 59 consider the accident condition of loss of the neutron shield material, these results were not used in the structural loading evaluation since complete loss of the solid neutron shield material is not postulated to occur. Instead thermal loads from the 37.8°C (100°F) normal operation case were used.

4.4.2 Discussions and Conclusions

The limiting thermal condition of 37.8°C (100°F) with solar heat load was used to determine the thermal loading for all cases. Provided that a solar shade is used whenever the ambient temperature exceeds 37.8°C (100°F), the use of the 37.8° (100°F) case to determine the thermal loads is acceptable.

Since the DSC will be in the transfer cask for relatively short periods compared to the storage lifetime, use of the short term accident temperature limit for maximum fuel temperature is acceptable. The maximum fuel temperature is below this limit for all of the cases considered.

Table 4.2 Summary of Component Temperatures as a Function of Ambient Temperatures

Temperatures (°F)

Ambient	DSC Inside HSM				DSC Inside TC	
	Cladding	Cladding Limit	Concrete	Concrete Limit	Cladding	Cladding Limit
70 (normal steady state operation)	691 (PWR) 782 (BWR)	724 (PWR) 790 (BWR)	175 (PWR) 158 (BWR)	200		
100 (off-normal, but not accident)	699 (PWR) 788 (BWR)	724 (PWR) 790 (BWR)	212 (PWR) 193 (BWR)	200[1]	(DSC Vacuum) 909 (PWR) 988 (BWR)	1058
125 (steady state)	705 (PWR) 793 (BWR)	1058	250 (PWR) 230 (BWR)	350		
125 (5 day adiabatic heatup)	836 (PWR) 923 (BWR)	1058	480 (PWR) 480 (BWR) (350 @ 40 hrs.)	350[2]		

1. If concrete temperatures of general or local areas exceed 200°F but would not exceed 300°F, no tests or reduction of concrete strength are required if Type II cement is used and aggregates are selected which are acceptable for concrete in this temperature range.

2. Use of any Portland cement concrete where "accident" temperatures may exceed 350°F requires submission of tests on the exact concrete mix (cement type, additives, water-cement ratio, aggregates, proportions) which is to be used. The tests are to acceptably demonstrate the level of strength reduction which needs to be applied, and to show that the increased temperatures do not cause deterioration of the concrete either with or without load.

5.0 CONFINEMENT BARRIERS AND SYSTEMS EVALUATION

5.1 Description of Review

The primary confinement boundaries for fission products which are contained in the spent fuel are the intact fuel cladding (no known or suspected gross cladding breeches) and the DSC, which is a welded steel cylinder that is vacuum dried and backfilled with helium. The HSM is designed to provide shielding, structural support, ventilation, and weather protection for the DSC, but is not part of the confinement boundary. Similarly, the TC is designed to provide shielding during handling and transfer operations, but is not part of the confinement boundary.

The main parts of the secondary confinement boundaries for both versions of the DSC are a shell, outer bottom and top cover plates, top and bottom shield plugs, and inner top and bottom cover plates. The basket assembly is not part of the confinement boundary. The only penetrations required in the DSC are in the siphon and vent block, which is a part of the top shield plug. Two penetrations (with quick disconnect fittings) for vacuum drying and backfilling with helium are located in this block. No credit for confining the helium atmosphere is taken by the disconnect fittings. Two cover plates that mate with the siphon block are seal welded over the penetrations after the drying and helium backfilling operations have been completed. No components are required to penetrate the DSC after helium backfilling is completed and the structural lid is welded in place. No penetrations are used during spent fuel storage.

Tables 8.1-4a and 8.1-4b of the SAR report the design basis internal helium pressure for the PWR and BWR versions of the DSC. The PWR canister has marginally higher internal pressure but in all cases is very low. For normal operations with intact fuel cladding on the design basis average ambient temperature day 21°C (70°F), the internal helium pressure is 34.5 kPag (5.0 psig). For a 37.8°C (100°F) ambient temperature day, the internal helium pressure is only 47.6 kPag (6.9 psig). The accident case considers a 52°C (125°F) day with the HSM vents blocked and 100 percent cladding failure with the release of all of the fuel rod fill gas and 30 percent of the fission gas generated in PWR assemblies irradiated to 40,000 MWD/MTU.

The DSC is designed to meet the requirements of ASME Code, Section III, Subsection NB, and constructed in accordance with the ASME Code, Section III, Article NB-4000.

5.2 Design Evaluation

The staff considered Paragraph (1) of 10 CFR 72.122(h) as pertinent to storage of spent fuel in DSC. It requires that "spent fuel cladding must be protected during storage against degradation that leads to gross ruptures" and "that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage." Also, 10 CFR 72.236(e) requires that the cask must be designed to provide redundant sealing of confinement systems.

The confinement barriers and systems design are considered acceptable if it is demonstrated that: (1) there is a high likelihood that the DSC internal helium atmosphere will remain intact; (2) there is no operable corrosion mechanism that will lead to failure of the DSC to provide confinement; (3) there is no long-term cladding degradation mechanism in a helium atmosphere which could cause significant degradation or gross ruptures; and (4) there is insufficient time for cladding or fuel degradation during cask dry-out or off-normal behavior that could pose operational problems with respect to the removal of fuel from storage.

An NRC review of the NUHOMS design was made and documented in Section 5.0 of the Safety Evaluation Report for NUHOMS-24P, Revision 1, dated April 1989 (Reference 39). The NRC staff review was directed at two aspects of the design: (1) the mechanical integrity of the DSC, and (2) the long-term behavior of the cladding in an inert environment. The review was also directed at the impact of cask dry-out and off-normal behavior on fuel removal. The present review was directed at examining the review made in Section 5.0 of the SER of NUHOMS-24P to ensure that the results of that review also apply to the Standardized NUHOMS-24P and 52-B DSCs.

Because all of the parts of the confinement boundary are fabricated from stainless steel, the DSC is adequately protected from corrosion mechanisms.

The staff reviewed DSC integrity from the point of view of weld quality and inspections, adequacy of leak check methods on welds, other leakage paths, and long-term helium migration. Reviewers also checked the calculated stresses in the DSC under normal, off-normal, and accident conditions in order to verify that they are in the acceptable range. Cyclic fatigue of the DSC was also reviewed.

The staff evaluated cladding degradation by reviewing the pertinent technical literature in order to identify known and postulated mechanisms of gross failure of fuel in an inert atmosphere. Based on the literature search, calculations were performed for postulated failures by the mechanism of diffusion controlled cavity growth using a conservative set of assumptions. This was the only failure mechanism considered likely under the DSC storage conditions. The staff also evaluated the possible long-term creep or sag of the fuel cladding under the storage conditions since creep could affect removal of the fuel from storage. The effects of oxidation during the fuel dry-out period were also considered.

In its analysis of the cavity growth mechanism, the NRC staff determined that the area of decohesion at the end of a 20-year storage life is less than 4 percent, not high enough to cause any concern. The NRC staff found that creep or sag of the fuel cladding might equal 0.05 cm (0.020 inch), much less than the clearance available for removal of the rods. For postulated fuel oxidation of defective fuel rods during cask dry-out or off-normal behavior, cladding strain was determined to be much less than 1 percent so that fuel defect extension or fuel powdering is not anticipated. For all these areas of potential fuel degradation, the NRC staff calculations for the B&W fuel gave such conservative results, that they can equally well be applied to the fuels which meet the fuel specification cited in this SER. The NRC staff concludes that the standardized NUHOMS system design provides sufficient means to ensure that the fuel cladding is adequately protected against degradation that leads to gross ruptures.

The staff verified that the design of the DSC provides redundant sealing.

5.3 Conclusions

The staff concludes that the standardized DSC design conforms to the relevant criteria in 10 CFR 72.122(h), and 10 CFR 72.236(e) regarding redundant sealing of confinement systems. Confinement is ensured by a combination of inspection techniques, including radiographic inspection, helium leak testing, and dye penetrant testing. The confinement capability of the empty DSC shell without either bottom or top plate assemblies is ensured by radiographic inspection of the longitudinal full penetration weld and the girth weld. Helium leak testing is performed to ensure adequate sealing of the inner bottom cover to the DSC cylindrical shell. The confinement capability of the loaded DSC is ensured by helium leak testing after welding and dye penetrant testing of the inner top cover plate and vent port block to the DSC shell. All partial penetration welds are multiple pass welds subjected to dye penetrant testing. The inner seal welds are also helium leak tested. The outer seal welds are dye penetrant tested.

A number of tests and specifications relevant to confinement integrity were proposed by the vendor and determined to be appropriate. These include:

- DSC vacuum pressure during drying (SAR Section 10.3.2)
- DSC helium backfill pressure (SAR Section 10.3.3)
- DSC maximum permissible leak rate of inner seal weld (SAR Section 10.3.4)
- DSC dye penetrant test of closure welds (SAR Section 10.3.5).

These are included as conditions for system use in Section 12.2 of this report.

The staff also requires the following condition for the use of the system:

1. If fuel needs to be removed from the DSC, either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of oxidized fuel and to prevent radiological exposure to personnel during this operation. This can be achieved with this design by the use of the penetration valves which permit a determination of the atmosphere within the DSC before the removal of the shield plug. If the atmosphere

within the DSC is helium, then operations can proceed normally with fuel removal either via the transfer cask or in the pool. However, if air is present within the DSC, then appropriate filters should be in place to preclude uncontrolled release of airborne radioactive particulate from the DSC via the penetration valves. This will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection is required as appropriate.

The above is included as a condition for system use in Section 12.1.2 of this report.

Because the DSC confinement barrier material is stainless steel, adequate provision for corrosion protection is part of the DSC design. Additionally, the fluence of the neutron flux for a 20-year period of storing the spent fuel is eight orders of magnitude less than the fluence encountered within an operating reactor. For this reason embrittlement due to neutron flux is not considered to be a concern. A discussion of neutron embrittlement follows.

The fuel cladding which has been placed within the DSC has been subject to extensive neutron irradiation while it was present in the reactor core producing power. A representative order of magnitude total neutron flux within the core of an operating nuclear power plant is approximately 1 E+12 neutrons/sq. cm.-sec. Core residence time at power for nuclear fuel prior to its placement in the NUHOMS ISFSI is about 28.8 months (80% power operation over a 36-month time period). This results in a total fuel cladding irradiation neutron fluence of 7.5 E19 neutrons/sq. cm. In comparison, the <u>bounding total</u> neutron emission per fuel assembly delineated in the NUHOMS fuel specification is 2.23 E8 neutrons/second. This source, distributed over the entire surface of the decay heat producing section of the fuel rods in a fuel assembly, results in an average neutron flux of 8.9 E+2 neutrons/sq. cm.-sec. Over the 20-year life of the ISFSI, this additional neutron fluence to the cladding would be 5.6 E11 neutrons/sq. cm. which is <u>eight orders of magnitude</u> lower than the fluence already accumulated during power operation. The neutron fluence to the fuel cladding represents an insignificant addition to the fluence absorbed by the cladding during power operation. The long-term ISFSI storage would, therefore, not be expected to create any neutron irradiation induced damage to fuel cladding beyond that already caused by irradiation of this fuel while residing in the reactor core during power operation.

The staff considered three potential mechanisms for the deterioration of the integrity of fuel rods. The first was potential failure of the cladding by the diffusion controlled cavity growth mechanism. The staff determined that the area of decohesion was less than 4 percent, not high enough to cause any concern. The second mechanism examined was creep of the fuel cladding. It was found to be a maximum of 0.05 cm (0.020 inches), much less than the clearance available for removal of the rods. The third mechanism examined was oxidation of the fuel during the dry-out period. Cladding strain was determined to be much less than 1 percent for postulated fuel oxidation of defective fuel rods. The staff concludes that the

DSC design has provided sufficient means to assure that the fuel cladding is adequately protected against degradation that leads to gross rupture.

6.0 SHIELDING EVALUATION

6.1 Design Description

The radiation shielding for the stored fuel assemblies is provided by a variety of shielding materials and operational procedures. The HSM has thick concrete walls and roof as well as a heavy shielded door and shielded air outlet vents to reduce radiation. The DSC has thick shield plugs on both ends to reduce the dose to plant workers. The TC has shielding incorporated in both ends as well as the entire shell.

From the operational side of the design, placement of demineralized water in the annulus between the TC and DSC provides shielding as well as reducing contamination of the DSC exterior as well as the TC interior surfaces. The use of water in the DSC cavity during placement of the DSC inner seal weld reduces direct and scattered radiation exposure. Temporary shielding is used during DSC draining, drying, inerting and closure operations.

6.2 Design Evaluation (Source Specification and Analysis)

The neutron and gamma ray radiation source terms were calculated for design basis fuels (for the PWR B&W 15x15 fuel and for the BWR GE 7x7 fuel). For both fuels a maximum initial enrichment of 4.0 wt% U-235 is assumed and a post-irradiation cooling time equivalent to five years is assumed. The PWR fuel is assumed to be subjected to an average fuel burnup of 40,000 MWD/MTU; the BWR fuel is assumed to be subjected to an average fuel burnup of 35,000 MWD/MTU.

Neutron source are based on spontaneous fission contributions from six nuclides (predominantly Cm-242, Cm-244, and Cm-246 isotopes) and (α, n) reactions due almost entirely to eight alpha emitters (predominantly Pu-238, Cm-242, and Cm-244). The fission spectrum used in shielding calculations is a weighted combination of the principal contributors. The total neutron source strength for PWR fuel is 2.23E8 neutrons per second per assembly (or 5.35E9 neutrons per second for 24 PWR fuel assemblies). The total neutron source strength for BWR fuel is 1.01E8 neutrons per second per fuel assembly (or 5.25E9 neutrons per second for 52 BWR fuel assemblies).

For the BWR fuel the neutron and gamma source strengths and gamma energy spectrum and decay heat were calculated using the ORIGEN 2 computer code (References 60, 61). ORIGEN 2 is a widely used and validated code which has been utilized and approved for previous ISFSI radiation source term calculations. The heavy metal weight and the weights of other materials of the fuel assembly are chosen to bound those for BWR fuel assemblies to give the highest possible values for neutron and gamma source strengths and decay heat.

Gamma radiation sources include 70 principal fission product nuclides within the spent fuel and several activation products and actinide elements present in the spent fuel and fuel assemblies. The gamma energy spectrum includes contributions for each source isotope as calculated by ORIGEN 2 calculations. The total gamma source strength for BWR fuel is 4.86E15 Mev/s/MTHM (or 1.37E17 photons/second for 52 BWR fuel assemblies).

For the PWR fuel neutron source data and neutron spectrum were taken from previous ORIGEN 2 calculations; the results of which bound data from the Office of Civilian Radioactive Waste Management (OCRWM) database (Reference 62). Gamma ray sources were determined using the OCRWM database with the gamma spectrum determined using the Microshield computer program (Reference 63). The spectrum results were segmented into the 18 energy group structure used in the shielding calculations with the results normalized to preserve the total gamma power calculated in the OCRWM database. The total gamma source for PWR fuel is 5.81E15 Mev/s/MTHM (or 1.79E17 photons/second for 24 PWR fuel assemblies).

The neutron energy spectrum used for the shielding analysis is given in Table 7.2-1a of the SAR for PWR fuel and in Table 7.2-1b of the SAR for BWR fuel. The gamma energy spectrum used for shielding analysis is given in Table 7.2-2a of the SAR for PWR fuel and in Table 7.2-2b of the SAR for BWR fuel.

The shielding analysis used the same suite of computer codes as those used in the NUHOMS-24P Topical Report. These computer codes are: ORIGEN-2, ANISN (Reference 64), QAD-CGGP (Reference 65), MICROSKYSHINE, and MICROSHIELD. Collectively these codes were used to calculate both the gamma and neutron direct and scattered dose rates and, as previously discussed, the radiation source terms for spent fuel assemblies. All of the these computer codes have been used and benchmarked throughout the nuclear industry.

6.3 Discussion and Conclusions

The staff's review of the standardized NUHOMS system shielding calculations included a combination of reviewing the files provided by the applicant and performing independent check calculations on the files. There were no independent audit calculations of the dose rates since the applicant has demonstrated its proficiency in the application of these same methods and the validation and verification of these computer codes for previous ISFSI applications.

The check calculations of the shielding analysis files did not reveal any arithmetic and/or other numerical errors or indicate any changes to be made in the calculations. The calculated dose rates appear to be comparable to previously calculated dose rates. This conclusion also applies to the applicant's calculation of direct and air-scattered dose rates in and around the HSM. Dose rates at locations of interest were calculated for 5-year cooled PWR fuel and presented in Figure 7.3-2 of the SAR. The SAR states that "Consistent with the relative design basis source strengths, the shielding analysis results for the NUHOMS-24P envelop those of the NUHOMS-52B systems." The calculation packages submitted by the applicant presented sufficient information to support this assumption.

The applicant has performed an extensive number of shielding dose rate analyses for the standardized NUHOMS system design using a neutron and gamma ray source term for 4.0 wt% PWR fuel irradiated to 40,000 MWD/MTHM and cooled for a period of five years. This source term has been shown by the applicant to bound that calculated for 4.0 wt% BWR fuel irradiated to 35,000 MWD/MTHM and cooled for a period of five years. For both the PWR and BWR fuels the source terms were calculated to be conservative relative to the fuel types of possible concern. The PWR neutron source data and source spectra were calculated using the ORIGEN-2 computer code and the results were found to bound data from the OCRWM database. Gamma ray sources were taken from the OCRWM database with the gamma spectrum determined using the MICROSHIELD computer program. The computer codes and methods used to delineate the neutron and gamma sources for the standardized NUHOMS system design are acceptable.

Using the ANISN, QAD-CGGP, MICROSKYSHINE, and MICROSHIELD computer codes, the applicant calculated both direct and air-scattered dose rates in and around the HSM. These codes and methods have been previously used by this applicant and were reviewed and approved for the NUHOMS-24P Topical Report. The calculated dose rates presented for the standardized NUHOMS design appear to be consistent and conservative relative to previously presented results.

Independent calculations of the applicant's shielding analysis files did not reveal any arithmetic and/or other numerical errors or indicate any changes to be made in the calculations. Based on a detailed review of the inputs, methods, computer codes, assumptions and dose rate results, including calculational checks of the shielding analysis files, the shielding design analysis for the standardized NUHOMS system has been found to be acceptable and should be sufficient to meet the requirements of 10 CFR 72.104 and 10 CFR 72.106 as required by 10 CFR 72.236(d). In accordance with 10 CFR 72.212(b)(2), users must perform written evaluations to establish that these requirements have been met.

7.0 NUCLEAR CRITICALITY SAFETY EVALUATION

From the standpoint of criticality safety, the standardized NUHOMS system consists of two separate designs; one for the storage of 24 irradiated PWR fuel assemblies which is referred to as the standardized NUHOMS-24P design; and the one for the storage of 52 irradiated BWR fuel assemblies which is referred to as the standardized NUHOMS-52B design. Therefore the criticality safety evaluation of the two designs will be discussed separately.

7.1 Design Description

7.1.1 Standardized NUHOMS-24P Design

Criticality safety, according to the vendor, is ensured by the inherent geometry and material characteristics of the standardized NUHOMS-24P system and by establishing specific criteria for acceptance of irradiated fuel assemblies for storage. There is not a specific design feature such as fixed neutron poisons intended to provide assurance of nuclear criticality safety. The system is designed to provide nuclear criticality safety during both wet loading and unloading operations.

7.1.2 Standardized NUHOMS-52B Design

Criticality safety, according to the vendor, is ensured through a combination of geometrical and neutronic isolation of fuel assemblies. Fixed neutron absorbers in the form of borated stainless steel plates are used to control the reactivity of the assembly of stored BWR fuel assemblies such that criticality safety is assured under optimum moderation conditions for all initial fuel enrichments equal to or less than 4.0 wt.% U-235.

7.2 Design Evaluation

7.2.1 Standardized NUHOMS-24P Design

In addressing nuclear criticality safety for the standardized NUHOMS-24P design, the staff has applied criteria in 10 CFR 72.124 and 10 CFR 72.236(c). 10 CFR 72.124 provides that the system should be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes occur. It states that the design of the system must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations. It calls for the design to demonstrate safety for the handling, packaging, transfer, and storage conditions, and in the nature of the immediate environment under accident conditions. It states that the design should be based on favorable geometry, permanently fixed neutron absorbing materials, or both. 10 CFR 72.236(c) requires that the cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

The design criteria proposed by the vendor in the SAR are that k_{eff}, remains below 0.95 during both normal operation and accident conditions. These design criteria were determined by the staff to be acceptable.

The criticality evaluation of the standardized NUHOMS system design was presented in SAR Section 3.3.3. The vendor performed criticality calculations to determine the most reactive fuel type and to show criticality safety of the standardized NUHOMS-24P system. According to the vendor, the B&W 15x15 fuel is the most reactive PWR fuel assembly and was selected as the design basis for the standardized NUHOMS-24P design.

The criticality safety analysis for the standardized NUHOMS-24P system presented in the SAR was performed using a calculational methodology consisting of several standard computer programs: The CASMO-2 (Reference 66) computer program was used to calculate the irradiated fuel actinide number density data as a function of burnup. The SAS2 (Reference 67) sequence in the SCALE-3 (Reference 60) criticality safety analysis code system was used to calculate the reactivity of the array of stored irradiated fuel assemblies. The SCALE-3 code system used in the analysis presented in the SAR was a mainframe version of the programs and included ORIGEN-S to perform fuel burnup, depletion, and decay calculations, and KENO-IV (Reference 68) code for criticality calculations. The cross sections used in the criticality safety analysis were the 123 group library from the SCALE system.

The KENO-IV code and the calculational methodology utilized to calculate k_{eff} was benchmarked against 40 critical experiments as presented in Section 3.3 of the SAR.

The criticality analysis presented in the SAR and supplementary response to requests for additional information were reviewed by the staff. Some independent and confirmatory calculations were also performed to verify important sensitivities in the criticality analysis.

7.2.2 Standardized NUHOMS-52B Design

For the standardized NUHOMS-52B design, the staff used the nuclear criticality safety criteria in 10 CFR 72.124 and 10 CFR 72.236(c). 10 CFR 72.124 provides that the system should be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes occur. It states that the design of the system must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations. It provides that the design should demonstrate safety for the handling, packaging, transfer, and storage conditions, and in the nature of the immediate environment under accident conditions. It also provides that the design should be based on favorable geometry, permanently fixed neutron absorbing materials, or both. 10 CFR 72.236(c) requires that the cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

In order to address the criteria of 10 CFR 72.124(b), the staff has considered the following. The staff used the bounding neutron flux of the 52B spent fuel and calculated the reaction rate from thermal neutron absorption in boron and then evaluated this rate for a 20-year storage life. The result of this calculation is a maximum depletion of boron of approximately 0.04% for 20 years, which is small compared to the design tolerance of the absorber material and can therefore be considered insignificant. Aside from this boron depletion mechanism due to thermal neutron absorption in boron, the staff has not postulated other mechanisms which could reduce the efficacy of the fixed neutron absorber.

The design criteria proposed by the vendor in the SAR are that k_{eff}, remains below 0.95 during both normal operation and accident conditions for optimum moderation density. These design criteria were determined by the staff to be acceptable.

The criticality evaluation of the standardized NUHOMS-52B design was presented in SAR Section 3.3.3. The vendor performed criticality calculations to determine the most reactive fuel type and to show criticality safety of the standardized NUHOMS-24P design. According to the vendor, the GE-2 7x7 fuel is the most reactive BWR fuel assembly and was selected as the design basis fuel for the standardized NUHOMS-52B.

The criticality safety analysis for the standardized NUHOMS-52B design presented in the SAR was performed with a calculational methodology using the microcomputer application KENO-Va (Reference 69) and the Hansen-Roach 16 group cross section working library. A small computer program, designated PN-HET, was developed by the vendor to automate the computation of resonance parameters necessary for mixing the Hansen-Roach cross sections.

The KENO-Va/PN-HET code system was benchmarked against 40 critical experiments as presented in a separate computer code QA verification document, KENO5A-QA (Reference 70).

The criticality analysis presented in the SAR and supplementary response to requests for additional information were reviewed by the staff. Some independent and confirmatory calculations were also performed to verify important sensitivities in the criticality analysis.

7.3 Conclusions

7.3.1 Standardized NUHOMS-24P Design

On the basis of the analysis presented in the SAR, the supplementary analysis presented in response to questions, and confirmatory calculations performed by the staff, it was determined that the standardized NUHOMS-24P design and proposed operating procedures are adequate to maintain the system in a subcritical configuration and to prevent a nuclear criticality accident, and therefore satisfy 10 CFR 72.124 and 10 CFR 72.236(c), subject to the key factors assumed by the vendor in the analysis. Specifically:

1. Criticality safety calculations presented in the SAR and independent confirmatory calculations performed by the staff show that criticality safety is ensured for a maximum U-235 initial enrichment equivalent to 1.45 wt.% which was determined for the design basis B&W 15x15 fuel assemblies.

2. The criticality safety analysis of the misloading of unirradiated fuel assemblies presented in the SAR and independent confirmatory calculations performed by the staff show that the array reactivity can be maintained subcritical ($k_{eff} < 0.95$) in this accident situation by filling the DSC with borated water before wet loading or unloading. The minimum level of boration required as determined by the staff analysis, based on 4.0 wt.% enrichment of unirradiated B&W 15x15 fuel assemblies was determined to be 2,000 ppm. The analysis presented in the SAR determined the minimum level of boration to be 1,810 ppm. In lieu of resolution of the difference between the staff and SAR analysis of the required minimum level of boration, the more conservative value of the staff analysis is taken. The SAR assumed that the B&W 15x15 fuel presents a bounding case.

The key factors and assumptions used by the vendor in the criticality safety analysis are as follows:

1. The maximum initial fuel enrichment evaluated for irradiated fuel assemblies is 4.0 wt.% U-235.

2. The DSC is filled with borated water during fuel loading and unloading operations. The required boron concentration is determined for maximum fuel enrichment.

3. Only irradiated fuel assemblies with an initial enrichment equivalent < 1.45 wt.% U-235 will be loaded into the DSC. The criticality acceptability curve of minimum burnup versus enrichment is shown in Figure 3.3-3 of the SAR.

4. Fuel assemblies are no more reactive than the design basis 15 x 15 rod array.

5. Accidents resulting in altered mechanical configuration of the array of fuel assemblies are not credible.

6. Accidents during dry storage that result in the flooding of the DSC with unborated water are not credible.

Key factors 1, 2, 3, and 4 are reflected in the fuel specification discussed in Section 12.2.1.

Previous evaluation of vendor topical reports and site-specific applications involving casks with large number of assemblies (e.g., 24) have addressed the potential for criticality when water is added to the cask or canister before fuel removal. Because NRC staff position does not yet allow for burnup credit, the past analyses have assumed a full load of fresh fuel and considered the case for optimum moderation. These analyses have been the limiting cases for nuclear criticality safety. Minimum boron concentration in the DSC cavity water, during wet loading and unloading operations, is discussed as a condition for system use in Section 12.2.15.

7.3.2 Standardized NUHOMS-52B Design

The conclusions of the analysis of the standardized NUHOMS-52B design are more straightforward since the standardized NUHOMS-52B system is designed to provide assurance of nuclear criticality safety under optimum moderation conditions for loading of unirradiated fuel assemblies of a maximum enrichment of 4.0 wt.% U-235. This simplification is due to the use of fixed neutron absorbers in the design. The initial SAR also requested certification of a low-enrichment design which contained no neutron absorber plates but limited the initial fuel enrichment. Independent criticality safety calculations performed by the staff did not confirm that criticality safety was ensured in this low enrichment design. The vendor has withdrawn the low-enrichment design from further consideration.

On the basis of the analysis presented in the SAR and subsequent revisions, and independent confirmatory calculations performed by the staff, it was determined that the standardized NUHOMS-52B system design and proposed operating procedures are adequate to maintain the system in a subcritical configuration and to prevent a nuclear criticality accident, and therefore satisfy 10 CFR 72.124 and 10 CFR 72.236(c), subject to the key factors assumed by the vendor in the analysis. Specifically:

1. Criticality safety calculations presented in the SAR and independent confirmatory calculations performed by the staff show that criticality safety is ensured for a maximum initial U-235 fuel enrichment of 4.0 wt.% which was determined for the design basis GE-2 7x7 fuel assembly.

2. The criticality safety analysis assumes a minimum boron density of 0.75 wt% boron in the borated stainless steel absorber plates.

The key factors and assumptions used by the vendor in the criticality safety analysis are as follows:

1. The maximum initial fuel enrichment of fuel assemblies stored in the standardized NUHOMS-52B system is 4.0 wt.% U-235.

2. The boron loading in the neutron absorber plates is a minimum of 0.75 wt.%.

4. Accidents resulting in an altered mechanical configuration of the array of fuel assemblies are not credible.

Key factors 1, 2, and 3 are reflected in the fuel specification discussed in Section 12.2.1.

8.0 RADIOLOGICAL PROTECTION EVALUATION

8.1 Design Description

The main radiation protection features of the standardized NUHOMS system design are described in Sections 7.1.2 and 7.3 of the SAR and include: (1) radiation shielding; (2) radioactive material confinement; (3) prevention of external surface contamination; and (4) site access control.

Shielding includes many features designed to reduce direct and scattered radiation exposure, including:

1. Thick concrete walls and roof on the HSM which limit the dose rate to site workers and the off-site population;

2. A thick shield plug on each end of the DSC to reduce the dose to workers performing drying and sealing operations, and during transfer of the DSC in the transfer cask and storage in the HSM;

3. Use of a shielded transfer cask for DSC handling and transfer operations which limits the dose rate to ISFSI and plant workers;

4. Filling of the DSC cavity and the DSC-transfer cask annulus with water during DSC closure operations to reduce direct and scattered radiation exposure ; and

5. Use of temporary shielding during DSC draining, drying, inerting and closure operations as necessary to further reduce direct and scattered radiation dose rates.

The confinement features of the standardized NUHOMS system control the release of gaseous or particulate radionuclides and are described in Section 3.3.2. These features include:

1. The cladding of the stored fuel assemblies, which provides the first level of confinement;

2. The DSC confinement pressure boundary which provides the second level of confinement. The DSC confinement boundary includes: the DSC shell, the inner seal weld primary closure of the DSC, the DSC shielded end plugs, the outer seal weld secondary closure of the DSC, and the DSC cover plates.

The DSC has been designed as a weld-sealed containment pressure vessel with no mechanical or electrical penetrations. All the DSC pressure boundary welds are inspected according to the appropriate articles of the ASME B&PV Code, Section III, Division 1, Subsection NB

(Reference 9). These criteria ensure that the weld metal is as sound as the parent metal. As pointed out in the description of the DSC in Section 3.2.1, the double seal welds at the top and bottom of the DSC do not comply with the ASME Code. Consequently, the weld inspection requirements are also not strictly in accordance with Section NB-5000 of the Code. The staff has accepted alternative inspection and test requirements in lieu of the Code.

Contamination of the DSC exterior and transfer cask interior surfaces is controlled by placing demineralized water in the transfer cask and DSC during loading operations, then sealing the DSC/cask annulus. In addition, surface contamination limits for the DSC have been established, and are discussed in Section 9.2.

Access to the site of the NUHOMS ISFSI would be restricted by a periphery fence to comply with 10 CFR 72.106(b) controlled area requirements. The details of the access control features will vary from site to site, but must meet the requirements of 72.106(b) for the access to the controlled area. In addition to the controlled area restrictions, access to the spent fuel is restricted by an HSM access door, which is welded in place. This door weighs approximately 2.7 t (3 tons) and would require heavy equipment for removal.

8.2 Design Evaluation

This section evaluates the radiation protection features of the standardized NUHOMS design separately with regard to (1) on-site occupational exposures under normal loading and storage conditions, and (2) off-site exposures under normal storage conditions and in the event of accidents.

8.2.1 On-Site Radiological Protection

Regulatory requirements for on-site radiological protection are contained in 10 CFR 20.1101 and 20.1201-1208 which require the licensee to provide the means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the objective of maintaining exposures as low as is reasonably achievable (ALARA).

Section 20.1201(a) of 10 CFR Part 20 states that the licensee shall control the occupational dose to individual adults to the dose limits specified in 1201(a)(1) and 1201(a)(2). Also, section 20.1101 of 10 CFR Part 20 states that each licensee shall develop, document, and implement a radiation protection program and that the licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

Section 72.126(a) provides that radiation protection systems shall be provided for all areas and operations where on-site personnel may be exposed to radiation or airborne radioactive materials.

Guidance for ALARA considerations is also provided in NRC Regulatory Guides 8.8 and 8.10 (References 10 and 11).

Radiation protection for on-site personnel is considered acceptable if it can be shown that the non-site-specific considerations (1) will maintain occupational radiation exposures at levels which are ALARA, (2) are in compliance with appropriate guidance and/or regulations, and (3) will ensure that the dose from associated activities to any individual does not exceed the limits of 10 CFR Part 20.

The calculational methods used in the estimation of on-site doses are described in detail in the SAR. These methods focused on the use of the ANISN, QAD-CGGP, MICROSKYSHINE, and MICROSHIELD (References 64, 65, and 63, respectively) radiation transport codes, as well as manual calculations, to calculate exposure rates around the DSC in a transfer cask and an HSM. Dose rate maps in the general vicinity of a 2x10 array and two 1x10 arrays containing 10-year-old fuel were constructed.

The calculational methods and results presented in the SAR and associated calculation packages were reviewed for completeness, correctness, and internal consistency. In addition, confirmatory calculations were performed for the gamma-ray dose rates at various locations around the DSC, TC, and HSM.

Radiation doses to on-site workers were not calculated in the SAR. Rather, a summary of the operational procedures which lead to occupational exposures are presented, as are the number of personnel required, the estimated time for completion, and the average source-to-subject distance. This information can be used with dose map results to assess the individual and collective on-site doses.

The SAR notes that experience with an operating standardized NUHOMS system has shown that the collective occupational dose associated with placing a canister of spent fuel into dry storage is less than 0.014 person-Sv (1.4 person-rem), and that the application of effective procedures by experienced ISFSI personnel can reduce the collective dose to below 0.01 person-Sv (1 person-rem) per canister. A detailed assessment of operator doses and the possible provision of management or administrative controls to meet ALARA criteria is the responsibility of the user in accordance with its 10 CFR Part 50 licensee's radiation protection program and 10 CFR Part 20.

Other workers at the nuclear power plant site will also be exposed to direct and air-scattered (skyshine) radiation during the transfer and storage phases of ISFSI operation. Examples of activities involving such exposure are surveillance of the HSMs, and site operations which are not associated with spent fuel storage but which are performed in the general vicinity of the storage area. Major factors influencing the magnitude of the exposures are the occupancy times and spatial distribution of workers, and the intensity of the radiation field. An assessment of the expected on-site doses incurred by site personnel not directly involved in ISFSI operations is the responsibility of the user in accordance with its 10 CFR Part 50 licensee's radiation protection program and 10 CFR Part 20.

8.2.2 Off-Site Radiological Protection

8.2.2.1 Normal Operations

Regulatory requirements for off-site radiological protection in 10 CFR Part 20 require that dose to members of the public should be kept within the limits of 20.1301 and should be ALARA. Section 72.104(a) of 10 CFR Part 72 requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual located beyond the controlled area shall not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other organ as a result of exposure to (1) planned discharges of radioactive materials (except for radon and its daughter products) to the general environment, (2) direct radiation from ISFSI operations, and (3) any other radiation from uranium fuel cycle operations within the region.

Off-site radiological protection features of the standardized NUHOMS system are considered acceptable if it can be shown that design and operational considerations, which are not site-specific, result in off-site dose consequences in compliance with the applicable sections of 10 CFR Parts 20 and 72, and that these doses to off-site individuals are ALARA.

The two principal design features which limit off-site exposures during normal operations are the confinement features of the double-seal welded DSC, and the radiation shielding of the DSC and the HSM. During transfer operations, shielding in the radial direction is provided by the transfer cask. The confinement features of the DSC control the release of gaseous or particulate radionuclides. There are no liquid effluents from the ISFSI. During normal operations, the only pathway of exposure to the off-site population is direct and scattered radiation from the stored fuel.

The review for off-site radiological protection mainly involved a detailed evaluation of the methods applied and the results obtained in the applicable SAR sections, supplemented by additional information (including detailed calculation packages) provided by the applicant on these methods and results. For the case of off-site doses from direct and scattered (or "skyshine") radiation, an evaluation was performed on the application of MICROSKYSHINE, MICROSHIELD, and manual calculation methods, which were used to calculate gamma-ray and neutron dose equivalent rates at various locations in and around the HSM and to construct a dose-versus-distance curve. The dose rates predicted by this curve for various off-site distances was used to assess the general level of compliance with the dose rate criteria of 10 CFR 72.104(a) and for 10 CFR Part 20.

The dose to an off-site individual residing at some distance from a filled standardized NUHOMS system array will vary depending on a number of factors, including fuel type, size, and geometry of the array, and the directional orientation of the receptor with respect to the array. A conservative estimation of the distance required to reduce the full-time occupancy dose rate from a filled ISFSI array to 0.25 mSv/yr (25 mrem/yr) is approximately 300 meters. Normal operation of a standardized NUHOMS HSM would comply with the dose rate criteria of 10 CFR 72.104(a), provided site-related factors allow for a sufficient distance to the controlled area boundary. As required by 10 CFR 72.212(b)(2)(iii), this must be evaluated by the user before storing fuel in a standardized NUHOMS system ISFSI.

8.2.2.2 Off-Normal Operations

Section 72.106(b) requires that any individual located on or near the closest boundary of the controlled area (at least 100 m) shall not receive a dose greater than 0.05 Sv (5 rem) to the whole body or any organ from any design basis accident.

Off-normal events and postulated accidents that could result in the loss of shielding or the release of radionuclides are analyzed in Sections 8.1 and 8.2 of the SAR. In particular, an accident resulting in an instantaneous release of 30 percent of fission gas inventory (mainly Kr-85) is assessed in Section 8.2.8. The SAR reports that this accident results in a dose at 300 meters from the ISFSI site of 0.53 mSv (0.053 rem) to the whole body and 0.067 Sv (6.7 rem) to the skin. These results were confirmed by independent calculations.

The dose to the whole body is well within the 0.05 Sv (5 rem) limit prescribed by 10 CFR 72.106(b). The calculated skin dose exceeds this limit by a small amount, although the conservative, generic nature of the assessment warrants that the DSC leakage event be further assessed for site-specific applications. It should also be noted that, as indicated in the SAR, no credible conditions have been identified which could breach the canister body or fail the double-seal welds at each end of the DSC. Thus, these dose results are only presented to bound the consequences that could conceivably result, and to evaluate compliance with the 10 CFR 72.106(b) requirement.

Other accidents are assessed in Section 8.2 (e.g., floods, tornados, earthquakes, accidental cask drop, blockage of air inlets and outlets, etc.), but the SAR concludes that none of these other accidents represent credible sources of off-site dose consequences.

8.3 Discussion and Conclusions

8.3.1 On-site Radiological Protection

The shielding, confinement, and handling design features of the standardized NUHOMS design conform to the on-site radiological protection requirements of 10 CFR Part 20 and are considered acceptable for the set of conditions assumed in this review. Dose rates calculated by the vendor for different locations around the standardized NUHOMS system design are significantly higher than those determined for previous NUHOMS designs. This is specifically reflected in the dose rate limits delineated in Operating Limit 12.2.7 of this SER.

Although independent review analyses and more exact dose calculation methods may result in lower dose rates, the relative dose rates for this design are still expected to be higher than comparably calculated dose rates for earlier NUHOMS designs. These relatively higher dose rates are not consistent with the objective of maintaining occupational exposures ALARA. Site-specific applications with this design should provide detailed procedures and plans to meet ALARA guidelines and 10 CFR Part 20 requirements with respect to the operation and maintenance of this standardized NUHOMS system ISFSI design. As discussed above, details of access control, surveillance, and other operational aspects affecting on-site exposure must be in compliance with existing licensee's radiation protection program.

8.3.2 Off-site Radiological Protection

The shielding and confinement design features of the standardized NUHOMS system design conform to the off-site radiological protection requirements of 10 CFR Part 72 and 10 CFR Part 20 and are considered acceptable for the set of conditions assumed in this review. The use of high-integrity double-seal welds on the DSC ensures that, during normal operation, there are no effluents from the standardized NUHOMS system. Off-site dose is, therefore, due strictly to direct and scattered radiation, the intensity of which is a function of distance and direction from the array. Site-specific factors such as the number of HSMs in the storage array, the distance and direction of the nearest boundary of the controlled area, the contribution of reactor plant effluents to the off-site dose, and resultant collective off-site dose must be considered in the compliance evaluation for a proposed standardized NUHOMS system at a specific site. This evaluation must be performed by each user to assure compliance with 10 CFR 72.212 and 10 CFR 20.1207. This requirement is contained in the conditions for system use in Section 12.2.18 of this SER.

9.0 DECOMMISSIONING/DECONTAMINATION EVALUATION

9.1 Design Description

The standardized NUHOMS system design recognizes the need for decommissioning at the end of its useful life. External contamination of the DSC is limited by its containment features and through the contamination control procedures used during DSC fuel loading. In particular, contamination levels on the external surface of the DSC are minimized by the use of uncontaminated water in the DSC and cask/DSC annulus during fuel pool loading operations. This prevents contaminated fuel pool water from contacting the DSC exterior. Also, there is no credible chain of events which would result in widespread contamination outside of the DSC.

The SAR also states that the DSC is designed to interface with a transportation system planned to transport canistered intact fuel assemblies (i.e., filled DSC's) to either a monitored retrievable storage facility (MRS) or a geologic repository. Until the transportation system is available, the DSCs are not approved at this time for transportation. If the fuel must be removed from the DSC at the reactor site before shipment, the DSC will likely require decontamination of the internal surfaces and disposal as low-level radioactive waste. Once the DSC's have been removed, only small amounts of residual contamination are expected to remain in the HSM passages, thereby facilitating easy decommissioning.

9.2 Design Evaluation

Section 72.130 of 10 CFR Part 72 provides criteria for decommissioning. It provides that considerations for decommissioning should be included in the design of an ISFSI and that provisions be incorporated to (1) decontaminate structures and equipment; (2) minimize the quantity of waste and contaminated equipment; and (3) facilitate removal of radioactive waste and contaminated materials at the time of decommissioning. Although 10 CFR 72.130 does not provide specific criteria for acceptance, the ISFSI must be designed for decommissioning. Therefore, the standardized NUHOMS system design has been reviewed against good nuclear engineering practices which include (1) means to control the spread of contamination and (2) a design which facilitates decontamination.

Section 72.30 of 10 CFR Part 72 defines the need for a decommissioning plan which includes financing. Such a plan, however, is not considered applicable to this review. The cost of decommissioning the ISFSI must be considered in the overall cost of decommissioning the reactor site. 10 CFR 72.236(i) requires that the cask be designed to facilitate decontamination to the extent practicable.

The standardized NUHOMS design places heavy reliance on the prevention of contamination on the outer surface of the DSC. If these levels are kept low, very little contamination will exist on the inner surfaces of the HSMs, and ease of decommissioning will be facilitated. Section 10.3.14 of the SAR specifies a limiting condition for operation (LCO) for smearable (non-fixed) surface contamination levels on the outer surface of the DSC. This specification states that smearable contamination levels shall be less than 36.5 Bq/100 cm^2 (2200 dpm/100 cm^2) (10^{-5} $\mu Ci/cm^2$) for beta-gamma emitters and 3.65 Bq/100 cm^2 (220 dpm/100 cm^2) (10^{-6} $\mu Ci/cm^2$) for alpha-emitting radionuclides. This specification corresponds to surface removable contamination limits in 10 CFR 71.87(i)(1).

The surveillance requirement for this LCO is to determine the contamination levels of the DSC by taking surface contamination surveys of the upper one foot of the DSC exterior while the DSC is in the transfer cask before making the first closure weld. This survey can be used as a representative sample of the DSC body. If the specified limits are exceeded, the annular space between the DSC and transfer cask will be flushed with demineralized water until the contamination levels are within these limits. By minimizing DSC contamination, the potential contamination of the internal surfaces of the HSM is kept to a minimum.

The design of the standardized NUHOMS system is based on the intended eventual disposal of each DSC following fuel removal. However, it is also possible that the DSC shell/basket assembly could be reused. Such an alternative would be dependent on economic and regulatory conditions at the time of fuel removal.

At this time, it is not known whether demolition and removal of the HSM can be performed by conventional methods. Uncertainty exists with respect to (1) the specific levels of contamination that might exist on the inner surfaces of the HSM and (2) contamination level criteria which will govern whether the HSMs can be disposed of as low-level radioactive waste or as ordinary rubble. The staff also notes that decommissioning of the DSC's, transfer cask, and other equipment are matters which will be properly addressed in site-specific decommissioning plans.

9.3 Conclusions

The staff concludes that adequate attention has been paid to decommissioning in the design of the standardized NUHOMS system considering the current state of knowledge.

The staff also acknowledges that decommissioning considerations are sometimes in conflict with other requirements. The reinforced structure of the HSM, for example, will require considerable effort to demolish. Although it is not likely that significant contamination can spread beyond the DSC, demolition of the HSM may generate slightly contaminated dust. However, the staff concurs that primary concern in such cases rests with operational safety considerations, and ease of decommissioning is a secondary consideration.

A specification is proposed by the vendor for maximum DSC exterior surface contamination in SAR Section 10.3.14. The primary reason for requiring a clean exterior surface of the DSC is to reduce the total amount of activity as a source of potential contamination for the TC and HSM interior surfaces. The SER includes this condition for system use in Section 12.2.12 of this report.

10.0 QUALITY ASSURANCE

Chapter 11.0,[1] "Quality Assurance," of Revision 2 of the Pacific Nuclear Fuel Services Group Certification Safety Analysis Report (SAR) for a general license in accordance with Subpart K of 10 CFR Part 72 describes the PNFS quality assurance program. The PNFS quality assurance program is applied to structures, systems, and components of the NUHOMS independent spent fuel storage system important to safety. Chapter 11.0 addresses each of the 18 quality assurance criteria of 10 CFR Part 72, Subpart G, "Quality Assurance," and it includes the commitment that PNFS will implement the quality assurance program controls described in Revision 1 of the VECTRA Quality Assurance Manual dated July 22, 1994. This manual has been reviewed and accepted by the NRC.

Chapter 11.0 of the SAR describes the graded quality assurance program that is applied by PNFS to the structures, systems, and components of the NUHOMS spent fuel storage system based on that structure, system, or component's importance to safety. Chapter 11 defines three quality categories (or levels of quality/quality assurance) for items important to safety, and there are some items that are not important to safety. Chapter 11 of the SAR describes the differences between the quality assurance program for each category. It also lists the quality category of each structure, system, and component of the NUHOMS spent fuel storage system.

The staff has reviewed PNFS's quality assurance program description given and referenced in Chapter 11.0 of the SAR.[2] The staff finds that the PNFS commitments meet the requirements of Subpart G of 10 CFR Part 72 and are, therefore, acceptable for the issuance of a general Certificate of Compliance in accordance with Subpart L of 10 CFR Part 72.

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1. SAR pages 11.1-1 through 11.3-5 identified as NUH-003, Revision 2, November 5, 1993.

2. The acceptance criteria for quality assurance for independent spent fuel storage installations, based on Subpart G of 10 CFR Part 72, is given in the Fuel Cycle Safety Branch (Currently the Storage and Transport Systems Branch) Technical Position of June 20, 1986.

11.0 OPERATIONS, MAINTENANCE, TESTING, AND RECORDS

11.1 Operations

10 CFR 72.234(f) requires as a condition of approval of the Certificate of Compliance that: "the cask vendor [PNFS] shall ensure that written procedures and appropriate tests are established before use of the casks. A copy of these procedures and tests must be provided to each cask user." Regulatory Guide 3.48, Section 9 (Reference 5) describes the information to be incorporated in operating procedures for loading, unloading, and preparation of the cask. For the Certificate of Compliance for the standardized NUHOMS system, the term "cask" in 10 CFR Part 72, Subpart L, and in Regulatory Guide 3.48 is applied to the full NUHOMS System.

Procedures described in the SAR were reviewed and evaluated as part of the staff preparation of this SER. Procedures for loading the DSC are in SAR paragraphs 5.1.1.2 through 5.1.1.6 and are summarized graphically in SAR Figure 5.1-1. These include descriptions of recommended procedures for loading, use of fluids to fill cavities, removal of moisture, sealing, on-site management, and placing and sealing in storage positions.

Procedures for unloading the DSC are in SAR paragraphs 5.1.1.8 and 5.1.1.9 and are graphically summarized in SAR Figure 5.1-2. These include descriptions, tests, special preparations for unloading, unsealing, removal of DSC and on-site transfer, opening, removal of IFAs and cask decontamination.

Procedures for preparation of the TC and DSC for use are in SAR paragraph 5.1.1.1 and are part of the graphical summarization in SAR Figure 5.1-1. These include descriptions of inspections, tests, and special preparations of the TC and DSC necessary to ensure that they are properly loaded, closed, decontaminated, and transferred.

Staff review of the procedures included in the SAR indicates that they are acceptable and are in full compliance with the requirements of 10 CFR 72.234(f) for written procedures and with the guidance of Regulatory Guide 3.48, Section 9, for descriptions of operating procedures. These descriptions provide adequate bases for users to develop more detailed written procedures to follow during cask operations.

11.2 Maintenance

10 CFR 72.234 (a) requires as conditions of approval of the Certificate of Compliance that maintenance must comply with the requirements of 10 CFR 72.236 which require that the "cask must be designed to store spent fuel safely for a minimum of 20 years and permit maintenance as required." Regulatory Guide 3.48, Section 9.4, provides guidance on description of the maintenance program. The staff based evaluation of the SAR descriptions of maintenance on the 10 CFR Part 72, Subpart L, requirements and Regulatory Guide 3.48 guidance.

The SAR describes maintenance for the standardized NUHOMS system in paragraph 5.1.3.5 which states that, as the system is totally passive, it does not require maintenance. To ensure that the ventilation airflow is not interrupted, the HSM is to be periodically inspected to ensure that no debris is in the airflow inlet or outflow openings. SAR Section 5.1.1.7 describes these monitoring operations.

The TC is expected to be maintained and prepared in accordance with the procedure for each DSC IFA loading, transfer, loading into the HSM cycle, and for the unloading process. A single TC may be used at a site. SAR Section 4.5 describes recommended procedures for inspection, maintenance, and repair of the TC.

Staff review of the provisions for and descriptions of maintenance included in the SAR indicates that they are acceptable and are in full compliance with the requirements of 10 CFR Part 72, Subpart L, and the guidance of Regulatory Guide 3.48.

11.3 Testing

10 CFR 72.234(a) requires as conditions of approval of the Certification of Compliance that testing must comply with the requirements of 10 CFR 72.236. 10 CFR 72.234(f) requires that the cask vendor ensure appropriate tests are established before use of the "casks" and that a copy of the tests must be provided to each user. 10 CFR 72.236(j) requires that the "cask must be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, and other defects that could significantly reduce its confinement effectiveness." 10 CFR 72.236(l) requires that "the cask and its systems important to safety must be evaluated by appropriate test or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions." Regulatory Guide 3.48, Section 9, provides guidance on describing tests in the SAR.

Descriptions of and requirements for testing in conjunction with fabrication of the standardized NUHOMS system components are included in the SAR. Descriptions of tests and inspections associated with preparation for loading and loading operations are included in SAR Sections 5.1.1.1 through 5.1.1.6. Inspections in conjunction with downloading operations are described in SAR Section 5.1.1.9. Instruments to be used during loading operations and their functions are listed in SAR Table 5.1-1. No instruments or control systems are used during the storage cycle due to the passive nature of the standardized NUHOMS system (SAR Section 5.4).

Recommended pre-operational testing is described in SAR Section 9.2. This includes the test program description and discussion. Recommended testing is also included in SAR Section 10, Operating Controls and Limits. These include:

- DSC pressure during drying and backfill (SAR Sections 10.3.2 and 10.3.3).
- Tests of DSC inner seal and closure welds (SAR Sections 10.3.4 and 10.3.5).

- HSM dose rates with DSC in storage (SAR Section 10.3.7).
- HSM temperature rise with DSC in place (SAR Section 10.3.8).
- TC dose rates (SAR Section 10.3.12).
- DSC surface contamination (SAR Section 10.3.14).
- Ambient temperatures before TC use for DSC transport (SAR Section 10.3.15).

Staff review of the provisions for and descriptions of testing included in the SAR indicates that they are acceptable and are in full compliance with the requirements of 10 CFR Part 72, Subpart L, and the guidance of Regulatory Guide 3.48.

11.4 Records

10 CFR 72.234(d) requires that the cask vendor ensure a record is established and maintained for each cask fabricated under the NRC Certificate of Compliance and describe the information to be included on the record. The SAR does not explicitly identify the information record specified in 10 CFR 72.234(d). The only statement is that, "The ISFSI records should be maintained by the licensee in accordance with the requirements in 10 CFR Part 72 and with the existing plant records retention practices." As the 10 CFR 72.234(d) requirement is placed on the vendor, regardless of the location where the records are maintained, the staff considers that the required assurance is not met in the SAR.

The required record does not require any data from the "cask" user other than name and address. The remaining data relates to cask fabrication and the Certificate of Compliance. Specifically, the data required to be recorded and maintained for each "cask" by the vendor VECTRA are [10 CFR 72.234(d)(2)]:

"(i) The NRC Certificate of Compliance number;
(ii) The cask model number; [The model number should be marked on each HSM, DSC and TC.]
(iii) The cask identification number; [A unique identification number should be marked on each HSM, DSC and TC.]
(iv) Date fabrication was started;
(v) Date fabrication was completed;
(vi) Certification that the cask was designed, fabricated, tested, and repaired in accordance with a quality assurance program accepted by NRC;
(vii) Certification that inspections required by paragraph 72.236(j) were performed and found satisfactory; and
(viii) The name and address of the cask user."

The data marked on the DSC and TC are among those required by 10 CFR 72.236(k) which requires that the data be conspicuously and durably marked and also include the empty weight. The staff considers that the HSM and its included DSC support assembly are important to safety; therefore, maintaining a record and marking the individual HSM would be consistent with the intent of Subpart L.

12.0 CONDITIONS FOR SYSTEM USE

This section presents the conditions which a potential user (general licensee) of the standardized NUHOMS system must comply with, in order to use the system under a general license that is issued according to the provisions of 10 CFR 72.210 and 10 CFR 72.212. These conditions have either been proposed by the system vendor, imposed by the NRC staff as a result of the review of the SAR, or are part of the regulatory requirements expressed in 10 CFR 72.212.

12.1 General Requirements and Conditions

12.1.1 Regulatory Requirements for a General License

Subpart K of 10 CFR Part 72 contains conditions for using the general license to store spent fuel at an independent spent fuel storage installation at power reactor sites authorized to possess and operate nuclear power reactors under 10 CFR Part 50. Technical regulatory requirements for the licensee (user of the standardized NUHOMS system) are contained in 10 CFR 72.212(b).

10 CFR 72.212(b)(2) requires that the licensee perform written evaluations, before use, that establish that: (1) conditions set forth in the Certificate of Compliance have been met; (2) cask storage pads and areas have been designed to adequately support the static load of the stored casks; and (3) the requirements of 10 CFR 72.104 "Criteria for radioactive materials in effluent and direct radiation from an ISFSI or MRS," have been met. 10 CFR 72.212(b)(3) requires that the licensee review the SAR and the associated SER, before use of the general license, to determine whether or not the reactor site parameters (including earthquake intensity and tornado missiles), are encompassed by the cask design bases considered in these reports.

10 CFR 72.212(b)(4) provides that, as a holder of a Part 50 license, the user, before use of the general 10 CFR Part 72 license, must determine whether activities related to storage of spent fuel involve any unreviewed safety issues, or changes in technical specifications as provided under 10 CFR 50.59. Under 10 CFR 72.212(b)(5), the general license holder shall also protect the spent fuel against design basis threats and radiological sabotage pursuant to 10 CFR 73.55. Other general license requirements dealing with review of reactor emergency plans, quality assurance program, training, and radiation protection program must also be satisfied pursuant to 10 CFR 72.212(b)(6). 10 CFR 72.212(b)(7), (8), (9) and (10) describe record and procedural requirements for the general license holder.

Without limiting the requirement identified above, site-specific parameters and analyses, identified in the SER, that will need verification by the system user, are as a minimum, as follows:

1. The temperature of 21°C (70°F) as the maximum average yearly temperature with solar incidence. The average daily ambient temperature shall be 37.8°C (100°F) or less (Reference SER Section 2.4.1).

2. The temperature extremes of 52°C (125°F) with incident solar radiation and -40°C (-40°F) with no solar incidence (Reference SER Section 2.4.1) for storage of the DSC inside the HSM.

3. The horizontal and vertical seismic acceleration levels of 0.25g and 0.17g, respectively (Reference SER Table 2-4).

4. The analyzed flood condition of 4.6 m/s (15 fps) water velocity and a height of 15.2 m (50 feet) of water (full submergence of the loaded HSM DSC) (Reference SER Table 2-4).

5. The potential for fire and explosion should be addressed, based on site-specific considerations (See SER Table 2-4 and related SER discussion).

6. The HSM foundation design criteria are not included in the SAR. Therefore, the nominal SAR design or an alternative should be verified for individual sites in accordance with 10 CFR 72.212(b)(2)(ii). Also, in accordance with 10 CFR 72.212(b)(3), the foundation design should be evaluated against actual site parameters to determine whether its failure would cause the Standardized NUHOMS systems to exceed the design basis accident conditions.

7. The potential for lightning damage to any electrical system associated with the standardized NUHOMS system (e.g., thermal performance monitoring) should be addressed, based on site-specific considerations (See SER Table 2.4 and related SER discussion).

8. Any other site parameters or consideration that could decrease the effectiveness of csk systems important to safety.

In accordance with 10 CFR 72.212(b), a record of the written evaluations must be retained by the licensee until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

12.1.2 Operating Procedures

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The operating procedures suggested generically in the SAR were considered appropriate as discussed in Section 11.0 of the SER and should provide the basis for the user's written operating procedure. The following additional procedure requested by NRC staff in Section 11.1 should be part of the user operating procedures:

If fuel needs to be removed from the DSC, either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of damaged or oxidized fuel and to prevent radiological exposure to personnel during this operation. This can be achieved with this design by the use of the purge and fill valves which permit a determination of the atmosphere within the DSC before the removal of the inner top cover plate and shield plugs, prior to filling the DSC cavity with borated water (see SAR paragraph 5.1.1.9). If the atmosphere within the DSC is helium, then operations should proceed normally with fuel removal either via the transfer cask or in the pool. However, if air is present within the DSC, then appropriate filters should be in place to preclude the uncontrolled release of any potential airborne radioactive particulate from the DSC via the purge-fill valves. This will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection in the form of respirators or supplied air should be considered in accordance with the licensee's Radiation Protection Program.

12.1.3 Quality Assurance

Activities at the ISFSI shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 50, Appendix B and which is established, maintained, and executed with regard to the ISFSI.

12.1.4 Heavy Loads Requirements

Lifts of the DSC in the TC must be made within the existing heavy loads requirements and procedures of the licensed nuclear power plant. The TC design has been reviewed under 10 CFR Part 72 and found to meet NUREG-0612 (Reference 14) and ANSI N14.6 (Reference 8). However, an additional safety review (under 10 CFR 50.59) is required to show operational compliance with NUREG-0612 and/or existing plant-specific heavy loads requirements.

12.1.5 Training Module

A training module shall be developed for the existing licensee's training program establishing an ISFSI training and certification program. This module shall include the following:

1. Standardized NUHOMS System Design (overview);
2. ISFSI Facility Design (overview);
3. Certificate of Compliance conditions (overview);
4. Fuel Loading, Transfer Cask Handling, DSC Transfer Procedures; and
5. Off-Normal Event Procedures.

12.1.6 Pre-Operational Testing and Training Exercise

A dry run of the DSC loading, TC handling and DSC insertion into the HSM shall be held. This dry run shall include, but not be limited to, the following:

1. Functional testing of the TC with lifting yokes to ensure that the TC can be safely transported over the entire route required for fuel loading, washdown pit and trailer loading.

2. DSC loading into the TC to verify fit and TC/DSC annulus seal.

3. Testing of TC on transport trailer and transported to ISFSI along a predetermined route and aligned with an HSM.

4. Testing of transfer trailer alignment and docking equipment. Testing of hydraulic ram to insert a DSC loaded with test weights into an HSM and then retrieve it.

5. Loading a mock-up fuel assembly into the DSC.

6. DSC sealing, vacuum drying, and cover gas backfilling operations (using a mock-up DSC).

7. Opening a DSC (using a mock-up DSC).

8. Returning the DSC and TC to the spent fuel pool.

12.1.7 Special Requirements for First System in Place

The heat transfer characteristics of the cask system will be confirmed by temperature measurements of the first DSC placed in service. The first DSC shall be loaded with 24 fuel assemblies, constituting a source of approximately 24 kW. The DSC shall be loaded into the HSM and the thermal performance will be assessed by measuring the air inlet and outlet temperatures for normal airflow. Details for obtaining the measurements are provided in Section 12.2.8, under "Surveillance."

A letter report summarizing the results of the measurements shall be submitted to the NRC for evaluation and assessment of the heat removal characteristics of the thermal design within 30 days of placing the DSC in service, in accordance with 10 CFR 72.4.

Should the first user of the system not have fuel capable of producing a 24 kW heat load, or be limited to a lesser heat load, as in the case of BWR fuel, the user may use a lesser load for the process, provided that a calculation of the temperature difference between the inlet and outlet temperatures is performed, using the same methodology and inputs documented in the SAR, with lesser load as the only exception. The calculation and the measured temperature data shall be reported to the NRC in accordance with 10 CFR 72.4. The calculation and comparison need not be reported to the NRC for DSCs that are subsequently loaded with lesser loads than the initial case. However, for the first or any other user, the process needs to be performed and reported for any higher heat sources, up to 24 kW for PWR fuel and 19 kW for BWR fuel, which is the maximum allowed under the Certificate of Compliance. The NRC will also accept the use of artificial thermal loads other than spent fuel, to satisfy the above requirement.

12.1.8 Surveillance Requirements Applicability

The specified frequency for each Surveillance Requirement is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.

For frequencies specified as "once," the above interval extension does not apply.

If a required action requires performance of a surveillance or its completion time requires period performance of "once per...," the above frequency extension applies to the repetitive portion, but not to the initial portion of the completion time.

Exceptions to these requirements are stated in the individual specifications.

12.2 Technical Specifications, Functional and Operating Limits

12.2.1 Fuel Specification

Limit/Specification:	The characteristics of the spent fuel which is allowed to be stored in the standardized NUHOMS system are limited by those included in Tables 12-1a and 12-1b.
Applicability:	The specification is applicable to all fuel to be stored in the standardized NUHOMS system.
Objective:	The specification is prepared to ensure that the peak fuel rod temperatures, maximum surface doses, and nuclear criticality effective neutron multiplication factor are below the design values. Furthermore, the fuel weight and type ensures that structural conditions in the SAR bound those of the actual fuel being stored.
Action:	Each spent fuel assembly to be loaded into a DSC shall have the parameters listed in Tables 12-1a and 12-1b verified and documented. Fuel not meeting this specification shall not be stored in the standardized NUHOMS system.

Surveillance: Immediately, before insertion of a spent fuel assembly into an DSC, the identity of each fuel assembly shall be independently verified and documented.

Bases: The specification is based on consideration of the design basis parameters included in the SAR and limitations imposed as a result of the staff review. Such parameters stem from the type of fuel analyzed, structural limitations, criteria for criticality safety, criteria for heat removal, and criteria for radiological protection. The standardized NUHOMS system is designed for dry, horizontal storage of irradiated light water reactor (LWR) fuel. The principal design parameters of the fuel to be stored can accommodate standard PWR fuel designs manufactured by Babcock and Wilcox, Combustion Engineering, and Westinghouse, and standard BWR fuel manufactured by General Electric and is limited for use to these standard designs. The analyses presented in the SAR are based on non-consolidated, zircaloy-clad fuel with no known or suspected gross cladding breaches (See Tables 12-1a and 1b.)

The physical parameters that define the mechanical and structural design of the HSM and the DSC are the fuel assembly dimensions and weight. The calculated stresses given in this SER are based on the physical parameters given in Table 12-1a,b and represent the upper bound.

The design basis for nuclear criticality safety is based on the standard Babcock & Wilcox 15x15/208 pin fuel assemblies with initial enrichments up to 4.0 wt.% U-235, and General Electric 7x7 fuel assemblies with initial enrichments up to 4.0 wt.% U-235, for the standardized NUHOMS-24P and NUHOMS-52B designs, respectively. The HSM is designed to permit storage of irradiated fuel such that the irradiated fuel reactivity is less than or equal to 1.45 wt.% U-235 equivalent unirradiated fuel for the NUHOMS-24P design, and less than or equal to 4.0 wt.% U-235 initial enrichment fuel for the NUHOMS-52B design.

The thermal design criterion of the fuel to be stored is that the maximum heat generation rate per assembly be such that the fuel cladding temperature is maintained within established limits during normal and off-normal conditions. Fuel cladding temperature limits were established by the applicant based on methodology in PNL-6189 and PNL-4835 (References 47, 48). Based on this methodology, the staff has accepted that a maximum heat generation rate of 1 kW per assembly is a bounding value for the PWR fuel to be stored, and that 0.37 kW per assembly is a bounding value for the BWR fuel to be stored.

The radiological design criterion is that the gamma and neutron source strength of the irradiated fuel assemblies must be bounded by values of the neutron and gamma ray source strengths used by the vendor in the shielding analysis. The design basis source strengths were derived from a burnup analysis for (1) PWR fuel with 4.0 weight percent U-235 initial enrichment, irradiated to a maximum of 40,000 MWD/MTU, and a post irradiation time of five years; and (2) BWR fuel with 4.0 weight percent U-235 initial enrichment, irradiated to a maximum of 35,000 MWD/MTU, and a post irradiation time of 5 years.

Table 12-1a PWR Fuel Specifications of Fuel to be Stored in the Standardized NUHOMS-24P DSC(1)

Title or Parameter	Specifications
Fuel	Only intact, unconsolidated PWR fuel assemblies with the following requirements
Physical Parameters	
Assembly Length	See SAR Chapter 3
Nominal Cross-Sectional Envelope	See SAR Chapter 3
Maximum Assembly Weight	See SAR Chapter 3(2)
No. of Assemblies per DSC	≤24 intact assemblies
Fuel Cladding	Zircaloy-clad fuel with no known or suspected gross cladding breaches
Thermal Characteristics	
Decay Heat Power per Fuel Assembly	≤1.0 kW (this value is maximum for any given assembly, and may not be averaged for all 24 assemblies)
Radiological Characteristics	
Burnup	≤40,000 MWD/MTU
Post Irradiation Time	≥5 years
Maximum Initial Enrichment	≤4.0 wt. % U-235
Maximum Initial Uranium Content	≤472 kg/assembly
Maximum Initial Equivalent Enrichment	≤1.45 wt. % U-235(3)
Neutron Source Per Assembly	≤2.23E8 n/sec with spectrum bounded by that in Chapter 7 of SAR
Gamma Source Per Assembly	≤7.45E15 photon/sec with spectrum bounded by that in Chapter 7 of SAR

(1) The limiting fuel specifications listed above must be met by every individual fuel assembly to be stored in the standardized NUHOMS-24P system. Any deviation constitutes an Unanalyzed Condition and Violation of the Certificate of Compliance.

(2) Design valid for fuel weights up to 762.8 kg (1,682 lb).

(3) Determined by the PWR fuel criticality acceptance curve shown in Figure 12.1.

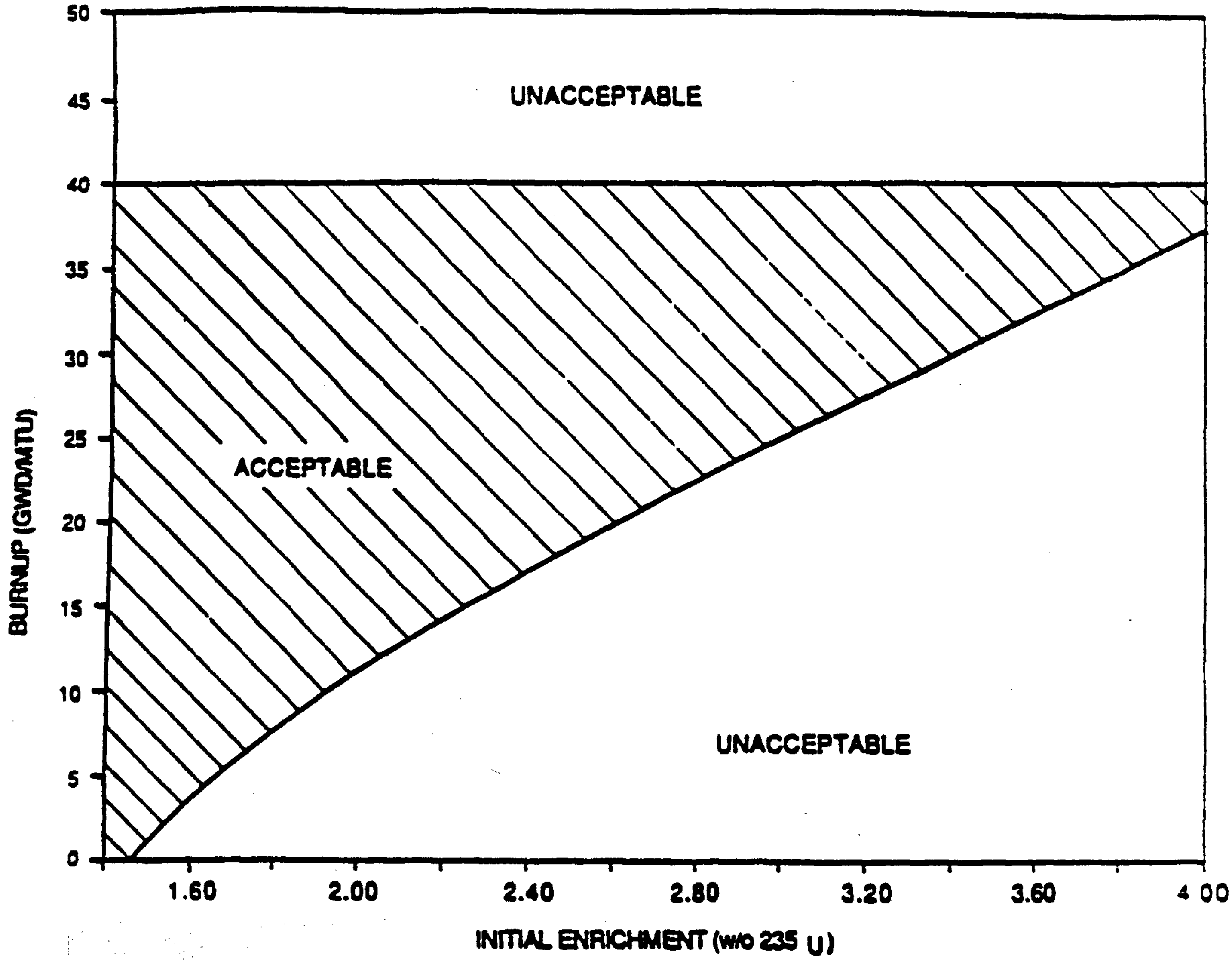

Figure 12.1

PWR Fuel Criticality Acceptance Curve

Table 12-1b BWR Fuel Specifications of Fuel to be Stored in the Standardized NUHOMS-52B DSC(1)

Title or Parameter	Specifications
Fuel	Only intact, unconsolidated BWR fuel assemblies with the following requirements
Physical Parameters	
Assembly Length	See SAR Chapter 3
Nominal Cross-Sectional Envelope	See SAR Chapter 3
Maximum Assembly Weight (w/fuel channels)	See SAR Chapter 3
No. of Assemblies per DSC	≤52 intact channeled assemblies
Fuel Cladding	Zircaloy-clad fuel with no known or suspected gross cladding breaches
Thermal Characteristics	
Decay Heat Power per Fuel Assembly	≤0.37 kW (this value is maximum for any given assembly, and may not be averaged for all 52 assemblies)
Radiological Characteristics	
Burnup	≤35,000 MWD/MTU
Post Irradiation Time	≥5 years
Maximum Initial Enrichment	≤4.0 wt. % U-235 (DSC with 0.75% borated neutron absorber plates)
Maximum Initial Uranium Content	≤198 kg/assembly
Neutron Source Per Assembly	≤1.01E8 n/sec with spectrum bounded by that in Chapter 7 of SAR
Gamma Source Per Assembly	≤2.63E15 photon/sec with spectrum bounded by that in Chapter 7 of SAR

(1) The limiting fuel specifications listed above must be met by every individual fuel assembly to be stored in the standardized NUHOMS-52B system. Any deviation constitutes an Unanalyzed Condition and Violation of the Certificate of Compliance.

12.2.2 DSC Vacuum Pressure During Drying

Limit/Specification:

Vacuum Pressure:	$\leq$ 0.4 kPa (3 mm Hg)
Time at Pressure:	$\geq$30 minutes following stepped evacuation
Number of Pump-Downs:	2

Applicability: This is applicable to all DSCs.

Objective: To ensure a minimum water content.

Action: If the required vacuum pressure cannot be obtained:

1. Confirm that the vacuum drying system is properly installed.
2. Check and repair, or replace, the vacuum pump.
3. Check and repair the system as necessary.
4. Check and repair the seal weld between the inner top cover plate and the DSC shell.

Surveillance: No maintenance or tests are required during normal storage. Surveillance of the vacuum gauge is required during the vacuum drying operation.

Bases: A stable vacuum pressure of 0.4 kPa ($\leq$3 mm Hg) further ensures that all liquid water has evaporated in the DSC cavity, and that the resulting inventory of oxidizing gases in the DSC is well below the 0.25 volume%.

12.2.3 DSC Helium Backfill Pressure

Limit/Specifications:

Helium 17.25 kPag (2.5 psig) ± 17.25 kPag (2.5 psig) backfill pressure (stable for 30 minutes after filling).

Applicability: This specification is applicable to all DSCs.

Objective: To ensure that: (1) the atmosphere surrounding the irradiated fuel is a non-oxidizing inert gas; (2) the atmosphere is favorable for the transfer of decay heat.

Action: If the required pressure cannot be obtained:

1. Confirm that the vacuum drying system and helium source are properly installed.

2. Check and repair or replace the pressure gauge.

3. Check and repair or replace the vacuum drying system.

4. Check and repair or replace the helium source.

5. Check and repair the seal weld on DSC top shield plug.

If pressure exceeds the criterion, release a sufficient quantity of helium to lower the DSC cavity pressure.

Surveillance: No maintenance or tests are required during the normal storage. Surveillance of the pressure gauge is required during the helium backfilling operation.

Bases: The value of 17.25 kPag (2.5 psig) was selected to ensure that the pressure within the DSC is within the design limits during any expected normal and off-normal operating conditions.

12.2.4 DSC Helium Leak Rate of Inner Seal Weld

Limit/Specification:

$\leq$ 1.0x10^{-2} kPa · cm^3/s (1.0 x 10^{-4} atm · cubic centimeters per second) (atm · cm^3/s) at the highest DSC limiting pressure.

Applicability: This specification is applicable to the inner top cover plate seal weld of all DSCs.

Objective:

1. To limit the total radioactive gases normally released by each canister to negligible levels. Should fission gases escape the fuel cladding, they will remain confined by the DSC confinement boundary.

2. To retain helium cover gases within the DSC and prevent oxygen from entering the DSC. The helium improves the heat dissipation characteristics of the DSC and prevents any oxidation of fuel cladding.

Action: If the leak rate test of the inner seal weld exceeds 1.0x10^{-2} kPa · cm^3/s (1.0x10^{-4} atm · cm^3/s):

1. Check and repair the DSC drain and fill port fittings for leaks.

2. Check and repair the inner seal weld.

3. Check and repair the inner top cover plate for any surface indications resulting in leakage.

Surveillance: After the welding operation has been completed, perform a leak test with a helium leak detection device.

Bases: If the DSC leaked at the maximum acceptable rate of 1.0x10^{-2} kPa · cm^3/s (1.0x10^{-4} atm · cm^3/s) for a period of 20 years, about 63,100 cc of helium would escape from the DSC. This is about 1% of the 6.3 x 10^6 cm^3 of helium initially introduced in the DSC. This amount of leakage would have a negligible effect on the inert environment of the DSC cavity. Reference: American National Standards Institute, ANSI N14.5-1987, "For Radioactive Materials—Leakage Tests on Packages for Shipment" (Appendix B3).

12.2.5 DSC Dye Penetrant Test of Closure Welds

Limit/Specification:

All DSC closure welds except those subjected to full volumetric inspection shall be dye penetrant tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section III, Division 1, Article NB-5000 (Reference 8.3 of SAR). The liquid penetrant test acceptance standards shall be those described in Subsection NB-5350 of the Code.

Applicability: This is applicable to all DSCs. The welds include inner and outer top and bottom covers, and vent and syphon port covers.

Objective: To ensure that the DSC is adequately sealed in a redundant manner and leak tight.

Action: If the liquid penetrant test indicates that the weld is unacceptable:

1. The weld shall be repaired in accordance with approved ASME procedures.

2. The new weld shall be re-examined in accordance with this specification.

Surveillance: During DSC closure operations. No additional surveillance is required for this operation.

Bases: Article NB-5000 Examination, ASME Boiler and Pressure Vessel Code, Section III, Division 1, Sub-Section NB (Reference 8.3 of SAR).

12.2.6 DSC Top End Dose Rates

Limit/Specification:

Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 2 mSv/hr (200 mrem/hr) at top shield plug surface at centerline with water in cavity.

b. 4 mSv/hr (400 mrem/hr) at top cover plate surface at centerline without water in cavity.

Applicability: This specification is applicable to all DSCs.

Objective: The dose rate is limited to this value to ensure that the DSC has not been inadvertently loaded with fuel not meeting the specifications in Section 12.2.1 of the SER and to maintain dose rates as low as is reasonably achievable during DSC closure operations.

Action:

a. If specified dose rates are exceeded, the following actions should be taken:

   1. Confirm that the spent fuel assemblies placed in DSC conform to the fuel specifications of Section 12.2.1.

   2. Visually inspect placement of top shield plug. Re-install or adjust position of top shield plug.

   3. Install additional temporary shielding.

b. Submit a letter report to the NRC within 30 days summarizing the action taken and the results of the surveillance, investigation and findings. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: Dose rates shall be measured before seal welding the inner top cover plate to the DSC shell and welding the outer top cover plate to the DSC shell.

Basis: The basis for this limit is the shielding analysis presented in Section 7.0 of the SAR.

12.2.7 HSM Dose Rates

Limit/Specification:

Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 4 mSv/hr (400 mrem/hr) at 1 m (3 feet) from the HSM surface.

b. Outside of HSM door on centerline of DSC 1 mSv/hr (100 mrem/hr).

c. End Shield wall exterior 0.2 mSv/hr (20 mrem/hr).

Applicability: This specification is applicable to all HSMs which contain a loaded DSC.

Objective: The dose rate is limited to this value to ensure that the cask (DSC) has not been inadvertently loaded with fuel not meeting the specifications in Section 12.2.1 of the SER and to maintain dose rates as low as is reasonably achievable at locations on the HSMs where surveillance is performed, and to reduce off-site exposures during storage.

Action:

a. If specified dose rates are exceeded, the following actions should be taken:

   1. Ensure that the DSC is properly positioned on the support rails.

   2. Ensure proper installation of the HSM door.

   3. Ensure that the required module spacing is maintained.

   4. Confirm that the spent fuel assemblies contained in the DSC conform to the specifications of Section 12.2.1.

   5. Install temporary or permanent shielding to mitigate the dose to acceptable levels in accordance with 10 CFR Part 20, 10 CFR 72.104(a), and ALARA.

b. Submit a letter report to the NRC within 30 days summarizing the action taken and the results of the surveillance, investagion and findings. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: The HSM and ISFSI shall be checked to verify that this specificantion has been met after the DSC is placed into storage and the HSM door is closed.

Basis: The basis for this limit is the shielding analysis presented in Section 7.0 of the SAR. The specified dose rates provide as-low-as-is-reasonably-achievable on-site and off-site doses in accordance with 10 CFR Part 20 and 10 CFR 72.104(a).

12.2.8 HSM Maximum Air Exit Temperature

Limit/Specification:

Following initial DSC transfer to the HSM or the occurrence of accident conditions, the equilibrium air temperature difference between ambient temperature and the vent outlet temperature shall not exceed 37.8°C (100°F) for ≥5 year cooled fuel, when fully loaded with 24 kW heat.

Applicability: This specification is applicable to all HSMs stored in the ISFSI. If a DSC is placed in the HSM with a heat load less than 24 kW, the limiting difference between outlet and ambient temperatures shall be determined by a calculation performed by the user using the same methodology and inputs documents in the SAR and SER.

Objective: The objective of this limit is to ensure that the temperature of the fuel cladding and the HSM concrete do not exceed the temperatures calculated in Section 8 of the SAR. That section shows that if the air outlet temperature difference is less than or equal to 37.8°C (100°F) (with a thermal heat load of 24 kW), the fuel cladding and concrete will be below the respective temperature limits for normal long-term operation.

Action: If the temperature rise is greater than that specified, then the air inlets and exits should be checked for blockage. If the blockage is cleared and the temperature is still greater than that specified, the DSC and HSM cavity may be inspected using video equipment or other suitable means. If environmental factors can be ruled out as the cause of excessive temperatures, then the fuel bundles are producing heat at a rate higher than the upper limit specified in Section 3 of the SAR and will require additional measurements and analysis to assess the actual performance of the system. If excessive temperatures cause the system to perform in an unacceptable manner and/or the temperatures cannot be controlled to acceptable limits, then the cask shall be unloaded.

Surveillance: The temperature rise shall be measured and recorded daily following DSC insertion until equilibrium temperature is reached, 24 hours after insertion, and again on a daily basis after insertion into the HSM or following the occurrence of accident conditions. If the temperature rise is within the specifications or the calculated value for a heat load less than 24 kW, then the HSM and DSC are performing as designed to meet this specification and no further maximum air exit temperature measurements are required. Air temperatures must be measured in such a manner as to obtain representative values of inlet and outlet air temperatures.

Basis: The specified temperature rise is selected to ensure the fuel clad and concrete temperatures are maintained at or below acceptable long-term storage limits.

12.2.9 Transfer Cask Alignment with HSM

Limit/Specification:

The cask must be aligned with respect to the HSM so that the longitudinal centerline of the DSC in the transfer cask is within ± 0.3 cm ($\pm 1/8$ inch) of its true position when the cask is docked with the HSM front access opening.

Applicability: This specification is applicable during the insertion and retrieval of all DSCs.

Objective: To ensure smooth transfer of the DSC from the transfer cask to HSM and back.

Action: If the alignment tolerance is exceeded, the following actions should be taken:

a. Confirm that the transfer system is properly configured.

b. Check and repair the alignment equipment.

c. Confirm the locations of the alignment targets on the transfer cask and HSM.

Surveillance: Before initiating DSC insertion or retrieval, confirm the alignment. Observe the transfer system during DSC insertion or retrieval to ensure that motion or excessive vibration does not occur.

Basis: The basis for the true position alignment tolerance is the clearance between the DSC shell, the transfer cask cavity, the HSM access opening, and the DSC support rails inside the HSM.

12.2.10 DSC Handling Height Outside the Spent Fuel Pool Building

Limit/Specification:

1. The loaded TC/DSC shall not be handled at a height greater than 203 cm (80 inches) outside the spent fuel pool building.

2. In the event of a drop of a loaded TC/DSC from a height greater than 38 cm (15 inches) (a) fuel in the DSC shall be returned to the reactor spent fuel pool; (b) the DSC shall be removed from service and evaluated for further use; and (c) the TC shall be inspected for damage and evaluated for further use.

Applicability: The specification applies to handling the TC, loaded with the DSC, on route to, and at, the storage pad.

Objective:

1. To preclude a loaded TC/DSC drop from a height greater than 203 cm (80 inches).

2. To maintain spent fuel integrity, according to the spent fuel specification for storage, continued confinement integrity, and DSC functional capability, after a tip-over or drop of a loaded DSC from a height greater than 38 cm (15 inches).

Surveillance: In the event of a loaded TC/DSC drop accident, the system will be returned to the reactor fuel handling building, where, after the fuel has been returned to the spent fuel pool, the DSC and TC will be inspected and evaluated for future use.

Basis: The NRC evaluation of the TC/DSC drop analysis concurred that drops up to 203 cm (80 inches), of the DSC inside the TC, can be sustained without breaching the confinement boundary, preventing removal of spent fuel assemblies, or causing a criticality accident. This specification ensures that handling height limits will not be exceeded in transit to, or at the storage pad. Acceptable damage may occur to the TC, DSC, and the fuel stored in the DSC, for drops of height greater than 38 cm (15 inches). The specification requiring inspection of the DSC and fuel following a drop of 38 cm (15 inches) or greater ensures that the spent fuel will continue to meet the requirements for storage, the DSC will continue to provide confinement, and the TC will continue to provide its design functions of DSC transfer and shielding.

12.2.11 Transfer Cask Dose Rates

Limit/Specification:

Dose rates from the transfer cask shall be limited to levels which are less than or equal to:

a. 2 mSv/hr (200 mrem/hr) at 1 m (3 feet) with water in the DSC cavity.

b. 5 mSv/hr (500 mrem/hr) at 1 m (3 feet) without water in the DSC cavity.

Applicability: This specification is applicable to the transfer cask containing a loaded DSC.

Objective: The dose rate is limited to this value to ensure that the DSC has not been inadvertently loaded with fuel not meeting the specifications in Section 12.2.1 of the SER and to maintain dose rates as low as reasonably achievable during DSC transfer operations.

Action: If specified dose rates are exceeded, place temporary shielding around affected areas of transfer cask and review the plant records of the fuel assemblies which have been placed in DSC to ensure they conform to the fuel specifications of Section 12.2.1. Submit a letter report to the NRC within 30 days summarizing the action taken and the results of the surveillance, investigation and findings. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: The dose rates should be measured as soon as possible after the transfer cask is removed from the spent fuel pool.

Basis: The basis for this limit is the shielding analysis presented in Section 7.0 of the SAR.

12.2.12 Maximum DSC Removable Surface Contamination

Limit/Specification:

36.5 Bq/100 cm^2 (2,200 dpm/100 cm^2) for beta-gamma sources
3.65 Bq/100 cm^2 (220 dpm/100 cm^2) for alpha sources.

Applicability: This specification is applicable to all DSCs.

Objective: To ensure that release of non-fixed contamination above accepted limits does not occur.

Action: If the required limits are not met:

a. Flush the DSC/transfer cask annulus with demineralized water and repeat surface contamination surveys of the DSC upper surface.

b. If contamination of the DSC cannot be reduced to an acceptable level by this means, direct surface cleaning techniques shall be used following removal of the fuel assemblies from the DSC and removal of the DSC from the transfer cask.

c. Check and replace the DSC/transfer cask annulus seal to ensure proper installation and repeat canister loading process.

Surveillance: Following placement of each loaded DSC/transfer cask into the cask decontamination area, fuel pool water above the top shield plug shall be removed and the top region of the DSC and cask shall be decontaminated. A contamination survey of the upper 0.3 m (1 foot) of the DSC and cask shall be taken. In addition, contamination surveys shall be taken on the inside surfaces of the TC after the DSC has been transferred into the HSM. If the above surface contamination limit is exceeded, the TC shall be decontaminated.

Basis: This non-fixed contamination level is consistent with the requirements of 10 CFR 71.87(i)(1) and 49 CFR 173.443, which regulate the use of spent fuel shipping containers. Consequently, these contamination levels are considered acceptable for exposure to the general environment. This level will also ensure that contamination levels of the inner surfaces of the HSM and potential releases of radioactive material to the environment are minimized.

12.2.13 TC/DSC Lifting Heights as a Function of Low Temperature and Location

Limit/Specification:

1. No lifts or handling of the TC/DSC at any height are permissible at DSC basket temperatures below -28.9°C (-20°F) inside spent fuel pool building.

2. The maximum lift height of the TC/DSC shall be 203 cm (80 inches) if the basket temperature is below -17.8°C (0°F) but higher than -28.9°C (-20°F) inside the spent fuel pool building.

3. No lift height restriction is imposed on the TC/DSC if the basket temperature is higher than -17.8°C (0°F) inside the spent fuel pool building.

4. The maximum lift height and handling height for all transfer operations outside the spent fuel pool building shall be 203 cm (80 inches) and the basket temperature may not be lower than -17.8°C (0°F).

Applicability: These temperature and height limits apply to lifting and transfer of all loaded TC/DSCs inside and outside the spent fuel pool building. 10 CFR Part 72 applies outside the spent fuel pool building and 10 CFR Part 50 applies inside the spent fuel pool building.

Objective: The low temperature and height limits are imposed to ensure that brittle fracture of the ferritic steels, used in the TC trunnions and shell and in the DSC basket, does not occur during transfer operations.

Action: Confirm the basket temperature before transfer of the TC. If no calculation or measurement of this value is available, then the ambient temperature may conservatively be used.

Surveillance: The ambient temperature shall be measured before transfer of the TC/DSC.

Bases: The basis for the low temperature and height limits is ANSI N14.6-1986 paragraph 4.2.6 (Reference 8) which requires at least 4.4°C (40°F) higher service temperature than nil ductility transition (NDT) temperature for the TC. In the case of the standardized TC, the test temperature is -40°C (-40°F); therefore, although the NDT temperature is not determined, the material will have the required 4.4° (40°F) margin if the ambient temperature is -17.8°C (0°F) or higher. This assumes the material service temperature is equal to the ambient temperature.

The basis for the low temperature limit for the DSC is NUREG/CR-1815. The basis for the handling height limits is the NRC evaluation of the structural integrity of the DSC to drop heights of 203 cm (80 inches) and less.

12.2.14 TC/DSC Transfer Operations at High Ambient Temperatures

Limit/Specification:
1. The ambient temperature for transfer operations of a loaded TC/DSC shall not be greater that 37.8°C (100°F) (when cask is exposed to direct insolation).
2. For transfer operations when ambient temperatures exceed 37.8°C (100°F) up to 52°C (125°F), a solar shield shall be used to provide protection against direct solar radiation.

Applicability: This ambient temperature limit applies to all transfer operations of loaded TC/DSCs outside the spent fuel pool building, the spent fuel pool building.

Objective: The high temperature limit 37.8°C (100°F) is imposed to ensure that:

1. The fuel cladding temperature limit is not exceeded.
2. The solid neutron shield material temperature limit is not exceeded, and
3. The corresponding TC cavity pressure limit is not exceeded.

Action: Confirm what the ambient temperature is and provide appropriate solar shade if ambient temperature is expected to exceed 37.8°C (100°F).

Surveillance: The ambient temperature shall be measured before transfer of the TC/DSC.

Bases: The basis for the high temperature limit is PNL-6189 for fuel clad limit, the manufacturer's specification for neutron shield and the design basis pressure of the TC internal cavity pressure.

12.2.15 Boron Concentration in the DSC Cavity Water (24-P Design Only)

Limit/Specification:

The DSC cavity shall be filled only with water having a boron concentration equal to, or greater than 2,000 ppm.

Applicability: This limit applies only to the standardized NUHOMS-24P design. No boration in the cavity water is required for the standardized NUHOMS-52B system since that system uses fixed absorber plates.

Objective: To ensure a subcritical configuration is maintained in the case of accidental loading of the DSC with unirradiated fuel.

Action: If the boron concentration is below the required weight percentage concentration (gm boron/10^6 gm water), add boron and re-sample, and test the concentration until the boron concentration is shown to be greater than that required.

Surveillance: Written procedures shall be used to independently determine (two samples analyzed by different individuals) the boron concentration in the water used to fill the DSC cavity.

1. Within 4 hours before insertion of the first fuel assembly into the DSC, the dissolved boron concentration in water in the spent fuel pool, and in the water that will be introduced in the DSC cavity, shall be independently determined (two samples chemically analyzed by two individuals).

2. Within 4 hours before flooding the DSC cavity for unloading the fuel assemblies, the dissolved boron concentration in water in the spent pool, and in the water that will be introduced into the DSC cavity, shall be independently determined (two samples analyzed chemically by two individuals).

3. The dissolved boron concentration in the water shall be reconfirmed at intervals not to exceed 48 hours until such time as the DSC is removed from the spent fuel pool or the fuel has been removed from the DSC.

Bases: The required boron concentration is based on the criticality analysis for an accidental misloading of the DSC with unburned fuel, maximum enrichment, and optimum moderation conditions.

12.2.16 Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building as a Function of Horizontal Acceleration and Loaded Cask Weight

Limit/Specification:	Seismic restraints shall be provided to prevent overturning of a loaded TC during a seismic event if a certificate holder determines that the horizontal acceleration is 0.40 g or greater and the fully loaded TC weight is less than 86,260 kg (190 kips). The determination of horizontal acceleration acting at the CG of the loaded TC must be based on a peak horizontal ground acceleration at the site, but shall not exceed 0.25 g.
Applicability:	This condition applies to all TCs which are subject to horizontal accelerations of 0.40 g or greater.
Objective:	To prevent overturning of a loaded TC inside the spent fuel pool building.
Action:	Determine what the horizontal acceleration is for the TC and determine if the cask weight is less than 86,260 kg (190 kips).
Surveillance:	Determine need for TC restraint before any operations inside the spent fuel pool building.
Bases:	Calculation of overturning and restoring moments.

12.3 Surveillance and Monitoring

Paragraph 10.2.3 of the SAR outlines a single surveillance requirement proposed by PNFS. However, as discussed below, there are many items subject to monitoring. The single item subject to surveillance is the HSM air inlet and outlet passages. They shall be inspected once every 4 days to ensure that they are clear of obstructions. The SER notes that this proposed surveillance frequency could result in exceeding the HSM concrete temperature limit of 177°C (350°F) for accident conditions of blocked inlets or outlets. The concrete temperature for this adiabatic heat-up will exceed 177°C (350°F) in approximately 40 hours. Furthermore, the maximum fuel clad temperature will be exceeded in a 5-day period. Although the vendor-proposed 4-day inspection frequency will prevent exceeding the fuel cladding temperature, the HSM would need to be removed from service if inlets or outlets are found to be substantially blocked, and it cannot be established that the blockage is less than 40 hours.

As a result of this situation, the NRC staff is requiring the following surveillance frequency for the HSM.

12.3.1 Visual Inspection of HSM Air Inlets and Outlets (Front Wall and Roof Birdscreen)

Limit/Surveillance: A visual surveillance of the exterior of the air inlets and outlets shall be conducted daily. In addition, a close-up inspection shall be performed to ensure that no materials accumulate between the modules to block the air flow.

Objective: To ensure that HSM air inlets and outlets are not blocked for more than 24 hours to prevent exceeding the allowable HSM concrete temperature or the fuel cladding temperature.

Applicability: This specification is applicable to all HSMs loaded with a DSC loaded with spent fuel.

Action: If the surveillance shows blockage of air vents (inlets or outlets), they shall be cleared. If the screen is damaged, it shall be replaced.

Basis: The concrete temperature could exceed 177°C (350°F) in the accident circumstances of complete blockage of all vents if the period exceeds approximately 40 hours. Concrete temperatures over 177°C (350°F) in accidents (without the presence of water or steam) can have uncertain impact on concrete strength and durability. A conservative analysis (adiabatic heat case) of complete blockage of all air inlets or outlets indicates that the concrete can reach the accident temperature limit of 177°C (350°F) in a time period of approximately 40 hours.

12.3.2 HSM Thermal Performance

Surveillance: Verify a temperature measurement of the thermal performance, for each HSM, on a daily basis. The temperature measurement could be any parameter such as (1) a direct measurement of the HSM temperatures, (2) a direct measurement of the DSC temperatures, (3) a comparison of the inlet and outlet temperature difference to predicted temperature differences for each individual HSM, or (4) 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. If air temperatures are measured, they must be measured in such a manner as to obtain representative values of inlet and outlet air temperatures. Also due to the proximity of adjacent HSM modules, care must be exercised to ensure that measured air temperatures reflect only the thermal performance of an individual module, and not the combined performance of adjacent modules.

Action: If the temperature measurement shows a significant unexplained difference, so as to indicate the approach of materials to the concrete or fuel clad temperature criteria, take appropriate action to determine the cause and return the canister to normal operation. If the measurement or other evidence suggests that the concrete accident temperature criteria 177°C (350°F) has been exceeded for more than 24 hours, the HSM must be removed from service unless the licensee can provide test results in accordance with ACI-349, appendix A.4.3, demonstrating that the structural strength of the HSM has an adequate margin of safety.

Basis: The temperature measurement should be of sufficient scope to provide the licensee with a positive means to identify conditions which threaten to approach temperature criteria for proper HSM operation and allow for the correction of off-normal thermal conditions that could lend to exceeding the concrete and fuel clad temperature criteria.

Table 12.3.1

Summary of Surveillance and Monitoring Requirements

Surveillance or Monitoring	Period	Reference Section
1. Fuel Specification	PL	12.2.1
2. DSC Vacuum Pressure During Drying	L	12.2.2
3. DSC Helium Backfill Pressure	L	12.2.3
4. DSC Helium Leak Rate of Inner Seal Weld	L	12.2.4
5. DSC Dye Penetrant Test of Closure Welds	L	12.2.5
6. DSC Top End Dose Rates	L	12.2.6
7. HSM Dose Rates	L	12.2.7
8. HSM Maximum Air Exit Temperature	24 hrs	12.2.8
9. TC Alignment with HSM	S	12.2.9
10. DSC Handling Height Outside Spent Fuel Pool Building	AN	12.2.10
11. Transfer Cask Dose Rates	L	12.2.11
12. Maximum DSC Surface Contamination	L	12.2.12
13. TC/DSC Lifting Heights as a Function of Low Temperature and Location	L	12.2.13

<u>Legend</u>

PL Prior to loading
L During loading and prior to movement to HSM pad
24 hrs Time following DSC insertion into HSM
S Prior to movement of DSC to or from HSM
AN As necessary
D Daily (24 hour frequency)

Table 12.3.1

Summary of Surveillance and Monitoring Requirements (Continued)

Surveillance or Monitoring	Period	Reference Section
14. TC/DSC Transfer Operations at High Ambient Temperatures	L	12.2.14
15. Boron Concentration in DSC Cavity Water (24-P Design Only)	PL	12.2.15
16. Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building as a Function of Horizontal Acceleration and Loaded Cask Weight	PL	12.2.16
17. Visual Inspection of HSM Air Inlets and Outlets	D	12.3.1
18. HSM Thermal Performance	D	12.3.2

<u>Legend</u>

PL Prior to loading
L During loading and prior to movement to HSM pad
24 hrs Time following DSC insertion into HSM
S Prior to movement of DSC to or from HSM
AN As necessary
D Daily (24 hour frequency)

13.0 REFERENCES

1. Pacific Nuclear Fuel Services, Inc., "Safety Analysis Report for the Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel," Docket No. 72-1004:

    a. NUH-003, Rev. 0, December 20, 1990.

    b. NUH-003, Rev. 1, September 25, 1991.

    c. NUH-003, Rev. 2, November 5, 1993.

    [Note: Where different submittals are at variance in addressing the same subject, the most recent submittal governs.]

2. Office of the Federal Register, Code of Federal Regulations, Title 10 - Energy, Chapter I - Nuclear Regulatory Commission (10 CFR), Revised as of January 1, 1993. Following parts referenced:

    a. 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

    b. 10 CFR Part 20, "Standards for Protection Against Radiation."

    c. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

    d. 10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the TR for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel Submitted by Nutech Engineers, Inc.," NUH-002, Rev. 1A, April 1989.

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