SAFETY EVALUATION REPORT RELATED TO THE TOPICAL REPORT FOR THE NUTECH HORIZONTAL MODULAR STORAGE SYSTEM FOR IRRADIATED NUCLEAR FUEL NUHOMS-24P SUBMITTED BY NUTECH ENGINEERS, INC.

U.S. Nuclear Regulatory Commission

Office of Nuclear Material Safety and Safeguards

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1.0 GENERAL DESCRIPTION

1.1 INTRODUCTION

1.1.1 Objective

This is a Safety Evaluation Report (SER), which documents the Nuclear Regulatory Commission (NRC) staff analysis and recommendations on the suitability and acceptability of the NUTECH "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS*-24P" (Reference 1), an independent spent fuel storage installation system (ISFSI), hereafter referred to as the Topical Report (TR). The TR uses the format of NRC Regulatory Guide 3.48 (Reference 2).

1.1.2 Scope

The review of the TR is oriented toward determining and justifying the extent that the TR can be used as a reference to satisfy the requirements of 10 CFR 72 (Reference 3) for an ISFSI license application. This use of the TR would be by inclusion by clear and specific reference or by repetition in the license application documents. The application contents and licensing documents, which would typically make maximum use of the TR, are the safety analysis report (SAR) (10 CFR 72.24) and the technical specifications (10 CFR 72.26), described in 10 CFR 72.44(c)).

The review also addresses the suitability of material contained in the TR to be incorporated by reference in the other documents required to be submitted with the license application, specifically: the decommissioning plan, emergency plan, environmental report, and quality assurance program. The review does not address potential reference in conjunction with the following licensing documents: physical security plan, design for physical protection, safeguards contingency plan, or personnel training program.

The review includes considerations of the appropriate parts of 10 CFR 20 for radiation protection during onsite handling, movement, and storage of spent fuel.

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* NUHOMS is a registered trademark of NUTECH Engineers, Inc.

The recommendations for approval of the NUTECH ISFSI system are limited to the level to which the system is defined. The drawings and descriptions in the TR do not constitute final construction drawings and specifications. However, except as otherwise indicated in the recommendations, the level of design and supporting rationale and analyses presented are adequate to permit the development of such designs and specifications following standard codes and practice, and sufficiently bound the final design as to not require further NRC detailed review.

This SER includes descriptions of the different functional elements of NUTECH ISFSI system; general design criteria; and evaluations of the designs, proposed operating procedures, proposed acceptance tests and maintenance program, radiological protection, decommissioning discussion, proposed operating controls and limits, and proposed quality assurance. In general, the SER has been prepared for use together with Reference 1. Figures, tables, and text of the TR are not repeated in the SER but are referenced, except where such repetition is considered essential for clarity of the SER.

The descriptions of the NUTECH ISFSI system included in Section 1.2 are for general orientation of the reviewer. The descriptions are believed to be accurate representations, but they did not form the basis for the detailed evaluations. The evaluation and recommendations are based directly on the contents of the TR (Reference 1).

1.1.3 Context

A topical report for an ISFSI constitutes a potential reference which may be cited in subsequent license applications to the NRC for permission to construct, own, use, and operate ISFSI at specific sites, or may be cited in subsequently submitted other topical reports. NRC action on a topical report may be approval, disapproval, or approval with limitations or other qualifications.

The principal use of an approved topical report in a license application is by inclusion (by repetition or specific reference) in the accompanying safety analysis report (SAR) and proposed technical specifications. Requirements for SAR are stated in 10 CFR 72.24.

Requirements for technical specifications are stated in 10 CFR 72.28 and 10 CFR 72.44. Incorporation of TR contents by reference is under the provisions of 10 CFR 72.18, which requires that those references are clear and specific. Designs and descriptions in the topical report, as it is approved, may be incorporated fully or partially. Changes and omitted material must however be fully addressed in the license application.

A topical report cannot constitute (by reference) all of an SAR or technical specification. Actual site conditions, procedures of the individual company making the application, and elements of a site's existing final safety analysis report (FSAR) also impact or must be addressed in the SAR. A topical report may provide a reference for all of the technical specification if all of the requirements of 10 CFR 72.26 and 10 CFR 72.44(c) are met and are specifically referenced in the technical specifications submitted with the license applications.

The format and content of an SAR generally follow the structure suggested by Regulatory Guide 3.48 where it is applicable. The NUTECH TR follows the same format. There is no specific requirement for the contents of a TR; it is the SAR that must be complete. TRs are reviewed to determine adequacy in meeting requirements for an SAR. TR elements become part of an SAR by reference or repetition. The TR is also reviewed for validity that the design of the included ISFSI systems meets the requirements for such systems. The requirements are principally as presented in 10 CFR 72, and as further implemented by Regulatory Guides 3.48 and 3.60 (Reference 4).

1.2 GENERAL DESCRIPTION OF NUHOMS-24P SYSTEM

The NUHOMS system is an ISFSI system that provides for horizontal, dry storage of irradiated nuclear fuel assemblies. The fuel assemblies are contained in a dry shielded canister (DSC) made of stainless steel and lead, which is transported within a heavily shielded transfer cask (TC), and which is placed inside a reinforced concrete horizontal storage module (HSM) for long term storage.

In addition to the DSC, TC, and HSM, the NUHOMS system also requires:

1. Handling and transfer equipment to load the DSC with fuel, to seal the DSC, to move the loaded DSC inside the TC from the fuel pool building to the HSM (elsewhere on the site), and to insert the DSC into the HSM; and

2. An infrastructure of procedures, interfaces with the host plant, personnel qualifications, organization, training, quality assurance, and support services. Figures 1.1 and 1.2 show schematically the major physical components and operations of the NUHOMS system.

The TR presents for review and approval a design in which the DSC holds 24 irradiated pressurized water reactor (PWR) fuel assemblies and in which the HSMs are arranged in back-to-back arrays. There may be any number of arrays; however, the overall exposure levels are dependent on the actual number of arrays, and must therefore be checked in any license application.

The designs of the HSM, DSC, TC, handling and transfer equipment, and nuclear fuel assemblies to be stored are described in more detail in the following subsections.

1.2.1 Horizontal Storage Module

HSMs are constructed in arrays of reinforced concrete and structural steel. An HSM within a back-to-back side-by-side array is 6.096 m (20') deep, 4.572 m (15') high (plus 0.914m (3') high air outlet shielding blocks), and has the DSC stored 2.642m (8'-8") on centers. A 3x2 HSM array would be 12.192 m (40') deep and 9.144m (30') across. The concrete walls and roof are intended to be of sufficient thickness to attenuate radiation so that the average contact dose rate on the outside surface of the HSM is less than 20 mrem/hour.

The TR reference design is based on an installation of six modules arranged in a 2x3 array on a load-bearing foundation. Each HSM can hold one DSC. The modules are arranged back-to-back so that loading of each module is accomplished through an opening in the front. The center of the opening is approximately 2.591m (8.5 feet) above the surface of the foundation.

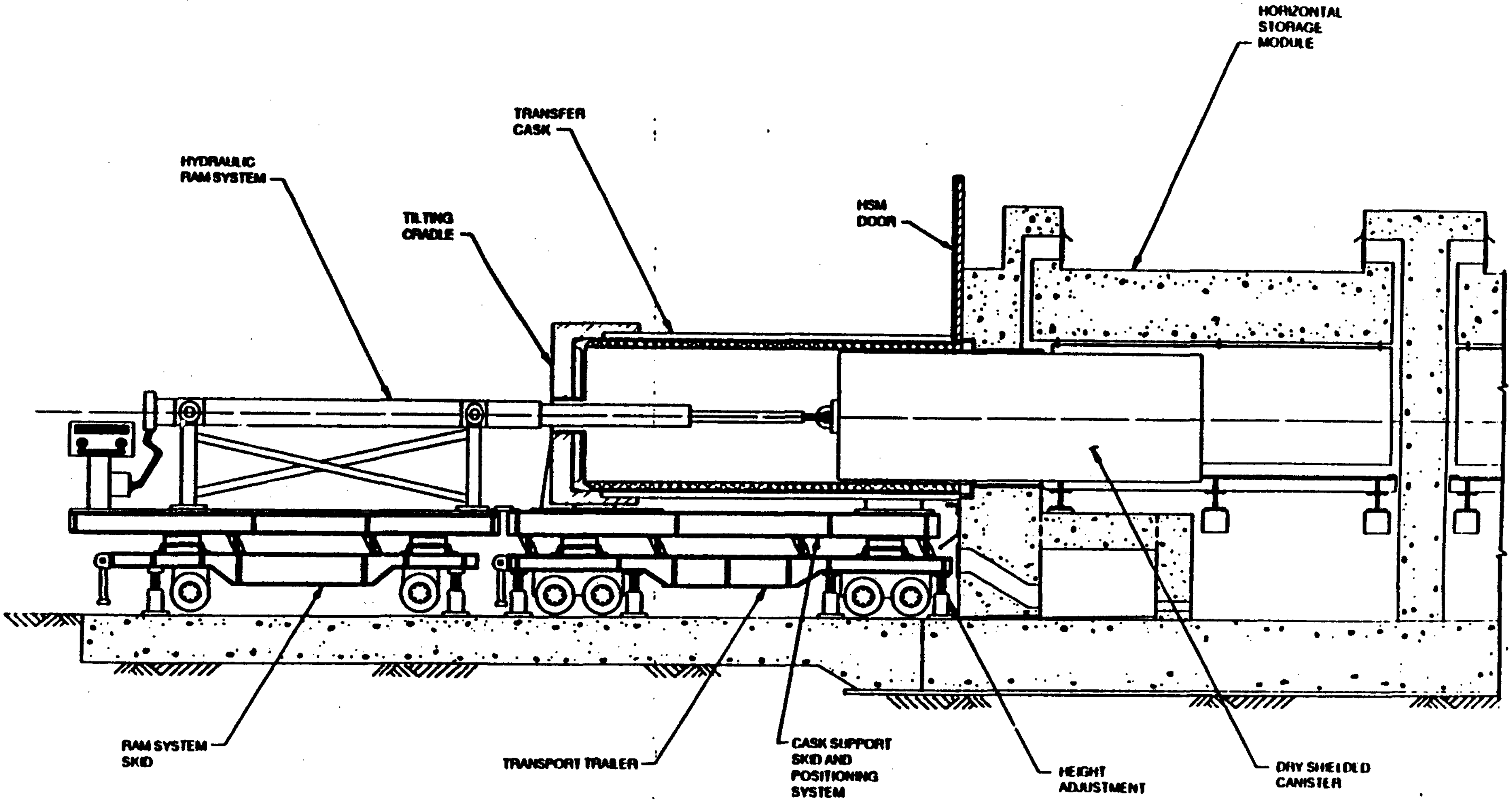

Figure 1.1

PRIMARY COMPONENTS OF THE NUHOMS SYSTEM

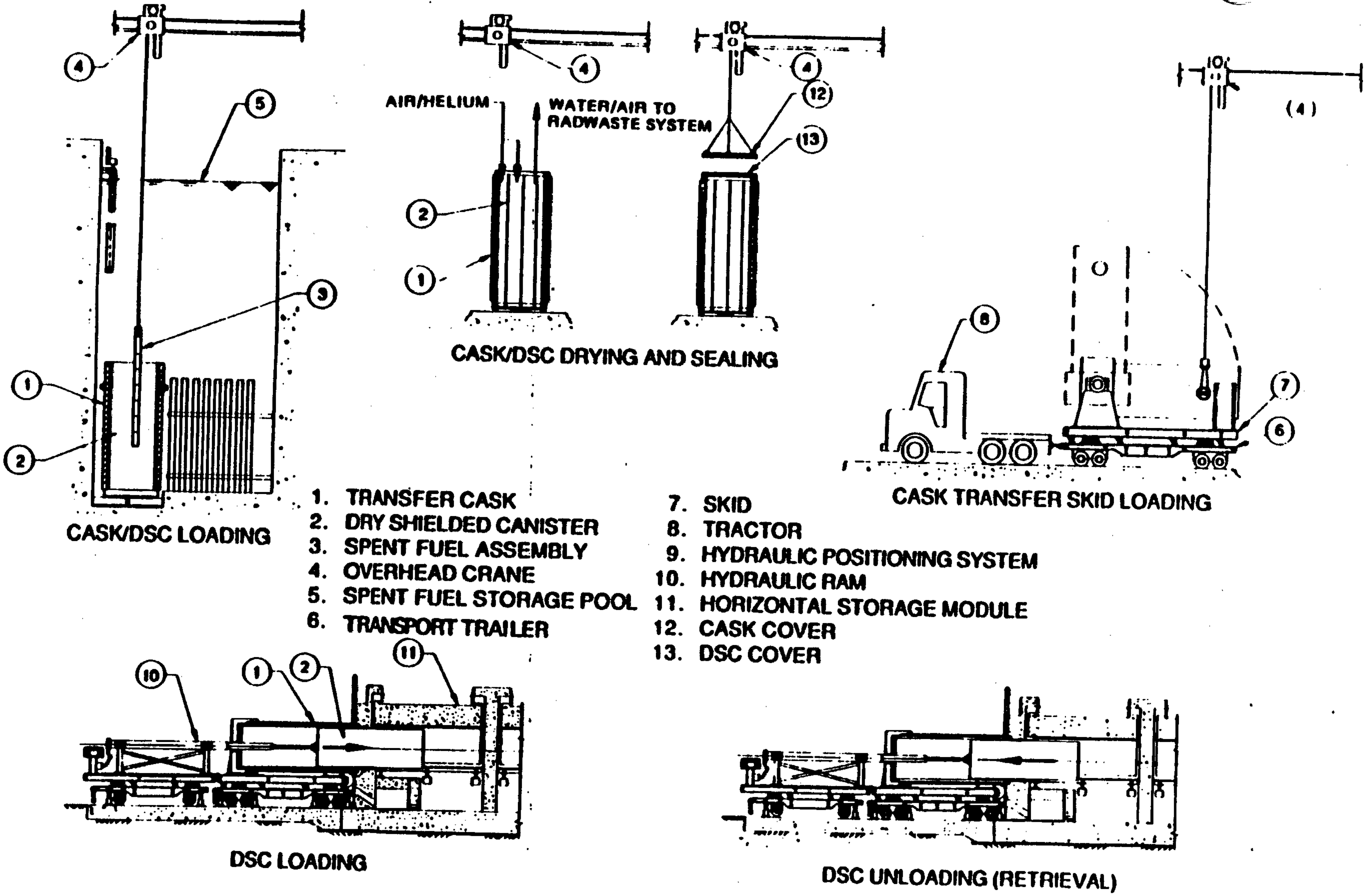

Figure 1.2

PRIMARY OPERATIONS FOR THE NUHOMS SYSTEM

After the HSM has been loaded with a DSC, a steel door is lowered down over the front opening and tack welded in place.

There are two steel rails inside the HSM running front-to-back which support the DSC while it is in storage. Each HSM has an air inlet on the front below the DSC opening and two air outlets on the roof to permit natural convective air cooling of the DSC while it is in storage. The inlet and outlets are shielded to reduce radiation doses at the exterior of the HSM.

1.2.2 Dry Shielded Canister

The DSC consists of a stainless steel (ASME SA-240, Type 304) cylindrical body, two shielded end plugs, and an internal basket to hold and support twenty-four irradiated pressurized water reactor (PWR) fuel assemblies.

The DSC body is a 15.9 mm (0.625") thick stainless steel cylinder. It has an outside diameter 1.708 m (67.25"). Its length is 4.724 m (186.00"). When welded shut, the DSC may be evacuated through valves, backfilled with helium at 1.2 bar. The valves may then be fully sealed by welding.

The internal basket is composed of twenty-four separately formed square cells. Structural support of the cells inside the DSC is provided by circular stainless-steel spacer disks. Longitudinal support of the disks is provided by four support rods that run the length of the canister from one end shield to the other. The cells and supporting assembly are fabricated of Type 304 stainless steel.

Each end of the DSC is equipped with a shielded end-plug so that when the canister is inside the transfer cask or the HSM, the radiation dose at the ends is limited. The top end shield is 184 mm (7.25") total thickness of stainless steel and lead. The bottom end shield is 147.0 mm (6.0") total thickness of stainless steel and lead.

The DSC has redundant seal welds for the top and bottom end plugs. The bottom is shop-welded during fabrication. The top cover plates are welded at the site after fuel is loaded in the DSC. The valve connections (drain and air purge lines) are also sealed at the site.

The DSC has a grapple attachment integrated with its bottom end to provide for insertion and withdrawal at the HSM by use of a trailer mounted hydraulic ram (ram design is left to site specific license application). The ram is inserted through the bottom port of the TC, is connected to the DSC, and inserts the DSC by pushing it out of the TC into the HSM or withdraws it by pulling it out of the HSM into the TC. The DSC slides on the HSM rails and internal TC surface in these operations.

1.2.3 Transfer Cask

The transfer cask consists of a structural steel and lead shell with a neutron shield water jacket and overflow tank, an integral steel bottom end incorporating a solid neutron shield, a bolted-on vented steel top cover incorporating a solid neutron shield, and a smaller diameter bolted-on steel bottom cover over the ram access port incorporating a solid neutron shield. The TC is equipped with a drain plug for draining the cask and provisions for filling and venting the neutron shield water jacket.

The TC has an outer diameter of 2.165 m (85.27") (exclusive of the overflow tank), an inner diameter of 1.727 m (68"), an inner clear length of 4.750 m (187"), and an overall outer length of 5.009 m (197.2").

The TC is intended to be hoisted by trunnions on its sides. The DSC is to be loaded with fuel assemblies in a vertical orientation within the TC. For transport the TC is placed in a specially designed carrying assembly and rotated to the horizontal position (as shown in Figure 1.2).

1.2.4 Handling and Transfer Equipment

In order to support the operation of the NUHOMS. system, several additional components are needed for the handling of both the fuel and the DSC and for the transfer of the loaded and sealed DSC to the HSM. Designs or selection of these items are left to the site-specific license application. They include the following major components:

1. Lifting assemblies or crane adaptor assemblies for the DSC, TC, DSC cover, and TC cover.

2. Welding machine suited to remote welding of the DSC cover.

3. DSC evacuation and helium backfill systems.

4. Transfer vehicle capable of moving the loaded cask across the site.

5. Jack support system for the transfer vehicle to be used to restrict relative motion between the ground (loading apron) and the trailer.

6. Cask positioning skid to adjust the cask position at the HSM to allow proper alignment before the DSC is transferred to the HSM.

7. Cask restraint system to prevent relative motion between the cask/skid and the HSM during inserting or withdrawing operation.

8. Optical alignment system to align the loaded cask with the HSM opening.

9. Ram and grapple apparatus to push the DSC from the TC into the HSM and to withdraw the DSC from the HSM into the TC.

10. Components to reverse the process in order to retrieve fuel assemblies from the DSC.

The staff has reviewed these components primarily from the point of view of feasibility. That is, these components have been reviewed only to determine if the staff believes that all operations required to support the NUHOMS system can be performed by current technology, that such equipment exists or can be fabricated, and that such a system could perform its required functions. Review and approval of all NUHOMS-24P ISFSI system physical components is left to the site specific license application, with the specific exceptions of the DSC, TC, and HSM.

1.2.5 Stored Materials

Each HSM holds one DSC and each DSC holds twenty-four irradiated PWR fuel assemblies. The proposed system is designed to permit storage of any PWR fuel with the following criticality and radiological characteristics:

1. Initial uranium content: 472 kg/assembly or less.

2. Initial enrichment: 4.0% (^{235}U equivalent) or less.

3. Fuel rod cladding of zircaloy.

4. No known or suspected cladding damage.

5. Irradiated fuel initial enrichment less than or equal to 1.45 weight percent ^{235}U unirradiated fuel.

6. Post irradiation cooling time such that:

   a) Decay Heat Power per Assembly <0.66 kW,
   b) Total Gamma Ray Source per DSC <3.85 x 10^{16} MeV/sec, (1.11 x 10^{17} gammas/sec)
   c) Total Neutron Source per DSC <3.715 x 10^{9}.

7. Initial fuel rod fill gas pressure of less than 480 psig.

A fuel assembly not meeting the specified conditions must be analyzed specifically before it can be stored in the proposed NUHOMS design.

1.3 IDENTIFICATION OF AGENTS AND SUBCONTRACTORS

No subcontractors for design are identified in subsection 1.4 of the TR, however Duke Power Company, Inc. is identified as responsible for the design of the HSM and performance of the criticality analysis.

1.4 GENERIC HORIZONTAL STORAGE MODULE ARRAYS

The TR is based on a 2x3 array of front-loaded HSMs. Although the TR states that other arrays are possible, none were presented for review and approval. Review of the design indicates that the following other HSM arrays are adequately included by the design of the 2x3 array: 2x1 and 2x2, using the exterior wall designs of the 2x3 array. Shielding calculations provided only cover the situation where HSM are installed back-to-back.

2.0 PRINCIPAL DESIGN CRITERIA

2.1 SUMMARY AND CONCLUSIONS

This section of the SER presents a review of the design criteria developed and presented in the TR to determine the suitability of the NUHOMS-24P design criteria. Sections 3 through 10 evaluate the use and satisfaction of the criteria in the designed system components. Subpart F of 10 CFR Part 72 sets forth design criteria for the design, fabrication, construction, testing and performance of structures, systems and components important to safety in an ISFSI. This section presents a discussion of the applicability of these criteria to the NUHOMS system and the degree to which the NUTECH, Inc. TR is in compliance with these criteria. Section headings in this section are the same as applicable subsections of Subpart F of Part 72.

Section 3 of the TR identifies sources for design criteria. These sources, and their acceptability are summarized in Table 2.1. The NRC staff concurs in the selection of sources in the TR, with the following exception:

ACI 349-85 was used in lieu of ACI 349-80 (Reference 5), which is cited in paragraph 6.17.2 of ANSI/ANS-57.9-1984 (Reference 6). This standard is endorsed with modifications in Regulatory Guide 3.60 by reference. There is no impact on the designs in the TR however and therefore NUTECH's use of ACI 349-85 in this TR for design of the HSM is considered acceptable. The NRC staff also notes that as Regulatory Guide 3.60 is a "guide," the use of a substitute determined to be acceptable to the NRC is satisfactory.

Section 3 of the TR also establishes design criteria used subsequently for design procedures and designs discussed in Sections 4 and 8 of the TR. These design criteria, as presented in Section 3 of the TR, are considered acceptable with the following exceptions:

1. There are two discrepancies in the methodology used by NUTECH for combining loads for the HSM. The factors to be applied to the dead load and the live load are not consistent with ANSI 57.9-84; however, the safety of the HSM is not compromised because of the design margin.

TABLE 2.1 DESIGN CRITERIA SOURCES CITED IN THE TR

(Sources are more fully described at TR Section 3.6)
(Similar citations within TR Section 3 are not repeated)

TR Ref	Source	Use	NRC Staff Comments
3.1.1.2, Tbl 3.1-3	NUREG/CR-2397 (ORNL-CSD-90)	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.2, Tbl 3.1-3	ORNL/TM-7431	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.2, Tbl 3.1-3	ANSI/ANS-5.1-1979	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.2, Tbl 3.1-3	A.D. Little, Inc., "Tech Spt for Rad Stds for Hi-Lvl Rad Waste Mgt"	Fuel Assembly Thermal Parameters	Acceptable
3.1.1.3, Tbl 3.1-4	NUREG/CR-2397 (ORNL/TM-7431)	Development of radiological criteria using ORIGEN calculations	Acceptable
3.1.2.2	ANSI 57.9-1984	Design std for cask handling crane	Acceptable
3.1.2.2	Reg Guide 1.60	Seismic Design Response Spectra	Acceptable
3.1.2.2	Reg Guide 1.61	Seismic Design Damping Values	Acceptable
3.1.2.2	ANSI/ANS 57.9-1984	Operational Handling Loads	Acceptable
3.1.2.2	ANSI/ANS 57.9-1984	Accident Drop Loads	Acceptable
3.1.2.2	ANSI/ANS 57.9-1984	Thermal and Dead Loads	Acceptable
3.1.2.2	Reg Guide 1.76	Tornado Wind Loads	Acceptable
3.2.1	NUREG 0800	Tornado Missiles	Acceptable
3.2.1.2	ANSI A58.1-1982	Tornado Wind MPH to Pressure Conversion	Acceptable
3.2.4	ANSI A58.1-1982	Snow and Ice Loads	Acceptable
3.2.5.1	ACI 349-85	Reinforced Concrete Design	Acceptable, however ACI 349-80 is currently approved by NRC (per Reg Guide 3.60)
3.2.5.1	ANSI/ANS 57.9-1984	Load Combinations for HSM Design	Acceptable
3.2.5.2	ASME B&PV Code (1983) Sect III, Div 1, Subsec NB for Class 1 comp.	DSC allowable stresses	Acceptable
3.2.5.3	ASME B&PV Code (1983) Sect III, Subsec NC for Class 2 comp.	TC allowable stresses	Acceptable
3.2.5.3	ANSI N14.6-1986	Allowable stresses for lifting trunnions in fuel bldg.	Acceptable
Tbl 3.2-1	AISC Code for Struct Steel	DSC Support Assy Design	Acceptable for design stresses, not for load combs.

TABLE 2.1 DESIGN CRITERIA SOURCES CITED IN THE TR (cont'd)

TR Ref	Source	Use	NRC Staff Comments
Tbl 3.2-1	ASME B&PV Code (1983) Sect III, Subsec NC for Class 2 comp.	Allowable stresses for lifting and support trunnions on site transfer	Acceptable
3.3.2.1	ASME B&PV Code (1983) Sect III, Div 1, NB	DSC pressure boundary weld inspection	Acceptable
3.3.4.2	STUDSVIK/NR-81/3	CASMO-2 Fuel Assy Burnup Prog.	Acceptable
3.3.4.2	ORNL, "SCALE-3:_"	Shielding Anal Seq. No. 2	Acceptable
3.3.4.2	ORNL, "SCALE-3:_"	Criticality Safety Anal Seq. No. 2	Acceptable
	SAND 87-0151	Major neutron absorbers	Acceptable
3.3.4.3	ANSI/ANS 57.2-1983	Criticality criteria	Acceptable
	ANSI/ANS 8.17-1984	Fuel burnup credit	Acceptable
3.3.4.4	ANSI/ANS 8.17-1984	Double contingency principle	Acceptable
3.3.7.1	PNL 6189	Temp limits for dry stored fuel	Acceptable

2. The design criteria for the DSC support assembly does not include the dead load of the DSC for the off-normal case; however, the actual analysis does include the DSC dead load.

3. The derivation of the allowable shear stress for the DSC support assembly as used by NUTECH would result in exceeding the code specified in ANSI 57.9-84 section 6.17.3.2.1 for steel design. Because NUTECH selected an overly conservative temperature in conjunction with the seismic event, the NRC judged that the material allowable was also conservative. There will not be a safety problem if a lower temperature is used in the derivation.

4. NUTECH proposed a 10% value of critical damping for the DSC and TC for the accident drop case. This value is higher than recommended by Regulatory Guide 1.61 (Reference 7). The staff evaluated this deviation and determined that 7% is a conservative estimate for the damping coefficient and also determined that no safety problems will occur for the drop if 7% damping is used.

2.2 FUEL TO BE STORED

The NUHOMS-24P system is designed for dry, horizontal storage of irradiated PWR fuel from nuclear power stations. Acceptable fuel characteristics are presented in subsection 1.2.5 of this report and elaborated in Table 3.2-1 of the TR. The principal design parameters of the fuel to be stored are intended to accommodate standard PWR fuel designs manufactured by Babcock and Wilcox, Combustion Engineering, Westinghouse and Advanced Nuclear Fuels.

The physical parameters of the DSC design are based on a hybrid set of design parameters which will accommodate standard fuel assembly arrays of (1) 15x15/208 and 17x17/264 designed by Babcock and Wilcox, and (2) 14x14/176, 15x15/216, and 16x16/236 designed by Combustion Engineering. The fuel assemblies 14x14/179, 15x15/204, and 17x17/264 designed by Westinghouse were listed in Table 3.1-2 of the TR for general reference only and are not bounded by the design case for the TR. The design case is B&W 15x15/208.

The design basis for nuclear criticality safety is based on standard Babcock and Wilcox 15 x 15 fuel assemblies with an initial enrichment of 4 weight percent ^{235}U. The design basis for radiation protection is based on 4.0 weight percent ^{235}U B&W 15 x 15 fuel irradiated to 40,000 MWd/MTHM at a specific power of 37.5 MW/MTHM with a post irradiation cooling time of ten years before being stored in the NUHOMS-24P system.

The fuel cladding temperature limits used by the applicant are based on the work of I.S. Levy, et al. (Reference 36). In developing limits, the applicant relied upon the following restrictions: (1) burnup $\leq$40,000 MWD/MTU, (2) rod fill pressure up to 480 psig, and (3) cooling times of ten years or more. These restrictions must be satisfied by the stored fuel. The last restriction limits the assembly power to 0.66 kW, and to 0.66 kW at ten years cooling time.

The results of this safety review are that the use of these design parameters for the fuels meet the requirements of 10 CFR Part 72 as applied to the DSC design, criticality design, and shielding design.

2.3 QUALITY STANDARDS

Quality standards for structures, systems and components important to safety are required by Sections 72.122(a) and 72.140 of 10 CFR 72. Sections 3.4 and 11 (which incorporates Section 11 of Reference 4 by reference) of the TR identify components of the NUHOMS-24P system that are classified as important to safety. A quality standard provides numerical criteria and/or acceptable methods for the design, fabrication, testing, and performance of the structures, systems and components important to safety. These standards should be selected or developed to provide sufficient confidence in the capability of the structure, system, or component to perform the required safety function. Since quality standards are generally embodied in widely accepted codes and standards dealing with design procedures, materials, fabrication techniques, inspection methods, etc., judgments regarding the adequacy of the standards cited in the TR are presented in the sections of this report where the standards are applicable.

2.4 PROTECTION AGAINST ENVIRONMENTAL CONDITIONS AND NATURAL PHENOMENA

Section 72.122(b) of 10 CFR 72 requires the licensee to provide protection against environmental conditions and natural phenomena. Section 3.2 of the TR describes the structural and mechanical criteria for tornado and wind loadings, tornado missile protection, flood protection, seismic design, snow, ice and dead loads, pressure and thermal loads resulting from normal operating conditions and accident conditions, normal and accident handling loads, accidental drop loads, and combined loads.

This section discusses the adequacy of the selected criteria for protecting the following components against environmental conditions and natural phenomena: (1) the reinforced concrete HSM and the HSM passive ventilation systems, (2) the DSC support assembly, (3) the DSC, including the internal basket components and the shielded end plugs, and (4) the on-site transfer cask, including the shield materials, structure and upper and lower trunnions. The above mentioned structures and component are important to safety because they contribute to the safe confinement of the radioactive spent fuel assemblies. The technical bases for determining the adequacy of these criteria are specified by the regulatory requirements to consider the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take account of limitations of data. Since the NUHOMS system was not designed for a specific site, the regulatory requirement is interpreted to mean that the NUHOMS system should be reviewed against the environmental conditions and natural phenomena provided for either by the limits specified in the TR or against the most severe of the natural phenomena that may occur within the boundaries of the United States. Table 2.2 summarizes the design criteria used in the TR for design or evaluation for normal operating conditions. Table 2.3 summarizes the criteria for off-normal operating conditions and Table 2.4 summarizes the criteria for the accident conditions.

As can be seen in Tables 2.2, 2.3, and 2.4, some of the design criteria for safety related components are not explicitly defined by codes or regulations. In some cases NUTECH has applied engineering judgment to determine a performance envelope or design criteria for the system based on the intent of 10 CFR 72.122. The SER review is oriented on satisfaction of Regulatory Guides 3.48 and 3.60 primarily, with recognition that these are

TABLE 2.2 SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
Horizontal Storage Module	Dead load	8.1.1.5	Dead weight including loaded DSC	ANSI 57.9-1984 ACI 349-85 and ACI 349R-85	Verified by SER
	Load combination	Tbl 3.2-5	Load combination methodology	ANSI 57.9-6.17.1.1	Acceptable if dead weight increased by 5% over estimated value. Acceptable if live load varied between 0% and 100% of estimated load to achieve most adverse conditions.
	Design Basis operating temp	8.1.1.5	DSC with spent fuel rejecting 15.8 kW decay heat. Ambient air temperature range $0^{o}F$ to $100^{o}F$	ANSI 57.9-1984	Verified by SER
	Normal handling loads	8.1.1.4	Hydraulic ram load: 20,000 lb. (25% loaded DSC weight)	ANSI 57.9-1984	Verified by SER
	Snow and Ice Loads	3.2.4	Maximum load: 110 psf (included in live load)	ANSI 57.9-1984	Verified by SER
	Live Loads	8.1.1.5	Design load: 200 psf	ANSI 57.9-1984	Verified by SER
Dry Shielded Canister	Dead Loads	8.1.1.2	Weight of loaded DSC: 72,000 lb. nominal, 80,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
	Design Basis Internal Pressure Load	8.1.1.1	DSC internal pressure ≤ 9.7 psig	ANSI 57.9-1984	Verified by SER
Structural Design		Tbl 3.2-6	Service Level A and B	Also see ASME B&PV Code Section III, Div 1, NB, Class 1, Service Level A,B	Verified by SER

TABLE 2.2 SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC	Design Basis Operating Temp Loads	8.1.1.2 8.1.2.2 Tbl. 3.2-6	DSC decay heat 15.8 kW. Ambient air temperature $0^{o}F$ to $100^{o}F$	ANSI 57.9-1984	Verified by SER
	Operational Handling Loads	8.1.1.2 Tbl. 3.2-6	Hydraulic ram load: 20,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
DSC Support Assembly	Dead Loads	8.1.1.4 Tbl. 3.2-7	Loaded DSC + self weight: 85,000 lb.	ANSI 57.9-1984 AISC Code	Verified by SER
	Operational Handling Load	8.1.1.4 Tbl. 3.2-7	DSC reaction load with hydraulic ram load: 20,000 lb.	ANSI 57.9-1984	Verified by SER
Transfer cask (on-site) Structure: Shell, rings, ends, etc.	Normal operating condition	Tbl. 3.2-8	Service Level A and B	ASME B&PV Code, Section III, Div. 1, NC, Class 2	Verified by SER
	Dead Loads	8.1.1.9	a) Vertical orientation, self weight + loaded DSC + water in cavity: 200,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
Structure: Shell, rings, ends, etc.			b) Horizontal orientation, self weight + loaded DSC on transfer skid: 193,000 lb. nominal, 200,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
	Snow and Ice Loads	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)	Verified by SER

TABLE 2.2 SUMMARY OF DESIGN CRITERIA FOR NORMAL OPERATING CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
Shell, rings, ends	Design Basis Operating Temp. Loads	8.1.1.9, 8.1.2.2	Loaded DSC rejecting 15.8 kW decay heat. Ambient air temperature range 0^{o}F to 100^{o}F.	ANSI 57.9-1984	Verified by SER
TC Upper Trunnions	Operational Handling Loads	8.1.1.9	a) Upper lifting trunnions while in fuel building:		
			i) stress must be less than yield stress for 6 times critical load of 115,000 lb./trunnion, nominal	ANSI N14.6-1978	Verified by SER
			ii) stress must be less than ultimate stress for 10 times critical load	ANSI N14.6-1978	Verified by SER Also see: NUREG-0612 and NOG-1-1983 and WRC-297
Upper Trunnions	Op. Handling	Append. C	b) Upper lifting trunnions for on-site transfer: 118,000 lb./trunnion, 94,000 lb./shear, 29,500 lb./trunnion axial.	ASME B&PV Code Section III, NC, Class 2	Verified by SER
Lower Trunnions	Op. Handling	8.1.1.9	c) Lower support trunnions: weight of loaded cask during down loading and transit to HSM	ASME B&PV, Section III, NC Class 2	Verified by SER Also see WRC-297
Shell	Op. Handling	8.1.1.9	d) Hydraulic ram load due to friction of extracting loaded DSC: 20,000 lb. enveloping	ANSI 57.9-1984	Verified by SER
Bolts	Normal operation	8.1.1.9 Tbl. 3.2-9	Service Levels A, B and C Ave. Stress less than 2 Sm Max. Stress less than 3 Sm	ASME B&PV Section III, NC, Class 2	Verified by SER

TABLE 2.3 SUMMARY OF DESIGN CRITERIA FOR OFF-NORMAL OPERATING CONDITIONS

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
HSM	Off-normal Temperature	8.1.1.5	-40^{o} to 125^{o}F ambient temperature	ANSI 57.9-1984	Verified by SER
	Jammed Condition Handling	8.1.1.4	Hydraulic ram load equal to 100% of DSC: 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER
	Load combination	Tbl. 3.2-5	Load combination methodology	ANSI 57.9-1984 - 6.17.1.1	Acceptable if dead weight increased by 5% over estimated value. Acceptable if live load varied between 0% and 100% to achieve the most adverse conditions
DSC	Off-normal Temperature	8.1.2.2	-40 to 125^{o}F ambient temperature	ANSI 57.9-1984	Verified by SER
	Off-normal Pressure	8.1.1.2	DSC internal pressure less than 9.7 psig	ANSI 57.9-1984	Verified by SER
	Jammed Condition Handling	8.1.2.1	Hydraulic ram load equal to 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER
Structural Design	Off-normal Conditions	Tbl. 3.2-6	Service Level C	ASME B&PV Section III, Div 1, NB, Class 1	
DSC Support	Jammed Condition Handling	8.1.1.4	Hydraulic ram load: 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER

TABLE 2.3 SUMMARY OF DESIGN CRITERIA FOR OFF-NORMAL OPERATING CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC Support	Load combination	Tbl. 8.2-11	Load combination methodology		Neglects deadload of DSC, which must be present. (Note SER verified that actual design did include DSC)
	Combined stresses	Tbl. 8.2-11	Calculation of allowable stresses	ANSI 57.9-1984, 6.17.3.2.1	Shear stress limit in TR Tbl. 8.2-11 is higher than allowed by code.
TC	Off-normal Temperature	8.1.1.8 8.1.2.2	-40 to 125^{o}F ambient temperature	ANSI 57.9-1984	Verified by SER
	Jammed Condition Handling	8.1.2.1	Hydraulic ram load: 80,000 lb. nominal	ANSI 57.9-1984	Verified by SER
Structural Design	Off-normal Conditions	Tbl. 3.2-8	Service Level C	ASME B&PV Section III, Div 1, NC, Class 2	Acceptable
Bolts	Off-normal Conditions	Tbl. 3.2-9	Service Level C Ave. stress less than 2 Sm Max. stress less than 3 Sm	ASME B&PV Section III, Div. 1, NC, Class 2	Acceptable

TABLE 2.4 SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
HSM	Design Basis Tornado	3.2.1	Max. velocity 360 mph Max. wind pressure 304 psf	NRC Reg. Guide 1.76 ANSI 58.1 1982	Adequate
	Load Combination	Tbl. 3.2-5	Load combination methodology	ANSI 57.9-84 6.17.1.1	Acceptable if dead weight increased by 5% over estimated value. Acceptable if live load varied between 0% and 100% to achieve most adverse conditions.
	DBT Missiles	3.2.1	Max. velocity 126 mph Types: Automobile 3967 lb. 8 in. diam shell 276 lb. 1 in. solid sphere	NUREG-0800 Section 3.5.1.4	Verified by SER
	Flood	3.2.2	Max. water height: 50 ft. Max. velocity: 15 ft/sec	10 CFR 72.122	Adequate for limit design. Licensee to determine site design parameters and check against ACI 349-80 equation 2.4.7.
	Seismic	3.2.3	Horizontal ground acceleration 0.25 g (both directions) Vertical ground acceleration 0.17 g 7% critical damping	NRC Reg Guides 1.60 and 1.61	Adequate
	Accident Condition Temperature	8.2.7.2	HSM vents (inlet/outlet) blocked for 48 hrs or less. HSM inside surface temp: 395°F	ANSI 57.9-1984	Verified by SER
	Fire and Explosions	3.3.6	Site specific	10 CFR 72.122(c)	Not designed for by NUTECH. Must evaluate on site specific basis.

TABLE 2.4 SUMMARY OF DESIGN CRITER[illegible]R ACCIDENT CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC	Accident Drop	8.2.5	Equivalent static deceleration: 75 g vertical end drop 75 g horizontal side drop 25 g corner drop with slapdown (corresponds to an 80 inch drop height)	10 CFR 72.122(b)	Verified by SER Also see i) EPRI report NP-4830 ii) LLNL report UCID-21246
			Structural damping during drop: 10%	Reg Guide 1.61	10% damping value exceeds R.G. 1.61 guidance. A 7% value has been evaluated by the staff and has been accepted. Verified by SER.
	Flood	3.2.2	Maximum water height 50 ft.	10 CFR 72.122	Adequate for limit design
	Seismic	3.2.3	Horizontal acceleration 1.0 g Vertical acceleration 0.68 g 3% critical damping	Reg. Guides 1.60 and 1.61	Verified by SER
	Accident Internal Pressure (Loss of cask neutron shield)	8.2.9	DSC internal pressure: 49.1 psig based on 100% fuel clad rupture and fill gas release, 30% fission gas release, and ambient air temperature = 125°F.	10 CFR 72.122(b)	Verified by SER
	Accident Internal Pressure (HSM vents blocked)	8.2.7	DSC internal pressure: 46.7 psig based on 100% fuel clad rupture and fill gas release, and ambient air temperature = 125°F. DSC shell temperature 455°F	10 CFR 72.122(b)	Verified by SER
	Accident Conditions	Tbl. 3.2-6	Service Level D	ASME B&PV Section III Div. 1, NB, Class 1	Acceptable with operational controls. See para. 10.3.2.9 of TR.

TABLE 2.4 SUMMARY OF DESIGN CRITE[illegible]R ACCIDENT CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Req. Guides Cited by NUTECH	NRC Staff Comments/Suitability
DSC Support Assembly	Seismic	3.2.3	DSC reeaction loads: horizontal acceleration 0.4g vertical acceleration 0.27 g 7% critical damping	Reg. Guides 1.60 and 1.61	Verified by SER
	Load combination	Tbl. 8.2-11	Load combination methodology	ANSI 57.9-84 6.17.3.2.1	Shear stress limit in TR Tbl. 8.2-11 is higher than allowed by code.
TC	Design Basis Tornado	3.2.1	Max. wind velocity 360 mph Max. wind pressure 397 psf	Reg. Guide 1.76 and ANSI 58.1-1982	Verified by SER
	DBT Missiles	3.2.1	Automobile 3967 lb. 8 in. diameter shell 276 lb.	NUREG-0800 Section 3.5.1.4	Verified by SER
	Flood	3.2.2	Cask use to be restricted by administrative controls.	10 CFR 72.122	Adequate, must verify license application invokes controls.
	Seismic	3.2.3	Horizontal ground acceleration 0.25 g (both directions) Vertical acceleration 0.17 g 3% critical damping	Reg. Guides 1.60 and 1.61	Verified by SER
	Accident Drop	8.2.5	Equivalent static deceleration 75 g vertical end drop 75 g horizontal side drop 25 g corner drop with slapdown (corresponds to an 80 inch drop height)	10 CFR 72.122(b)	Verified by SER Also see i) EPRI report NP-4830 ii) LLNL report UCID-1246
			Structural damping during drop 10%	Reg. Guide 1.61	10% damping exceeds R.G. 1.611 guidance; however, 7% has been evaluated by the staff and accepted. Verified by SER

TABLE 2.4 SUMMARY OF DESIGN CRITERIA FOR ACCIDENT CONDITIONS (cont'd)

Component	Design Load Type	TR Reference	Design Parameters	Applicable Codes or Reg. Guides Cited by NUTECH	NRC Staff Comments/Suitability
TC bolts	Accident Drop	Tbl. 3.2-9	Service Level D	ASME B&PV Section III, NC, Class 2	Verified by SER
TC Structural Design	Accident	Tbl. 3.2-8	Service Level D	ASME B&PV Section III, NC, Class 2	Verified by SER Acceptable with operational controls. (See para. 10.3.2.9 of TR)
	Fire and Explosions	3.3.6	Site specific	10 CFR 72.122(c)	Not designed for by NUTECH. Must evaluate on site specific basis.
	Internal Pressure	-	Not applicable because DSC provides pressure boundary	10 CFR 72.122(b)	Verified by SER

"guides." Where deviations from these guides occur, or for areas not specifically covered, the NRC staff has reviewed NUTECH's selection or derivation of criteria using the principles of the guides, accepted codes, and engineering practices as standards.

There are cases where the minimum criteria used by NUTECH are considered unacceptable (see Tables 2.2, 2.3, and 2.4), being contrary to the Regulatory Guides or accepted codes and standards. This is not prima-facie evidence that there is a safety problem, however, since the actual design may have exceeded the minimum criteria to the extent of satisfying acceptable higher criteria. In addition, different codes use different design bases, combinations of factored loads, and derivation of allowable stresses. Thus apparent problems may not exist on more detailed examinations of the actual design (SER Section 3).

Some load sources and load combinations that would normally be included in a structural analysis have been omitted by NUTECH. The basis is typically that another, more severe loading case envelopes the condition. For example, the design basis tornado (DBT) wind loadings are typically higher than non-DBT wind loadings. These rationale have been reviewed by the NRC staff in conjunction with review of the criteria, and except where noted to the contrary, have been determined to be acceptable.

NUTECH defined the following normal operating events: dead weight loads, design basis internal pressure loads, design basis thermal loads, operational handling loads, and design basis live loads. The criteria associated with these loads are presented in Table 2.2. The staff has reviewed these criteria and with the following exceptions, considers them to be acceptable.

1. Failure to comment that dead load should be, or was, increased by 5% over the estimated value, as stated in ANSI/ANS 57.9-84 section 6.17.1.1, and applicable to both concrete and steel design. [NOTE: Analytical results suggest that design margins exceeded this value and thus it does not result in a safety concern.]

2. Failure to comment that the live load should be, or was, varied between 0% and 100% of estimated load to simulate the most adverse

conditions for the structure. [NOTE: The discussions of actual analyses that indicate that the worst case loading conditions were assumed and/or the design margin of the actual design cause this not to be a safety concern.]

Off-normal events that can be expected to occur on a moderate frequency were postulated by NUTECH. They included: jammed DSC during transfer, off-normal temperatures (-40^{o}F to 125^{o}F), and off-normal pressurization of the DSC. The criteria associated with these conditions are shown in Table 2.3. The staff has reviewed these criteria and considers them to be acceptable with the following exceptions.

1. The design criteria (Table 3.2-1 of the TR) for the DSC support assembly off-normal case include the dead load of the support assembly (about 5000 lbs.) and handling loads due to the jammed DSC, but not the dead load of the DSC itself. In the actual analysis (TR section 8.1.2.1.B) a vertical load corresponding to the DSC dead weight was used. As a result the omission of the dead load from the table presenting the criteria is not a safety problem.

2. An increase in allowable shear stress of 50% was used for the accident condition criteria for the DSC support assembly. This exceeds the absolute 40% maximum increase allowed by ANSI/ANS 57.9-84 section 6.17.3.2.1 for steel design. The allowable shear stress used was determined by factoring a tensile yield stress based on an elevated temperature. This temperature (600^{o}F) would not be approached except under an accident condition of blockage of air inlets and outlets (TR section 8.2.7.2). As the critical stress level is produced by seismic forces the allowable is, in effect, based on the simultaneous occurrence of two "accidents" (a 48-hour vent blockage in 125^{o}F ambient air, and an earthquake). NUTECH used an extreme off-normal (less than "accident") temperature for determining the tensile yield stress. From that, they determined the allowable shear stress in a non-thermal accident (i.e., $0.4xF_yx1.4$) which provides an allowable shear stress higher than that used. As a result of the overly conservative derivation, the allowable shear stress is not considered to be a safety problem.

As stated by 10 CFR 72.122, those structures of an ISFSI that are important to safety must be able to withstand the effects of accident conditions resulting from extreme environmental conditions, natural phenomena and postulated accidents. The extreme environmental and natural phenomena conditions include: 1) tornado winds and tornado missiles, 2) flood, 3) earthquakes, and 4) lightning. The accident conditions include: 1) loss of HSM air outlet shielding blocks, 2) blockage of the HSM air inlets and outlets, 3) accidental internal pressure in the DSC, 4) postulated DSC leakage, and 5) a postulated drop of the DSC (a drop distance of 80 inches while in the transfer cask) resulting in a 75g deceleration if dropped in the vertical or the horizontal orientations, and a 25g deceleration if dropped on the corner with subsequent slapdown. The bases associated with each of these load conditions are discussed below and are summarized in Table 2.4. The staff has reviewed these criteria and considers them to be acceptable as defined in Table 2.4 with the exceptions discussed below.

Three of the exceptions noted in Section 2.1 have been discussed. Two relate to the methodology for combining loads for dead weight and live load for the HSM. A third relates to the derivation of the shear stress limit for the DSC support structure. See previous discussion. The fourth exception (noted in Section 2.1) to the criteria used for the designs of the DSC and its internal basket assembly is the value proposed for damping of the DSC during the drop accident. The selection of 10% as the value for critical damping for the accident drop case deviates from guidance provided by Regulatory Guide 1.61 (Reference 7). The Regulatory Guide suggests that a damping value of 4% for welded steel structures and 7% for bolted steel structures be used for calculating loads in seismically loaded structures in nuclear power plants. The DSC is a completely welded structure and the cask is welded except for the top lid which is bolted. Thus a conservative damping value based on Regulatory Guide 1.61 would be 4%. The NRC evaluated this deviation from the Regulatory Guide and determined that 7% is acceptable based on several sources in the open literature as well as several additional technical considerations. References 8 and 9 indicate that welded steel structures stressed to levels at or just below the yield point of the material have critical damping values of 5%, and if the yield point of the material is exceeded, as NUTECH predicts in the event of a drop, then 7-10% damping can be expected. A more recent study (Reference 10) also shows a strong correlation of increased damping as the stress levels increase from linear elastic to stress levels in the plastic region for structures with inherently ductile materials.

Damping associated with impact is somewhat different than damping associated with seismic events. Components in nuclear power plants subjected to a seismic event may be suspended so that they are free to vibrate at their natural frequencies. Portions of a structure (such as a cask) which are located on the direct load path during a drop impact event are likely to be critically damped, i.e., 100% (Reference 11). At the point of contact, freedom of movement of the dropped object is reduced because of the high compressive forces between the object and the target. At locations other than the direct load path, vibration of the object will not be critically damped. At the impact areas for both the DSC and the TC the local stress levels exceed yield stresses, but remain below the allowable stress level for Service Level D (see Reference 12, ASME Boiler and Pressure Vessel Code, Section III). Thus, at the contact area, stress levels will exceed yield and a higher damping level will exist.

Based on the above references and observations, the NRC staff accepts 7% as a conservative damping coefficient for the drop accident case for both the DSC and the TC. The fact that NUTECH used 10% does not pose a safety concern because the NRC staff calculated the DSC acceleration levels associated with the dynamic load factors for 7% damping and found that the acceleration levels used by NUTECH were equal to or greater than those accelerations determined by the NRC staff.

The HSM and the TC are designed to withstand DBT and tornado generated missiles. The transfer cask resistance to DBT and the potential safety hazard of a tornado generated missile was evaluated. The DSC was not designed or evaluated for DBT or tornado generated missiles as it would be continually housed in either the TC or the HSM when outside of the fuel pool building, and both the TC and HSM were shown to provide satisfactory shielding of the DSC. Safety of the fuel rods and the containers while in the fuel pool building is outside the scope of the TR.

The design and/or safety evaluation criteria used for DBT and tornado generated missiles as described in the TR are considered acceptable. [NOTE: Adverse effects might result from overturning of the TC with contained DSC while on its transporter, which would be bracketed by the separately analyzed drop scenario. Another accident scenario analyzed was the possible puncture of the neutron shield water reservoir around the TC. The puncture and drainage of the TC neutron shield under detectable conditions (such as following a tornado) are not considered significant safety problems.]

The DSC and HSM design criteria include flood parameters of 15 fps velocity and 50-foot flood height. These flood conditions are assumed to exist during the normal situation of the DSC in storage in the HSM. The TR indicates that plant procedures are expected to be sufficient to avoid need to design or assess the case where the DSC is within the transfer cask during a flood. These are considered to be satisfactory assumptions for the TR. A site-specific license application, therefore, will be required to either validate these assumptions or provide further analysis if more severe flood parameters may occur.

A horizontal acceleration of 0.25g was established as a basis for seismic design in Section 3.2.3 of the TR. This selected acceleration is acceptable to the staff as a representative value for use in the TR. This acceptance recognizes that a site-specific evaluation will be required to establish geological and seismological requirements for each site-specific ISFSI application, as required by 10 CFR 72.102.

The vertical acceleration of 0.17g established in Section 3.2.3 of the TR is acceptable to the staff since this value is consistent with the Regulatory Guide 1.60 requirement that the vertical acceleration be 2/3 of the horizontal acceleration.

2.5 PROTECTION AGAINST FIRE AND EXPLOSION

The NUTECH TR does not specifically address protection of the NUHOMS system from potential fires or explosions. Instead, it relegates such analyses to a site-specific situation.

There are no flammable or explosive materials used in the construction and operation of the DSC or the HSM. Nevertheless, site-specific conditions can exist with the potential for fire and explosions in or around the HSM and DSC. Therefore, any application of the NUHOMS system to a specific site must analyze the consequences of fires and explosions and provide for protective and mitigative measures, as deemed necessary. NUTECH stated that the DSC has been calculated to withstand an external pressure of 21.7 psi. This external pressure is that which would result from immersion in fifty feet of water, a postulated accident considered in the flood analysis, Section 8.2.4 of the TR.

2.6 CONFINEMENT BARRIERS AND SYSTEMS

Subpart F of 10 CFR 72 provides the general design criteria and within that subpart, 72.122(h) deals with confinement barriers and systems. For the NUHOMS-24P system, 72.122(h)(1) is relevant to the dry storage of spent fuel as follows: "The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. This may be accomplished by canning of consolidated fuel rods or unconsolidated assemblies or other means as appropriate" (Reference 3).

The TR 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 accepts that the helium atmosphere will be maintained during storage. The staff then analyzed the impact of long-term storage on the behavior of spent fuel, using a diffusion controlled cavity growth (DCCG) mechanism as the basis for this calculation since it appears that this damage mechanism is the only one applicable to these storage conditions. Under the influence of stress and temperature, this damage mechanism progresses by the nucleation and growth of cavities along grain boundaries.

The staff also evaluated the impact of the cask dry-out procedure and off-normal operation on the behavior of the spent fuel. This evaluation was based on the concerns for the potential oxidation and creep of the fuel, respectively. All of these concerns and the evaluation results are discussed in Section 5, Confinement Barriers and Systems.

The analyses are predicated on the knowledge and control of the character of the spent fuel loaded into the DSC, particularly the quantity, specific power, and age of the fuel assemblies, and the heat dissipation properties of the system. The thermal evaluation is addressed in Section 4.

2.7 INSTRUMENTATION AND CONTROL SYSTEMS

Instrumentation and control systems are addressed in Part 72.126 of 10 CFR which requires 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 TR takes the position that because of the passive nature of the NUHOMS-24P system, no instrumentation is necessary. Since the DSC was conservatively designed to perform its containment function during all worst-case conditions, as can be shown by analysis, there is no need to monitor the internal cavity of the DSC for temperature or pressure during normal operations. The staff concurs with the position that instrumentation and control systems are not required for the NUHOMS-24P.

2.8 CRITERIA FOR NUCLEAR CRITICALITY SAFETY

The requirement stated in 10 CFR 72.124 is that spent fuel handling, transfer and storage systems be designed to be maintained in a subcritical configuration and to ensure that before a nuclear criticality accident is possible, at least two unlikely independent, concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design safety margins must reflect design uncertainties including uncertainties in handling, transfer, and storage conditions, data and methods used in calculations, and adverse accident environments. Section 72.124 also requires that the design be based on either favorable geometry or permanently fixed neutron absorbing materials. A criticality monitoring system is required in each area where special nuclear fuel is handled, used, or stored, except where the material is packaged in its stored configuration. Section 3.3.4 of the NUHOMS TR addresses nuclear criticality safety, and Section 7 of this report reviews the criticality analysis.

The acceptance criteria for nuclear criticality safety established in the present review was a 95% probability/95% confidence effective multiplication factor of 0.95 for storage and 0.98 for loading. The initial enrichment of unirradiated fuel assemblies was assumed in both cases. The maximum effective reactivity factor includes method and cross section biases, uncertainties in design parameters, and assumes optimal moderation conditions. Optimal moderation might occur if moderator boiling temperatures were reached; however, unirradiated fuel provides no moderator heating. The circumstances can be conceived for optimal moderation for irradiated fuel; however, the reactivity of the fuel would be significantly reduced.

2.9 CRITERIA FOR RADIOLOGICAL PROTECTION

Parts 72.24, 72.104(a), and 72.106(b) of 10 CFR 72 require the licensee to provide the means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR 20, for limiting the annual dose equivalent to any individual beyond the controlled area, and for meeting the objective of maintaining exposures as low as reasonably achievable (ALARA). Part 72.126 of 10 CFR requires 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.

Part 20.101(a) of 10 CFR 20 states that any individual in a restricted area shall not receive in any period of one calendar quarter from radioactive material and other sources of radiation a total occupational dose in excess of 1.25 rems to the whole body. Part 20.101(b) states that, under certain conditions, the quarterly dose limit to the whole body is 3 rems in any calendar quarter. Guidance for ALARA considerations is also provided in NRC Regulatory Guides 8.8 and 8.10 (References 13 and 14).

The TR establishes shielding criteria for the NUHOMS module of an average external surface dose of less than 20 mrem/hr. In addition, criteria were established of 200 mrem/hr for the transfer cask side surfaces and 100 mrem/hr on the DSC top lead plug. The shielding capability of the system relies primarily upon the bulk concrete shielding of the NUHOMS module and the DSC top lead plug.

The radiological protection design features of the NUHOMS-24P are described in Sections 3 and 7 of the TR. These features consist of (1) radiation shielding provided by the transfer cask, DSC, and HSM; (2) radioactive material confinement within the DSC, specifically the integrity of the double seal welds; (3) prevention of external surface contamination; and (4) site access control. Access to the site of the NUHOMS-24P array, although not specifically addressed in the TR, would be restricted by a periphery fence to comply with 10 CFR 72.106 controlled area requirements. The details of the access control features are site-specific, and would be described in the applicant's site license application.

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 as low as reasonably achievable (ALARA), (2) are in compliance with appropriate guidance and/or regulations, and (3) will assure that the dose from associated activities to any individual does not exceed the limits of 10 CFR 20.

Off-site radiological protection features of the NUHOMS-24P system are deemed acceptable if it can be shown that design and operational considerations which are not site-specific result in off-site dose consequences which are (1) in compliance with 10 CFR 72.104(a) for normal operations and anticipated occurrences, (2) in compliance with 10 CFR 72.106(b) for design basis accidents, and (3) are as low as reasonably achievable.

Based on analyses presented in the TR, the staff concludes that the NUHOMS-24P system, if properly sited, meets the requirements for on-site and off-site radiological protection, including the incorporation of ALARA principles. Radiological alarm systems and systems for monitoring effluents are not required in the NUHOMS design because of the integrity of the double seal weld on the DSC.

2.10 CRITERIA FOR SPENT FUEL AND RADIOACTIVE WASTE STORAGE AND HANDLING

Pursuant to 10 CFR 72.128(a), the licensee is required to design the spent fuel and radioactive waste storage systems 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. Part 72.128(b) further requires that radioactive waste treatment facilities 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 in this SER in the preceding subsections of this Section. The TR 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 under either normal operating or accident conditions.

The staff agrees that the design of the NUHOMS-24P 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 CRITERIA FOR DECOMMISSIONING

Part 72.130 of 10 CFR provides criteria for decommissioning. It requires that considerations for decommissioning 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.

Part 72.30 of 10 CFR defines the need for a decommissioning plan, which includes financing. Such a plan, however, is only appropriate to a site-specific situation, and 10 CFR 72.30 is therefore considered not applicable to this review.

The NUHOMS-24P TR claims that the DSC is designed to interface with a transportation system capable of transporting intact canistered assemblies to either a monitored retrievable storage (MRS) facility or a geologic repository. The TR does not identify any transportation system that could accept the DSC, hence the staff cannot comment about this claim. However, if the fuel must be removed from the DSC, the internal surface of the DSC will be contaminated and may be slightly activated. After the interior is cleaned to remove loose contamination, the DSC can be disposed of as low-level waste, or possibly even as scrap.

The current design of the 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.

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

1. There is no credible chain of events that 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 TR uses the following surface removable contamination limits as a guide:

Beta-gamma emitters:	10^{-4} uCi/cm^2
Alpha emitters:	10^{-5} uCi/cm^2

A detailed decommissioning plan, which would consider such factors as decommissioning options available, likely further use of the site, environmental impact, and available waste transportation and disposal capabilities, would be developed on a site-specific basis.

The staff concludes that adequate attention has been paid to decommissioning in the design of the NUHOMS-24P.

3.0 STRUCTURAL EVALUATION

3.1 SUMMARY AND CONCLUSIONS

This section presents the results of the NRC staff evaluations of the structural analyses and designs included in the TR. The evaluations are made against the accepted criteria, as evaluated in Section 2 of this SER. Where the criteria were not acceptable the staff examined the design results and compared those against acceptable codes, standards, or engineering practice, as appropriate. Thus, the actual design might be acceptable even if the stated design criteria are not acceptable.

The system descriptions presented in the TR are reviewed at two levels to determine: 1) whether the designs and descriptions are in themselves acceptable, and 2) the extent that the system description and analyses satisfy the requirements for a potential site-specific license application, and might thus be incorporated by reference or repetition in the application documentation.

This review includes an evaluation of all structural design criteria, analysis methodologies, material specifications, allowable stress levels and structural analyses. The staff has reviewed the structural design of the NUHOMS system proposed by NUTECH and confirms that the design is in compliance with 10 CFR 72.122 with the exceptions outlined below.

The staff has reviewed the structural analysis methodologies used in evaluating the structures and found them to be acceptable with the following exceptions:

1. Some discrepancies between the TR statements and the actual loads used in the HSM dead load analysis (see Table 8.1-9 and Section 8.1.1.5A in the TR) exist. The staff evaluated these discrepancies and concluded that the results of the analysis are satisfactory.

2. The method of accounting for concrete creep and shrinkage for the HSM was not documented in the TR; however, the staff reviewed the method used and determined that the methodology is appropriate.

3. The methodology used by NUTECH to calculate local DSC shell bending was not considered conservative by the NRC staff. A different model was selected by the staff and evaluated. The staff concludes that even with the more conservative method, the DSC design is adequate.

4. The type of finite element used by NUTECH to model the DSC and TC is a two-dimensional isoparametric element that only calculates membrane stress unless two elements are used together to model the shell thickness. Since NUTECH only used one element to model the thickness no bending stresses were calculated. However, the ASME Code requires that bending stresses as well as membrane stresses be evaluated for Class 1 components. The staff evaluated the results using alternative methods and concurs that the resulting design is satisfactory despite the flaw in the methodology.

The staff has reviewed the material specifications and allowable stress levels used in evaluating the system and confirmed that these data are in compliance with 10 CFR 72.122 with the following exception:

The material allowable stresses were evaluated for 400°F by NUTECH for the DSC (see Table 8.2-9c of the TR). The staff notes that the maximum temperature experienced by the DSC is 513°F for the Service Levels C and D. Even though the Code does not require an evaluation of thermal stresses for Service Levels C and D, the appropriate material allowable stresses must be used. The staff made this adjustment to lower allowables and concludes that the design for the DSC is satisfactory.

The staff has reviewed the structural analyses and designs presented in the TR for satisfaction of the requirements of 10 CFR 72.122 and finds that they are acceptable with the following exceptions:

1. The thicknesses of the top and bottom cover plates as modeled by the computer analyses <u>do not</u> agree with the thicknesses of the plates in the design drawings. The staff used a ratio of the squares of the thicknesses to multiply the computer listings by in order to estimate the stresses. All stress levels were found satisfactory after this adjustment.

2. Although the TR did not provide any drawings or analysis of the DSC seismic restraint, NUTECH did provide responses to the staff's request for

additional information on this restraint. The staff performed an independent check to confirm that adequate shear area is provided by the design. NUTECH should include details of this design in the revision of their TR.

The exceptions noted by the staff in the areas of methodology, material specifications and allowable stresses, as well as the analysis for all systems important to safety, do not result in safety concerns. They are noted in this SER as a matter of record.

The staff reviewed the structural analyses and designs presented in the TR to determine the extent that the TR would satisfy requirements for a site-specific license application SAR as expressed in 10 CFR 72.24. The staff finds that the description and design of the HSM and the integral DSC support assembly, the DSC, and the TC are satisfactory for incorporation in a SAR by reference for the following technical information requirements, to the extent concerned with the structural design, with the exception of site-specific considerations (such as HSM foundation and validation of maximum accident condition parameters):

1. Description and discussion, per 10 CFR 72.24(b).

2. Design, per 10 CFR 72.24(c).

3. Impact on public health and safety, per 10 CFR 72.24(d), but only to the extent of protecting against accidental rupture and/or exposure of nuclear reactor fuel, and not as relates to radiological or other hazards (see other corresponding SER sections).

4. Selection of license conditions and specifications as relate to the structural design, per 10 CFR 72.24(g).

5. Proof of functional adequacy or reliability, per 10 CFR 72.24(i) (requirements related to design or materials).

Section 4.2.1 of the TR lists codes and standards applicable to the fabrication and construction of the components, equipment, and structures identified through the TR. The staff has reviewed these and considers them acceptable with the following exceptions:

1. The reference to the AISC Code Eighth Edition should be to the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," effective November 1, 1978, which is contained in the AISC Steel Construction Manual, Eighth Edition (Reference 15).

2. The ACI 318-83 (Reference 16) code is listed as the code of construction for the HSM in Section 4.2.1 of the TR, and is specified in Appendix E, drawing DUK-03-1000, sheet 1 for construction of the reinforced concrete. This code is not endorsed by Reg. Guide 3.60 or ANSI/ANS 57.9-1984, which cites the ACI 349-80 (Reference 5) code as suitable for concrete design. However, the NRC staff performed an extensive comparison of ACI 318-83 with ACI 349-80 and concluded that <u>for the purposes of construction only</u> the impact on the HSM would be negligible. ACI 349-80 has more stringent quality assurance requirements than ACI 318-83, but the assessment of the staff is that there will be no safety consequences as a result of using ACI 318-83 as the code of construction.

3. Reference to non-specific codes of construction for transfer equipment are not considered appropriate since the designs for such equipment are not included in the TR. The listing of codes for other than the HSM (with DSC support assembly), DSC, or TC does not provide a suitable reference. The appropriate codes should be identified in the site-specific SAR.

3.2 DESCRIPTION OF REVIEW

This section evaluates the structural capability of the HSM, DSC, and transfer cask to withstand loads due to normal operating conditions, off-normal conditions, accident conditions, environmental conditions, and natural phenomena. The review addresses the assumed loads, the material properties, and the allowable stress limits. The review provides an evaluation of the structural analysis in the TR for each of the components and systems important to safety.

The SER review only addresses forces and conditions external to the fuel pool building. Determination of the adequacy of the design for normal, off-normal, and accident conditions within the fuel pool building at a specific site will be addressed as part of the 10 CFR 50 safety review.

3.2.1 Applicable Parts of 10 CFR 72

The parts of 10 CFR 72 that are applicable to the structural evaluation are as follows: 72.122(a), which deals with quality standards; 72.122(b), which requires that structures important to safety be protected against environmental conditions and natural phenomena, as well as appropriate combinations of effects including accident conditions; 72.122(c), which requires protection against fires and explosions; 72.122(f), which requires design to permit inspection, maintenance, and testing; and 72.122(h), which requires protection of the fuel cladding against degradation and gross rupture.

The structural descriptions, analyses, and designs in the TR were reviewed for potentially meeting the requirements of 10 CFR 72.24 for a site-specific license application. Although there is no stated scope for an ISFSI TR, the utility of the TR is in providing already approved elements of site-specific license application documentation, in which it may be incorporated by specific reference (per 10 CFR 72.18) or repetition. The requirements of 10 CFR 72.24, which might be met by the structural descriptions, analyses, and designs of a TR, are section 72.24: "(b) description and discussion of structures;" "(c) design, including criteria, bases, special information, and codes and standards;" "(d) analysis and evaluation of the design and performance;" "(g) subjects for license conditions and technical specifications (per 10 CFR 72.44);" and "(i) need for demonstration of functional adequacy or reliability."

3.2.2 Review Procedure

The TR was reviewed to determine compliance with the applicable parts of 10 CFR 72 outlined above. The systems comprising the NUHOMS ISFSI including the HSM, DSC support assembly, DSC, and transfer cask were considered first as systems and secondly as individual parts making up complete systems. Normal operating conditions, off-normal operating conditions, extreme natural phenomena, and accident conditions and resulting loading combinations that were analyzed by NUTECH were reviewed for completeness by the staff.

3.2.2.1 Design Descriptions

A brief description of the NUHOMS ISFSI was given in the first section of this report. A more detailed description of the design in this section highlights aspects of the design that are important to the structural evaluation.

The safe storage of irradiated fuel is provided by the DSC and HSM. The DSC provides confinement of radioactive material. The HSM and DSC provide shielding for biological protection. The DSC and TC provide shielding during handling and transportation of the fuel. Both the DSC-TC and DSC-HSM combinations must provide for adequate steady-state heat transfer.

The HSM is a reinforced concrete structure that provides projectile impact and weather protection for the DSC and serves as the primary biological shield for the irradiated fuel during storage.

The HSM is designed to be constructed of 5000 psi (minimum specified compressive strength) normal weight (145 pcf minimum density) concrete with Type II Portland cement meeting the requirements of ASTM C150. The aggregate is to meet the specifications of the ASTM C33. The reinforcing steel is #9 bars ASTM A615 Grade 60 spaced 6" on centers each way each face, top, sides, front, back, and foundation.

The HSM wall thicknesses were designed to meet shielding requirements and was checked against structural criteria. The walls protect the DSC against tornado generated missiles, which effectively bounds reasonable impact accidents, as well as other environmental conditions, natural phenomena, and accidents.

The structural properties of the concrete when subjected to the elevated surface temperature for the long term are discussed in Section 3.3.2 of this report.

The HSM is designed to accommodate the transfer of the DSC from and back into the TC. This is provided by an oversize inset steel collar forming the opening of the HSM storage vault. The collar includes an access opening sleeve into which the top of the horizontal TC may be slid. The adjacent external face of the HSM includes connection points to provide a tensile reaction against the piston mechanism used to drive the DSC out of TC and into the HSM.

The DSC end plugs and the canister shell provide confinement and radiation shielding. The bottom end sandwiches lead between an outer plate and an inner plate of Type 304 stainless steel. The bottom end plug also includes a grapple attachment assembly for insertion and removal from the HSM. The top plug is formed by two covers, separately welded to the DSC shell. The inner cover and the outer cover are manufactured from Type 304 stainless steel with lead placed between these cover plates. The DSC ends serve as pressure boundaries. The welds are multiple pass and are to be tested either ultrasonically or by multilevel dye penetrant examination. In addition, a helium leak test will be performed after welding the inner top cover in place.

The DSC shell will be assembled using longitudinal and circumferential full penetration butt welds. These welds are to be fully radiographed and inspected according to the requirements of the ASME B&PV Code Section III, Division 1, Subsection NB (Reference 12). The material is 0.625 inch thick 304 stainless steel.

The canister encloses a basket assembly, which can house twenty-four irradiated fuel assemblies. The basket consists of eight spacer discs of Type 304 stainless steel that are fixed axially by four 3.0 inch diameter 304 stainless steel rods running the length of the canister. There are twenty-four square section guide sleeves of Type 304 stainless steel that house the spent assemblies. The primary structural function of the spacer discs and axial support rods is to maintain dimensional stability for the guide sleeves that house the spent fuel in the event of a vertical or horizontal drop. The axial location of the spacer discs corresponds to the grid spacing of the specific fuel to be stored.

The DSC rests on a fabricated support rail assembly that rests on brackets attached by anchor bolts cast in the interior walls of the HSM. The support rails are also welded to the cast-in-place sleeve forming the HSM front opening. Thermal expansion of the support rails and crossbeams is allowed by using slotted bolt holes. Corrosion of the structural carbon steel will be retarded by either zinc paint, galvanizing, or hard plating.

During loading of the DSC into the HSM, frictional loads between the DSC and the support rails in the HSM will be reduced by the use of a dry film lubricant applied to both sliding surfaces. The particular product selected by NUTECH is Everlube 823, which was designed for radiation service.

The DSC is prevented from sliding longitudinally along the rail during a seismic event by seismic restraints. Permanent restraints are welded to the rails at their inside ends and a removable restraint is attached to the access sleeve at the HSM front opening after placement of the DSC.

A specially designed TC is a major component of the NUHOMS-24P system. The DSC is placed in the TC prior to loading spent fuel rods into it and remains in the TC until it is pushed from the TC into the HSM. The DSC is loaded, sealed, drained, and the TC is drained prior to departure from the fuel rod storage pool enclosure. The DSC will be pulled from the HSM back into the TC for removal and will remain in the TC at least until the fuel rods are removed.

The TC provides radiation shielding and a protective enclosure for the DSC. During transportation from the fuel pool building to the HSM site, the TC provides DBT projectile impact protection. The TC does not provide a pressure boundary in addition to the DSC liner.

The TC design consists of: (1) a bolted-on vented top cover plate, (2) a lower water-tight bolted-on cover sized to permit the grapple of the hydraulic ram to enter and act on the DSC, (3) an annulus between the TC and DSC that can be filled with fluid and drained, and (4) a neutron shielding fluid-filled jacket with external expansion tank.

The TC cylindrical wall section is comprised of a 1/2 inch thick type 304 stainless steel inner shell, 3.5 inches of cast in place lead, a 1.5 inch thick ASME SA-516, Grade 70 structural steel shell, a 3.0 inch radial width fluid chamber, and a .13 inch thick type 304 stainless steel outer neutron shield (tank) shell. The neutron shield is to be filled with a water-antifreeze mixture.

The lower end of the TC forms a radiation shield, provides for fluid retention, and has a bolted-on cover over the access port. The end is formed of type 304 stainless steel 2 inch thick inner and a 3/16 inch thick outer cover plates and approximately 2 3/4 inches of solid neutron shield (Borosilicone (reg) No. 237) in between. The upper TC cover is formed of ASTM A516, Grade 70 steel 3 inch thick inner and 1/4 inch outer cover plates and an intermediate 2 inch thick solid neutron shield (also Borosilicone).

The TC is assembled by welding the concentric cylindrical walls and the lower end to heavy forged ring assemblies made of ASTM SA-182, Type F304N steel at the top and bottom of the cask. Lead is poured to fill the annulus formed by the inner two shells. Hydraulic fittings, tubing, and an external expansion tank permit use of the annulus formed by the middle and outer shells as a water-filled neutron shield.

The TC has two pairs of trunnions. The upper pair is used to lift the cask vertically and to support it while in a horizontal orientation. The lower pair is used for support on the transfer trailer and serves as pivots for rotating between vertical and horizontal orientation (and vice versa) when the TC is on the transfer trailer.

The upper end of the TC fits within a receiving collar at the opening of the HSM to provide continuous radiation shielding during DSC transfer into and out of the HSM from/into the TC.

The TC does not constitute a radioactive material confinement boundary, although it is essential for radiation shielding. As a result, the TC is not required to meet as stringent design criteria as the DSC. The TC is designed to meet the requirements of subsection NC, for Class 2 components, of Reference 17.

3.2.2.2 Acceptance Criteria

The structural integrity of the NUHOMS HSM, DSC, DSC support assembly, and TC are judged adequate if it can be demonstrated that the stresses induced by the loads noted below in Section 3.3.1 are lower than the allowable stress limits for the components important to safety and that all other material properties are consistent with applicable code requirements. The allowable stress limits are documented in the TR (Reference 1, Section 3.2.5, Tables 3.2-4, 3.2-5, 3.2-6, 3.2-7, 3.2-8, and 3.2-9).

3.2.2.3 Review Method

The method of review used to assure that the TR was in compliance with 10 CFR 72 involved several steps. The TR was reviewed first for completeness, to ensure that all areas specified by 10 CFR 72 Subpart F were addressed, and that the standard format, content, and design guidance specified by Regulatory Guides 3.48 (Reference 2) and 3.60 (Reference 4) were followed to the extent applicable for a non site-specific TR to be used in conjunction with subsequent site-specific license applications. Sources cited by the TR were reviewed to determine applicability to the design of the NUHOMS system. Section 3 of the TR, which sets out the design criteria, was examined critically for appropriateness. Assumptions stated in the TR were assessed with respect to those suggested by the professional societies which guide design practice for pressure vessels and reinforced concrete structures for nuclear safety related items. The societies and their respective codes are: the American Society of Mechanical Engineers (ASME Boiler and Pressure Vessel Code for Nuclear Power Plant Components, Section III, Division 1, Subsection NB, Class 1 Components, 1983) for the DSC and the American Concrete Institute (ACI 349-80, ACI 349R-80 and 1984 Supplement to Code Requirements for Nuclear Safety Related Concrete Structures) for the HSM. The design of the DSC support system was compared to requirements of the Manual of Steel Construction published by the American Institute of Steel Construction. The design of the TC was reviewed against the requirements of the ASME B&PV Code (Section III, Subsection NC, Class 2 Components, 1983). The design of the lifting trunnions of the TC was reviewed against the requirements of ANSI N14.6-1986 (Reference 18).

Secondly, Section 8 of the TR, which covers the analysis of the design events, was reviewed in detail. This included verifying all calculations that could be executed without resorting to computer models. The finite element computer models performed by NUTECH were verified for accuracy by examining the input and output printouts for all ANSYS and STRUDL (References 19 and 20) computer runs that were referenced in the TR, and NUTECH post-processor codes. All results that were included in the TR (Tables 8.1-7, 8.1-7a, 8.1-8, 8.1-9, 8.1-9a, 8.1-10, 8.1-10a, 8.1-10b, 8.2-3, 8.2-7, 8.2-7a, 8.2-9, 8.2-9a, 8.2-9b, 8.2-10, 8.2-11, 8.2-12, 8.2-13, 8.2-14, and 8.2-15 (Reference 1)) were either verified by hand calculations or by examining the computer printouts. No independent computer analysis was performed.

3.2.2.4 Key Assumptions

Assumptions made by staff reviewers in verifying NUTECH's models are discussed on a case-by-case basis in the following sections.

3.3 DISCUSSION OF RESULTS

The following evaluation covers loads, materials, stress intensity limits, and structural analyses results.

3.3.1 Loads

The loads specified in the TR for use in designing the NUHOMS system are described in this section together with comments by the staff regarding their acceptability. Loads are described for normal operating, off-normal operating and accident conditions. The staff evaluation of the design criteria sources was summarized earlier in Table 2.1.

3.3.1.1 Normal Operating Conditions

Section 8.1.1 of the TR defines the normal operating conditions of the NUHOMS system. The normal operating loads of the NUHOMS system are dead weight loads, design basis internal pressure loads, design basis thermal loads, operational handling loads, and design basis live loads. The staff evaluation of criteria associated with these loads is summarized in Table

2.2 for the system components, and includes sources and the results of the staff review.

3.3.1.2 Off-Normal Conditions

Section 8.1.2 of the TR defines the off-normal events. These are events that are expected to occur on a moderate frequency. The events included are: a jammed DSC during HSM loading or unloading, and extreme ambient temperatures (-40°F and 125°F). The staff evaluation of criteria associated with each of these loads is summarized in Table 2.3, and includes sources and the results of the staff review.

3.3.1.3 Accident Conditions

Section 8.2 of the TR defines the accident conditions resulting from extreme environmental conditions, natural phenomena conditions, and postulated accidents, which include the following conditions:

1. Loss of HSM air outlet shielding blocks.
2. Tornado winds and tornado generated missiles.
3. Design basis earthquake.
4. Design basis flood.
5. Accidental TC drop with loss of neutron shield.
6. Lightning.
7. Debris blockage of HSM ventilation air inlets and outlets.
8. Postulated DSC leakage.
9. Pressurization due to fuel cladding failure within the DSC.

The staff evaluation of design criteria associated with the accident loads is summarized in Table 2.4, and includes sources and the results of the staff review.

3.3.2 Materials

The mechanical properties of all materials used in the fabrication of components important to safety are listed in Section 8.1.1, Table 8.1-2 of the TR. The source identified in these tables for properties of steel is the ASME Boiler and Pressure Vessel Code, Section III-1 Appendices. This source is an acceptable standard and is in compliance with the quality requirements of 10 CFR 72 Part F.

The source identified in TR Table 8.1-2 for the mechanical properties of concrete and reinforcing steel is the Handbook of Concrete Engineering (Reference 21), a document that is not considered to constitute a standard meeting the quality requirements of 10 CFR 72.122. NUTECH supplements these data by a review of concrete behavior under sustained elevated temperatures that is presented in Appendix D to the TR. The Appendix D data are substantiated by a number of references, most of which are publications of the American Concrete Institute and the Portland Cement Association (PCA). Both of these organizations publish recognized standards consistent with the quality requirements of 10 CFR 72.122(a).

The temperature of some HSM concrete may exceed 150°F under "normal conditions" (sustained ambient temperature up to 100°F), over 200°F under off-normal conditions (ambient temperature to 125°F for 48 hours), and up to 395°F for accident conditions (ambient temperature of 125°F with all vents plugged for 48 hours, see TR Table 8.1-12). The analytical procedures by which these projected temperatures were calculated were reviewed, with appropriate amplification provided by direct informal contact, and were determined to be acceptable.

The ACI Code, ACI 349-85, used for the TR design criteria provides for limits on concrete temperatures as follows: general limit for concrete in structures of 150°F, 200°F for local areas for long terms periods, 350°F for concrete for short term periods, and 650°F for local areas (if due to steam or water jets) in an accident or other short time period. Section A.4.3 of the ACI Code indicates that higher temperatures may be allowed if tests are provided to evaluate the reduction in strength and this reduction is applied to the design allowables, and that evidence be provided that verifies that the increased temperatures do not cause deterioration of the concrete either with or without load.

The TR includes a review of concrete behavior under sustained elevated temperatures (TR Appendix D), and a strength reduction is applied (TR Table 8.1-2, $f_c' = 4.5$ ksi for 5 ksi concrete) and used in the comparison of calculated versus allowable moments (TR Table 8.2-10).

The NRC staff reviewed the submitted discussion of concrete at high temperatures, reduction of allowable stresses (based on ultimate compressive strength), extent of concrete affected, and the ACI Code provisions, and concludes that the HSM design is acceptable with regard to the projected temperatures.

The review also addressed the issue of need for evidence that the continued long term temperatures will not result in degradation of the concrete. It is noted that the HSM for the NUHOMS-7P design is projected to have higher operating temperatures than the HSM which is addressed by this SER and that a condition of the NRC approval of Reference 22 is that results of field tests on the concrete be submitted after that system is placed in use.

The NRC staff considers that because of the following results, there is no requirement for subsequent or further submission of test data on possible concrete degradation under elevated temperatures for the NUHOMS-24P system HSM: 1) the relatively small concrete areas affected by elevated (over 150^{o}F) temperatures, 2) the small magnitude of the elevated temperature, and 3) the empirical data on concrete behavior on-hand. This is subject to further NRC review if data or analyses made available to the NRC after final action on this TR suggest that a problem could exist.

The sources identified in TR Table 8.1-2 for the structural properties of lead are not recognized standards consistent with the quality requirements of 10 CFR 72.122(a). However, the material strength properties for lead shown in the TR were used in a conservative way that would not invalidate the analysis. No bending stiffness is assumed to be imparted by lead shielding plug because coupled nodes are used at the interface of lead and steel. This coupling of nodes permits only tension or compression and no shear forces to be transmitted through the lead. Thus the staff concludes that the way the data were used meets the intent of the quality requirements of 10 CFR 72.122(a) for material properties.

3.3.3 Stress Intensity Limits

The mechanical properties of the structural materials used in the design of the NUHOMS system are listed in Table 8.1-2 of the TR. These properties, including allowable stress intensities, are listed as a function of temperature for a variety of materials as described below.

The ASME Boiler and Pressure Vessel Code, Section III-1, was identified in Table 8.1-2 of the TR as the source for stress allowables for the two Type 304 stainless steels, the SA-516 carbon steel, the SA-564 and SA-533 alloy steels used for the DSC and TC shell and trunnions, Grades A36 and A325 steel and SA-193 alloy steel used for bolts. As discussed in Section 3.3.2 of this SER, these stress allowables are acceptable to the staff for use in the design of the NUHOMS system.

The Handbook for Concrete Engineering is identified in Table 8.1-2 of the TR as the source for stress allowables for concrete and reinforcing steel. As discussed in Section 3.3.2 of this SER, the staff does not concur in the use of this handbook as the authoritative source for concrete and reinforcing steel allowable stresses. However, the staff has reviewed the pertinent ACI and PCA data and concurs in the stress allowable values as presented in Table 8.1-2 of the TR for these materials.

3.3.4 Structural Analysis

3.3.4.1 HSM

A linear elastic structural analysis of a one foot section of a ten bay HSM was performed, using the STRUDL finite element computer program to determine the worst internal forces due to normal, off-normal, environmental and accident loading conditions. The combinations of the resultant forces were performed based on the requirements of ANSI-57.9-1984.

The staff reviewed and accepts the finite element modeling techniques of the HSM reinforced concrete structure. The following presents an overview of the evaluation.

3.3.4.1.1 Normal Operating Conditions

The HSM concrete structure was analyzed for the effects of dead loads (including effects of creep and shrinkage), live loads, and design basis temperature loads. In addition, the HSM door was analyzed for dead loads and normal handling loads. The HSM door supports are not designed to withstand dropping the door during closing or opening. The NRC staff reviewed this situation and determined that failure of the supports and possible drop of the door to the ground level would not constitute a safety problem. The staff did determine that the door and support design and method of locking (welding) are acceptable.

The HSM concrete structure analysis results are presented in TR Section 8.1.1.5 and Table 8.1-10 for all of the load combinations considered. These results are presented in the form of maximum moments and shears, which are compared with the ultimate moment and shear capacities of the respective structural section. The maximum moments and shears are developed for normal conditions in load combinations 1, 2, and 4 of TR Table 8.2-10 (included in this SER as Table 3.1). It is seen from this table that these maximum moments and shears are significantly lower than the associated capacities of the module.

HSM dead and live load analyses

The dead weight of HSM plus the weight of DSC and the DSC support assembly were considered. The actual concentrated loads (reactions) due to the dead weight of the DSC and the DSC support assembly were calculated and reported in Table 8.1-9 of the TR. However, these loads were not used in the finite element analysis as stated in TR Section 8.1.1.5.A. Instead, one-sixth of the total weight of the DSC was applied at the embedded support connection. Even though there is almost a factor of two difference in the actual load and the one-sixth estimated loads, the staff accepted this condition since only the properties of a one foot section of the HSM were used. The staff used the actual load to perform hand calculations to verify suitability of the design. The vendor performed a series of computer runs based on alternating loaded and empty HSMs to determine the worst set of internal HSM forces. The dead weight of the HSM was not considered as was stated in TR Section 8.1.1.5.A. The weight of the reinforced concrete was distributed throughout the members of the finite element model.

A live load of 200 psf was applied to the HSM roof to envelope all live loads. The resulting calculated maximum dead and live loads are tabulated in Table 8.1-10 of the TR. The staff reviewed the tabulated results as well

Table 3.1 SUMMARY REVIEW OF HSM STRUCTURAL DESIGN

HSM ENVELOPING LOAD COMBINATION RESULTS
(Table 8.2-10, Reference 1)

Load[1] Combination		Loading Combination Description	Maximum Loading		Capacities[7]		NRC Staff Comments
			V_{max} (k)	M_{max} (k.in.)	V_u (k)	M_u (k.in.)	
1	Norm	1.4D + 1.7L	4.8	233	43.8	3570	Acceptable
2	Norm	1.4D + 1.7L + 1.7H	4.8	233	43.8	3570	Acceptable
3	Accid	0.75(1.4D + 1.7L + 1.7H + 1.7T + 1.7W)	34.5	867	43.8	3570	Acceptable
4	Norm	0.75(1.4D + 1.7L + 1.7H + 1.7T)	34.5	867	43.8	3570	Acceptable
5	Accid	D + L + H + T + E	42.8	1220	43.8	3570	Acceptable
6	Accid	D + L + H + T + F	40.9	1100	43.8	3570	Acceptable
7	Off-normal & Accid	D + L + H + T_a	79.4	2800	104.[8]	3570	Acceptable

D = Dead Weight, E = Earthquake Load, F = Flood Induced Loads, H = Lateral Soil Pressure Load, L = Live Load, T = Normal Condition Thermal Load, T_a = Off-normal or Accident Condition Thermal Load, W = Tornado Wind and Missile Loads — Acceptable

Notes:

1. Load combinations are based on ANSI-57.9 as shown in Table 3.2-5 of the TR. — Acceptable
2. Maximum loads shown are irrespective of locations. — Acceptable
3. Thermal accident load (T_a) is based on 125°F ambient with air inlets and outlets blocked. — Acceptable
4. V_{max}, V_u, M_{max}, and M_u calculated per 12" section of HSM. — Acceptable
5. Results of load combinations 3 through 7 are based on cracked section. Others based on uncracked sections. — Acceptable
6. Material properties taken at 400°F for all load combinations. — Acceptable
7. V_u values based on allowable shear for deep flexural members, ACI 349-85, Section 11.8. [Note: ACI 349-80 identical.] — Acceptable
8. The shear capacity V_c is calculated using Equation 11-29 of ACI 349-85. [Note: ACI 349-80 identical.] — Acceptable

as the computer output, and concurs with the results as summarized in Table 3.1 of the SER.

Concrete creep and shrinkage analysis

The strains due to creep and shrinkage were calculated and then the total axial change in length were calculated for the HSM members. From a knowledge of the axial change in length, it was possible to calculate the axial forces. The TR does not document the method used for determining the column in Table 8.1-10 of the TR for "creep effects." The staff discussed this problem with the vendor's contractor and reviewed the computer inputs of the dead weight load case and found that the member forces due to creep and shrinkage were combined with the HSM dead weight load case where these effects increased the calculated forces. The staff accepted the approach and the load case.

HSM thermal analysis

Analyses were performed for ambient operating temperatures of $0^{o}F$, $70^{o}F$ and $100^{o}F$. The results of the heat transfer analysis of the $100^{o}F$ ambient with solar heat flux case indicated a maximum local temperature of $179^{o}F$ as shown in Figure 8.1-3a of the TR. The results are tabulated in TR Table 8.1-10. This localized temperature is within the allowable limits of ACI 349-85, Section A.4.1.

From a knowledge of the thermal gradient from the heat transfer analysis, the internal axial forces and bending moments were calculated and applied to the members of the HSM model to determine resultant forces and moments. The results are based on the uncracked section properties of concrete. The cracked section moment of inertia of the HSM members were calculated and then the ratio of cracked to uncracked moment of inertias were multiplied by the moment values taken from the computer output to determine the actual moment values of the cracked sections for the thermal load case. The approach was found acceptable. The staff reviewed the calculations, checked tabulated results in TR Table 8.1-10 against the computer output and found them acceptable.

The staff concludes that the structural design of the HSM for normal operating conditions is acceptable.

3.3.4.1.2 Off-Normal Conditions

The effect of increased temperatures due to high ambient temperatures (125°F) was the only off-normal event considered in the TR to have an effect on the HSM. A thermal analysis was performed for this event as described in Section 4.3.2.2 of this SER. The results from this thermal analysis were used to perform a structural analysis on the HSM, as reported in Section 8.1.2.2 of the TR. Load combination 7 on Table 3.1 is applicable to this condition; however, the effects of elevated temperature due to the accident condition of vent blockage envelop this off-normal condition.

It is stated in Section 8.1.2.2 of the TR that this structural analysis considered the effect of a cracked cross-section when performing stiffness calculations. The staff agrees with the use of this cracked section analysis procedure since it is permitted as a special case in the ACI 349-80 Code. The staff has reviewed the procedure used to perform this cracked section analysis together with a review of the special conditions placed on its use by the ACI 349-80 code. The staff concludes that the results from this stress analysis are acceptable.

The only other off-normal event considered in the TR is the jamming of a canister against either the DSC support or HSM components during loading or unloading. Although the effect of this off-normal event on the DSC and the DSC support assembly within the HSM is considered in the TR, there is no actual analysis of this event for the HSM reinforced concrete structure. The staff has performed the analysis for the effect of this off-normal event on the HSM and concludes that the effect of this loading is negligible.

3.3.4.1.3 Accident Conditions

<u>HSM accident load analysis</u>

The postulated accident conditions for design specified by ANSI/ANS 57.9-1984 and other credible accidents which could affect the safety of the NUHOMS system were considered. The postulated accident conditions addressed in the TR include:

1. Loss of HSM air outlet shielding blocks.
2. Tornado winds and tornado generated missiles.
3. Design basis earthquake.
4. Flood.
5. Lightning.
6. Debris blockage of the HSM ventilating air inlets and outlets.

Loss of HSM air outlet shielding blocks

Air outlet shielding blocks are the only components of the NUHOMS system that are not designed to withstand tornado generated missiles. The vendor argues that there are no structural or thermal consequences of the loss of the shield blocks from the HSM. Increases to off-site radiological dose are discussed in Section 10 of this SER. The staff reviewed the drawings of the shielding blocks as shown in the TR and concurs that minimal structural damage to the HSM will result if the shield blocks are lost due to tornado generated missiles. There may be some difficulty in replacing the shield blocks if the method of attachment is via bolts and the bolts are damaged by the tornado. There may be much more difficulty encountered in replacing the shield blocks if they are cast in place. As the drawings supplied with the TR do not indicate the type of attachment, the question will need to be addressed in a site-specific application.

Tornado winds and tornado missiles

The analyses performed to evaluate the effect of tornado winds and tornado missiles is presented in Section 8.2.2 of the TR. The tornado wind analysis includes an evaluation of the possible overturning of an unanchored module and the computation of wind induced maximum moments and shears in an anchored module. Design basis for this postulated accident analysis was taken from NRC Regulatory Guide 1.76 and NUREG-0800 3.5.3 and 3.5.1.4 (References 23 and 24). The bending moments and shear forces at critical locations in the HSM member were calculated by performing a linear elastic finite element analysis. The resulting moments and shear forces are tabulated in TR Table 8.2-3.

The analysis performed to evaluate potential sliding and/or overturning of an unanchored module showed that a single unanchored module (or multiple modules) would not overturn or slide when subjected to the tornado wind event. Tie-downs or anchorage between the HSM and its foundation would still be used to reduce the potential risk of sliding. The NRC staff concurs with the analyses that anchors (e.g., monolithic construction, reinforcing steel) should be used.

The computation of wind induced maximum moments and shears was performed using selected critical sections and finite element analysis. The maximum moments and shears resulting from this analysis are presented in TR Table 8.2-3 and are included in the combination of loads analysis (combination 3 on Table 3.1). The NRC staff considers the analyses and results to be acceptable.

Analyses are included in TR Section 8.2.2.2 to evaluate the effect of a penetrating missile and a massive missile impact on the HSM. These analyses consider the impact of a 276 pound, 8-inch diameter blunt-nosed steel object on the HSM walls/roof and on the 3-inch thick steel door at the front of the HSM, and evaluate the impact of a 3,976 pound automobile on the side wall of the HSM. The outer walls and roof of the HSM are not less than 36" thick reinforced concrete. The staff performed hand calculations to determine equivalent static impact force and then the maximum bending moments. The staff reviewed and accepts the analyses of each postulated case and the results.

Design basis earthquake

Analyses performed to evaluate the effect of earthquakes on the HSM are presented in Section 8.2.3 of the TR. These analyses include an evaluation of the possible overturning and sliding of an unanchored single module and the computation of seismically induced maximum moments and shears in an anchored module.

A horizontal acceleration of 0.25 g and a vertical acceleration of 0.17 g were used as bases for seismic design. To evaluate the seismic response of the HSM, an equivalent static seismic analysis was performed. To determine the HSM fundamental frequency, the STRUDL-DYNAL finite element model of single section of HSM was developed. Based on the computer output, the lowest fundamental frequency was 25 Hz.

The corresponding vertical and horizontal accelerations for 25 Hz were multiplied by a factor of 1.5 for the members of the HSM finite element model. The values of the acceleration used for the analysis were slightly higher than the actual calculated values for the HSM members. Even though the calculated values of the DSC support structure acceleration are higher than the acceleration values for the HSM structure, the values for the HSM structure were used to determine the seismic forces for the DSC support structure. The staff reviewed this discrepancy and accepts the resulting shear forces and moments tabulated in TR Table 8.2-3 since the combination is summed absolutely.

The maximum moments and shears resulting from the seismic analysis of the HSM are shown in TR Table 8.2-6. They were used in the combination of loads design validation, shown in Table 3.1 of this SER (combination 5). The analyses and results are considered to be acceptable to the NRC staff.

The analyses performed to evaluate potential sliding and overturning of an unanchored module showed that a single unanchored module would not either overturn or slide when subject to the design earthquake. The staff has reviewed these analyses and concurs with their results.

HSM flooding analysis

The analysis performed to evaluate the effect of flood on the HSM is presented in Section 8.2.4 of the TR. The analysis assumed a 50 foot static head and 15 fps maximum flow velocity. This analysis demonstrates that a single, unanchored, submerged module (or multiple modules) would not slide or overturn under the design conditions. Maximum shears and moments due to flood forces were calculated and used in the combination of loads expression (Table 3.1, combination 6). The NRC staff has reviewed and concurs with the analyses and results. However, each site-specific application must validate that its potential flood parameters are within those assumed in the TR or a separate analysis would be required.

Lightning

The lightning protection will be provided at the site. The TR also states that resulting current surge from the lightning will not affect the normal operating condition of the HSM. The staff agrees and accepts the statements in Section 8.2.6 of the TR.

Blockage of HSM ventilation air inlets and outlets

The analysis performed to evaluate the effect of air inlet and outlet blockage is presented in Section 8.2.7 of the TR. Section 8.2.7.1 states that the structural consequences due to the weight of debris blocking the air inlets and outlets are bounded by the structural consequences of tornado and earthquake accidents. The staff agrees with this statement.

An analysis was performed to determine the effect of high temperatures caused by the blockage of both air inlets and outlets on the structural behavior of the HSM. The results from this analysis were used in the combination of loads (Table 3.1, combination 7).

The complete blockage of the HSM ventilation air inlets and outlets was considered as an accident condition. The thermal effects of this accident result from the increased temperatures of the DSC and the HSM at extreme ambient condition of 125^{o}F. NUTECH evaluated this blockage for 48 hours, at which time it was assumed that corrective action would be completed and natural circulation air flow would be restored to the HSM.

At the end of the 48 hours of blockage, the maximum HSM inside surface temperature was calculated to be 395^{o}F. The staff reviews conclude that this high temperature is an acceptable temporary localized condition for the HSM based on the limitation from ACI 349-85-A.4.2. NUTECH's contractor calculated the linear thermal gradients and then calculated the fixed moments and forces. These were input to the STRUDL finite element model to determine the internal forces in the HSM members. Then the internal forces and moments were modified to account for the concrete cracked section properties.

The resultant calculated thermal moment and shear forces are tabulated in Table 8.2-3 of the TR. The staff reviewed the calculations, checked the tabulated results against results from the computer output, and find them acceptable.

HSM load combination

The maximum bending moments and shear forces due to normal and off-normal loads are listed in TR Table 8.1-10. Similarly, the results due to the accident loads are listed in TR Table 8.2-3. The combination of the resultant bending moments and the shear forces was performed based on the requirements of ANSI-57.9-1984 and the results are tabulated in TR Table 8.2-10.

The combinations were checked against the calculated ultimate capacities of the concrete at $400^{o}F$ based on the formulas from ACI 349-85. The staff agrees and accepts the load combination results.

3.3.4.2 DSC and Internals

3.3.4.2.1 DSC Normal Operating Conditions

The dry shielded canister was analyzed for: (1) dead weight loads, (2) design basis internal pressure, (3) design basis operating temperature loads, and (4) normal operation handling loads. The canister internal parts were analyzed for: (1) dead weight loads, and (2) design basis operating temperature loads. Table 3.2 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 NUTECH and the corresponding stress as verified by the NRC staff. Each stress value was compared to the allowable stress intensity 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.

Table 3.2

DSC STRESS ANALYSIS RESULTS
FOR NORMAL LOADS
Service Level A

DSC Component	Stress Type	Stress (ksi) Dead Weight NUTECH	Dead Weight NRC	10.7 psig Int. Pressure NUTECH	10.7 psig Int. Pressure NRC	$100^{\circ}F$ Thermal NUTECH	$100^{\circ}F$ Thermal NRC	Normal Handling NUTECH/NRC	Allowable* Level A & B
DSC Shell	Pri Memb	0.1	0.2	0.5	0.6	N/A	N/A	0.2	18.7
	Memb + Bend	3.7	13.4	0.5	0.6	N/A	N/A	1.8	28.0
	Pri + Second	3.7	13.4	6.2	6.6	17.5	17.8	-	56.1
Inner Top Cover Plate	Pri Memb	0.1	0.7	0	0	N/A	N/A	0.1	18.7
	Memb + Bend	0.5	0.5	4.6	4.6	N/A	N/A	0.3	28.0
	Pri + Second	0.2	0.2	3.4	7.2	0.3	0.3	N/A	56.1
Outer Top Cover Plate	Pri Memb	0.1	0.2	N/A	N/A	N/A	N/A	0.1	18.7
	Memb + Bend	0.4	0.4	4.6	4.6	N/A	N/A	0.3	28.0
	Pri + Second	0.2	0.2	3.4	7.2	1.0	1.0	N/A	56.1
Bottom Cover Plate	Pri Memb	0.1	1.2	0	0	N/A	N/A	0.7	18.7
	Memb + Bend	0.3	0.3	1.	1.	N/A	N/A	1.6	28.0
	Pri + Second	0.3	0.3	0.5	1.5	1.7	4.	0.8-2.4	56.1
Spacer Disc	Pri Memb	0.5	0.5	0	0	N/A	N/A	0	18.7
	Memb + Bend	0.3	0.6	N/A	N/A	46.5	46.5	N/A	56.1

*Allowable stress for Service Levels A and B

Primary Membrane	Sm	= 18.7 ksi
Primary Memb + Bend	1.5 Sm	= 28.0
Primary + Secondary	3.0 Sm	= 56.1
Shell, Disc and end plates for $400^{\circ}F$	SA 204 Type 304	

Dead weight loads for DSC

The dead load analysis for the DSC is presented in Section 8.1.1.2.A of the TR. Both beam bending and shell bending were considered. For the beam bending, a two-span continuously loaded beam, simply supported at three locations corresponding with the DSC support structure, was assumed. The maximum membrane plus bending stress for this condition is 200 pounds per square inch or 0.2 ksi, which is below the 18.7 ksi for ASME allowable stress for Service Level A. The canister was also modeled for local shell bending by considering that the total dead weight was supported uniformly by the two continuous T-section support rails. The NRC staff checked the reference cited in the TR and concludes that it is not a conservative model because shape of the elastic deformation in Bednar (Reference 25) is not consistent with the actual deformed shape caused by two support rails. The NRC staff used a more conservative approach from Roark (Reference 26). The results shown in Table 3.2 are below the ASME Code allowable stress.

Design basis internal pressure

Table 8.1-4 of the TR shows eight cases for operating and accident pressures. NUTECH used the ANSYS (Reference 19) finite element code to model the internal pressure load for the top and bottom portions of the DSC. NUTECH used 1 psig for the internal pressure and then multiplied the stress results by a factor corresponding to the particular load case per TR Table 8.1-4.

Figures 8.1-5 and 8.1-6 of the TR show how NUTECH used symmetry to model the top and bottom portions of the DSC. It is seen that a single element was used to model the thickness of the steel shell, as well as the inner and outer top cover plates and the lower cover plate. The ANSYS user's manual describes the particular element type that NUTECH used as a "two-dimensional isoparametric element," which has two degrees of freedom at each node. It was used by NUTECH as an axisymetric ring element. In this configuration, the computer code only calculates membrane stresses. It is possible to calculate bending stresses with this element, provided two or more elements are used through the thickness of the shell. NUTECH did not model the DSC by using two elements in the thickness. Therefore, none of the bending stresses shown in the TR summary tables are, strictly speaking, bending stresses. Membrane and shear stresses are calculated at the centroid of the element and referred to the edge face and/or to the node as an output option. NUTECH used these output options to "estimate" bending stresses. Section III of the ASME Code requires that bending as well as membrane stresses be evaluated for Class 1 components.

For the internal pressure case, the DSC has local bending stresses at the gross structural discontinuity between the thick cover plates and the shell and also in the middle of the flat end plates. The NRC staff calculated the shell bending stresses at the shell/end plate discontinuity. The method used is given in Roark (Reference 26, p. 465). The result was 1.1 ksi for bending stress. NUTECH reported 6.2 ksi by using the stresses referred from the centroid of the shell element to the inside face. From this procedure, the NRC staff concludes that although the ANSYS program does not calculate bending stresses, the stress used by NUTECH is higher than the bending stress calculated by the NRC staff. In both cases, the stress is substantially below the Code allowable for primary plus secondary stresses for Levels A and B (56.1 ksi).

Similar checks were made by the NRC staff for bending stresses in the DSC inner and outer top cover plates and the bottom cover plate. In all cases the calculated stresses are below the allowable level.

As a final observation regarding NUTECH's computer modeling of the top and bottom portions of the DSC, the NRC staff noted that the thicknesses of the plates as modeled in the computer analyses do not agree with the thicknesses of the plates in the design drawings. In order to predict approximately correct stresses for the plates in question, the NRC staff multiplied the stresses as listed in the computer output by the ratio of the squares of the thicknesses involved. These stresses are shown in the summary Table 3.2 of this SER in the columns headed by "NRC."

Design basis operating temperature

NUTECH 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, they have sized the spacer disc smaller than the inside diameter of the DSC shell to preclude induced thermal stresses. However, NUTECH did perform five different finite element analyses to determine thermal stresses for differential expansion of the shell, the spacer disc, and the shell/end-cover interface. These analyses were performed at ambient conditions of 100°F, except for one case where the shell was analyzed at 125°F ambient temperature.

The thermal stresses are always defined as "secondary stresses" by the ASME 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 100°F, the maximum primary plus secondary stress for all thermal cases considered is 46.5 ksi for the spacer disk. The allowable stress is 56.1 ksi. The staff has reviewed all the documentation provided with the TR and concurs that thermal stresses for the DSC for normal operations meet ASME Code requirements. They are shown in Table 3.2 of the SER.

Operational handling loads for DSC

The only normal operational handling load considered by NUTECH was due to the axial force of 20,000 pounds due to the hydraulic ram acting against the DSC bottom assembly. The resulting stresses are much lower than allowable stress as shown in Table 3.2.

DSC internal basket analyses

Section 8.1.1.3 of the TR discusses the stress analysis considerations of the basket components, i.e., the spacer disc, the 24 guide sleeves and the four 3-inch diameter support rods. The spacer disc was analyzed using a finite element program for the 75 g drop case. For the normal dead weight, the stress levels were divided by 75. The results show stress values lower than the Code allowables.

Because the axial location of the spacer discs coincides with the grid spacers of the fuel assemblies, the weight of the fuel assemblies is transmitted directly to the spacer disc. Thus the guide sleeves and support rods only have to resist their self weight, which is trivial for the spacer disc spacing of 21 inches.

3.3.4.2.2 DSC Off-Normal Events

Three off-normal events were evaluated by NUTECH for the DSC. They were off-normal pressure, jammed DSC during transfer and off-normal temperature. The off-normal temperature of 125^{o}F ambient and the jammed DSC bound the range of loads.

Jammed DSC during transfer

The basis for the postulated off-normal event involving jamming of the DSC during transfer into the HSM is axial misalignment of the DSC. Should this occur, the hydraulic ram could exert an axial force equal to the static weight of the DSC of 80,000 pounds, before a relief valve would prevent further load. The bending stress in the bottom cover plate of the DSC 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 3.3 of this report.

Binding of DSC during transfer

A variation of the jammed case involves an angular misalignment of the DSC with respect to the HSM. This condition also results in stresses lower than the allowables.

DSC off-normal thermal/pressure analysis

The off-normal temperature range was taken as -40^{o}F to 125^{o}F for the DSC inside the HSM. The off-normal thermal analysis is the basis for higher thermal stresses for the spacer disc, and the cause of higher internal pressures causing higher shell and end plate stresses. The table in the TR which reports these stresses (Table 8.1-7a) does not, in fact, show higher thermal or pressure stresses for any component except the DSC shell for the thermal event. Pressure stresses are shown to be constant and thermal stresses for the spacer disc are also shown constant. (Compare TR Tables 8.1-7 and 8.1-7a.)

Table 3.3

DSC STRESS ANALYSIS RESULTS
FOR OFF-NORMAL LOADS
Service Level B

DSC Component	Stress Type	Stress (ksi): Internal Pressure 10.7 psig		Thermal 125^{o}F		Off-Norm. Hand		Allowable*
		NUTECH	NRC	NUTECH	NRC	NUTECH	NRC	
DSC Shell	Pri Memb	0.5	0.6	N/A	N/A	1.2	1.2	18.7
	Memb + Bend	0.5	0.6	N/A	N/A	7.0	7.0	28.
	Pri + Second	6.8	6.6	20.9	21.6	-	-	56.1
Inner Top Cover Plate	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	4.6	4.6	N/A	N/A	0	0	28.0
	Pri + Second	3.4	7.2	0.3	1.3	0	0	56.1
Outer Top Cover Plate	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	4.6	4.6	N/A	N/A	0	0	28.0
	Pri + Second	3.4	7.2	1.0	1.8	0	0	56.1
Bottom Cover Plate	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	1.0	1.	N/A	N/A	6.5	6.5	28.0
	Pri + Second	0.5	1.5	1.7	4.4	3.1	9.5	56.1
Spacer Disc	Pri Memb	0	0	N/A	N/A	0	0	18.7
	Memb + Bend	N/A	N/A	46.5	50.7	0	0	56.1

*Allowable stress is taken for Service Level B for SA 204 Type 304 material at 400^{o}F.

The NRC staff evaluated the TR and concluded that NUTECH did not perform a finite element analysis for the spacer disc for the higher temperature. In order to estimate the higher thermal stress, the staff multiplied the thermal stress for the normal case by 1.09, a factor obtained by comparing DSC outer surface temperatures for 100°F and 125°F ambient conditions (see Table 8.1-12 of TR). The assumption made by the staff is that the thermal stresses are linearly proportional to the temperature. The resulting higher estimated thermal stress is 50.7 ksi as shown in Table 3.3 of the SER. This level is still lower than the allowable of 56.1 ksi.

For the pressure stresses, the staff concluded that NUTECH did not make a separate evaluation for the higher pressure due to the off-normal case. Although the NUTECH documentation is not accurate, the staff can accept the results in TR Table 8.1-7a for the pressure stress since they are well below allowables. This conclusion is based on the very small difference in internal pressure of the helium for the off-normal case. The partial pressure is 5.9 vs. 6.1 psig for 100°F and 125°F, respectively.

<u>DSC load combinations for normal and off-normal conditions</u>

Table 3.2-5a of the TR outlines the different load combinations considered for normal and off-normal conditions. These conditions correspond to Service Levels A and B of the ASME Code. Altogether there were four combinations for both service levels; however, due to the fact that NUTECH did not present data in their TR for the off-normal thermal case and off-normal pressure case, the NRC staff combined load combinations A3 and A4, as well as B2 and B3 for the purposes of presenting the results shown in Table 3.4 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.

3.3.4.2.3 DSC Accident Conditions

Section 8.2 of the TR defines the accident conditions associated with the 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 transfer cask with the DSC inside. Of these accidents, the drop case is by far the most severe. NUTECH

Table 3.4

DSC LOAD COMBINATIONS FOR
NORMAL AND OFF-NORMAL OPERATING CONDITIONS

DSC Component	Stress Type	Stress (ksi) Case A1 NUTECH	Case A1 NRC	Case A2 NUTECH	Case A2 NRC	Case A3/A4[1] NUTECH	Case A3/A4[1] NRC	Case B1 NUTECH	Case B1 NRC	Case B2/B3[2] NUTECH	Case B2/B3[2] NRC	Case B4 NUTECH	Case B4 NRC	Allowable Level*
DSC Shell	Pri Memb	0.1	0.2	0.6	0.8	0.8	1.0	1.8	2.0	1.8	2.0	0.6	0.8	18.7
	Memb + Bend	3.7	13.4	4.2	14.0	6.0	15.8	11.2	21.0	11.2	21.0	4.2	14.0	28.0
	Pri. + Second	3.7	13.4	27.4	37.8	27.4	37.8	27.4	37.8	30.8	41.2	30.8	41.6	56.1
Inner Top Cover Plate	Pri Memb	0.1	0.7	0.1	0.7	0.2	0.8	0.1	0.7	0.1	0.7	0.1	0.7	18.7
	Memb + Bend	0.5	0.5	5.1	5.1	5.4	5.4	5.1	5.1	5.1	5.1	5.1	5.1	28.0
	Pri + Second	0.2	0.2	3.9	7.7	3.9	7.7	3.9	7.7	3.9	8.7	3.9	8.7	56.1
Outer Top Cover Plate	Pri Memb	0.1	0.2	0.1	0.2	0.2	0.3	0.1	0.2	0.1	0.2	0.1	0.2	18.7
	Memb + Bend	0.4	0.4	5.0	5.0	5.3	5.3	5.0	5.0	5.0	5.0	5.0	5.0	28.0
	Pri + Second	0.2	0.2	4.6	8.4	4.6	8.4	4.6	8.4	4.6	9.2	4.6	9.2	56.1
Bottom Cover Plate	Pri Memb	0.1	1.2	0.1	1.2	.8	1.9	0.1	1.2	0.1	1.2	0.1	1.2	18.7
	Memb + Bend	0.3	0.3	1.3	1.3	2.9	2.9	7.8	7.8	7.8	7.8	1.3	1.3	28.0
	Pri + Second	0.3	0.3	2.5	5.8	3.3	8.2	5.6	15.3	5.6	15.7	2.5	6.2	56.1
Spacer Disc	Pri Memb	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	18.7
	Memb + Bend	0.3	0.6	46.8	47.1	46.8	47.1	46.8	47.1	46.8	51.3	46.8	51.3	56.1

*Allowable stress is taken for Service Levels A and B for SA 204 Type 304 Material at 400°F.

1. Load cases A3 and A4 were combined into one case because the stresses for the normal and off-normal pressure cases were not supplied by NUTECH.
2. Load cases B2 and B3 were combined into one case because the stresses for the thermal case with the DSC inside the cask or inside the HSM at $T_{ambient}$ = 125°F were not supplied by NUTECH.

classified the thermal accidents and the drop accidents as Service Level D conditions and the remaining accidents 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 Code does not require any stress analysis because of the ASME definition of thermal stresses as "secondary" stresses or "self-relieving" stresses. The only way in which NUTECH was required to give any consideration of the accident thermal cases was in a reduction of material properties at the higher temperature.

Following is a discussion of the results of the accident review.

DSC seismic condition

NUTECH considered the response of the DSC to a seismic event when it is resting on the two support rails. They first performed a rigid-body stability analysis to show no possibility of roll-out. For this purpose they used a factor of 1.5 times .25 g and .17 g for the horizontal and vertical accelerations. The 1.5 factor accounted for the elevation of the DSC in the HSM. No roll-out was possible.

Next NUTECH calculated the natural frequency of the shell ovaling mode and the beam bending mode of vibration. Since the frequency for the ovaling mode was 13.8 Hertz, NUTECH applied an amplification factor of 2.5. (See Regulatory Guide 1.60, Reference 27). The resulting spectral accelerations were 1.0 g and 0.68 g for horizontal and vertical directions, respectively. To account for possible multi-mode excitation, NUTECH used a "safety factor" of 1.5. The total equivalent static load factor used to simulate the seismic event was 1.0 g for the vertical direction and 1.5 g for the horizontal direction.

The stress intensities for the 1 g vertical case were calculated by factoring the dead load analysis results by 1.0. The stress intensities for the 1.5 g horizontal case were calculated by assuming that the DSC is supported by a single T-section rail inside the HSM. Lateral bending stresses were summed absolutely with vertical bending stresses to obtain a combined stress of 21 ksi. The NRC used a more conservative model for lateral bending (Reference 26, Table 5, case 1) and obtained 27.7 ksi. Both values are below the 33.7 ksi allowable for Service Level C.

DSC flood condition

The flood conditions postulated by NUTECH consisted of a 50 foot static head of water and a 15 foot per second flow velocity. It will be necessary for each license applicant to demonstrate that these conditions bound the flood conditions for each individual site.

The static head resulted in a 21.7 psi external pressure which caused 1.2 ksi stresses in the DSC shell and 19.4 ksi and 9.9 ksi stresses in the outer top cover plate and bottom cover plate, respectively. These stress levels are below the 33.7 ksi allowable levels for Service Level C. The NRC calculations as well as the NUTECH calculations are reported in Table 3.5 of this SER.

DSC accident pressure

The bounding DSC internal accident pressure is 49.1 psig according to Table 8.1-4 of the TR. This internal pressure could occur if the transfer cask neutron shield were lost during transfer operations on a day when the ambient air temperature is 125°F. Further assumptions were that all cladding failed and that 100% of the fill gas and 30% of the fission gas were released inside the DSC. Under these unlikely conditions, the internal pressure could reach 49.1 psig. Table 3.5 of this SER shows the stress results of this case. All stress intensities are lower than the allowables.

It should be noted that NUTECH used 400°F as the appropriate temperature to select the allowable stresses for the materials in the DSC (TR Table 8.2-9c). However, Table 8.1-13 of the TR indicates that the DSC shell reaches a maximum temperature of 513°F for this accident; therefore, the NRC staff used lower material allowable stresses for this case and all accident load combinations that have this load case as a part of the load combination.

Table 3.5

DSC STRESS ANALYSIS RESULTS
FOR ACCIDENT CONDITIONS
Service Level C

DSC Component	Stress Type	Stress (ksi) Seismic NUTECH	Seismic NRC	Flood 50' NUTECH	Flood 50' NRC	Accident Pressure 49.1 NUTECH	Accident Pressure 49.1 NRC	Thermal 125°F[1] NUTECH	Thermal 125°F[1] NRC	Accident Handling NUTECH	Accident Handling NRC	Allowable*
DSC Shell	Pri Memb	-	-	1.2	1.2	2.6	2.6	N/A	N/A	1.2	1.2	22.4
	Memb + Bend	21.0	27.7	-	-	6.5	2.6	N/A	N/A	7.0	7.0	33.7
	Pri + Second					31.2	30.6	N/A	N/A			N/A
Inner Top Cover Plate	Pri Memb	-	-		-	0	0	N/A	N/A	0	0	22.4
	Memb + Bend	-	-		-	23.2	23.1	N/A	N/A	0	0	33.7
	Pri + Second	-	-		-			N/A	N/A	0	0	N/A
Outer Top Cover Plate	Pri Memb	-	-	-	-	0	0	N/A	N/A	0	0	22.4
	Memb + Bend	-	-	13.5	19.4	23.2	23.1	N/A	N/A	0	0	33.7
	Pri + Second	-	-	-	-			N/A	N/A	0	0	N/A
Bottom Cover Plate	Pri Memb	-	-	-		0	0	N/A	N/A	0	0	22.4
	Memb + Bend	-	-	7.6	9.9	4.9	4.9	N/A	N/A	6.5	6.5	33.7
	Pri + Second	-	-	-				N/A	N/A		9.5	N/A
Spacer Disc	Pri Memb	0	0.5	-	-	0	0	N/A	N/A	0	0	22.4
	Memb + Bend	0	0.6	-	-	0	0	N/A	N/A	0	0	N/A

*Allowable stress for Service Level C

P_m larger of 1.25 m or Sy = 22.4
$P_L = P_L + P_B$ = larger of 1.8 Sm or 1.5 Sy = 33.7

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.

DSC thermal accident cases

NUTECH indicates in TR Table 3.2-5a that thermal accident cases were considered as separate load cases and as a part of the load combinations. The NRC staff has noted in this SER that the ASME Code does not require any stress evaluation for thermally induced stresses for Service Levels C and D. Since NUTECH categorized the two thermal accident cases as Service Levels C and D, they were not obliged to evaluate them. However, the material properties for load conditions, such as the case of the pressure stress at the higher DSC temperature, as described in the preceding paragraph, should have been taken at the higher temperature. The NRC staff did this in all tables in this SER. The results are satisfactory.

DSC load combinations for Service Level C accident conditions

Table 3.6 shows the results of seven load combinations. Load combinations, as defined in Table 3.2-5a of the TR, are a bit misleading because case C4 and C5 are actually the same, as well as cases C6 and C7. The only difference in both of these sets of cases is thermal stresses, which NUTECH did not evaluate. As may be seen from Table 3.6, all stresses are below allowable levels.

Discussion of cask drop

Because the cask drop accidents postulated by NUTECH cause the highest stresses in 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. It should be pointed out that the drop situations that were postulated by NUTECH all involve dropping the TC with the DSC inside at a maximum height of 80 inches. The NRC staff considers these assumptions reasonable, because the DSC will always be in the TC or inside the HSM whenever it is outside of the building which houses the spent fuel pool. The requirements of 10 CFR 72 must be met whenever the irradiated fuel is outside of the spent fuel pool building. Inside the building, 10 CFR 50 governs. The centerline of the HSM is located at 102 inches above the base pad and therefore the maximum drop height would be about 60 inches for the DSC, should it fall off of the transport trailer during loading or during

Table 3.6

DSC LOAD COMBINATIONS FOR ACCIDENT
Service Level C Cases[1]

DSC Component	Stress Type	Case[2] C1 NUTECH	Case[2] C1 NRC	Case C2 NUTECH	Case C2 NRC	Case C3 NUTECH	Case C3 NRC	Case[3] C4/C5 NUTECH	Case[3] C4/C5 NRC	Case[4] C6/C7 NUTECH	Case[4] C6/C7 NRC	Allowable
		Stress (ksi)										
DSC	Pri. Memb	2.7	2.0	1.8	1.4	2.9	3.0	2.7	2.8	3.9	4.0	22.4
	Memb + Bend	31.2	30.3	3.7	13.4	12.0	17.8	10.2	16.0	17.2	21.9	33.7
Inner Top Cover Plate	Pri. Memb	0.1	0.7	0.1	0.7	0.2	0.8	0.1	0.7	0.1	0.7	22.4
	Memb + Bend	23.7	23.6	5.1	5.1	24.0	23.9	23.7	23.6	23.7	23.6	33.7
Outer Top Cover Plate	Pri. Memb	0.1	0.2	0.1	0.2	0.2	0.3	0.1	0.2	0.1	0.2	22.4
	Memb + Bend	23.6	23.5	18.1	19.8	23.9	23.8	23.6	23.5	23.6	23.5	33.7
Bottom Cover Plate	Pri. Memb	0.1	1.2	0.1	1.2	0.8	1.9	0.1	1.2	0.1	1.2	22.4
	Memb + Bend	5.2	5.2	8.9	11.2	6.8	6.8	5.2	5.2	11.7	11.7	33.7
Spacer Disc	Pri. Memb	0.5	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	22.4
	Memb + Bend	0.3	1.2	0.3	0.6	0.3	0.6	0.3	0.6	0.3	0.6	33.7

1. Secondary stresses are not required for Service Level C.
2. Seismic stresses are considered "mechanical loads" and must be combined with DW and accident pressure for C1.
3. Thermal stresses are secondary and need not be evaluated for Service Level C. Therefore, both C4 and C5 are identical cases.
4. Because thermal stresses need not be evaluated for Service Level C, cases C6 and C7 are identical.

transport between the spent fuel pool building and the ISFSI site. Thus, 80 inches is conservative.

Five different drop orientations were considered: (1) a horizontal drop, (2 and 3) a vertical end drop onto the top or bottom of the TC, and (4 and 5) a corner drop at an angle of 30° onto the top or bottom corner of the TC. The drop height was 80 inches for all orientations.

The magnitude of the deceleration for each case was defined in Section 3 of the TR as 75 g for either vertical or horizontal drop orientations and 25 g for the corner drop. NUTECH based these values on an EPRI report (Reference 28), which described a method of predicting maximum decelerations of casks as a function of drop height, target hardness (i.e., hardness of concrete pad), and cask orientation.

Because Reference 28 does not document the deceleration time history, it was necessary for the NRC staff to establish what the representative time histories and damping coefficients for the three orientations would be, in order to predict appropriate dynamic load factors (DLF). NUTECH provided additional material which included references to drop test data for a 90 ton rail cask (Reference 29). The time histories from this reference were used to determine the DLFs for the different drop orientations. As discussed in Section 2 of this SER, the DLFs are also dependent on structural damping. 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 NUTECH. The NRC staff concluded that the DLFs for the vertical, horizontal, and corner drops are 1.50, 1.75, and 1.25, respectively. These factors, when multiplied times the unfactored decelerations levels, produced values of 73.5 g, 66.5 g and 25.0 g for the three drop orientations. These values compare favorably with the deceleration values of 75 g, 75 g, and 25 g selected by NUTECH in their design criteria. Based on the above review, the NRC staff finds that the deceleration levels used by NUTECH are appropriate for the drop cases considered.

The deceleration levels specified by NUTECH provide a margin of safety for ensuring the fuel integrity against the effects of impact according to Reference 30. The reference indicates that, for the type of fuel which the NUHOMS-24P system was designed around, there is ample safety margin to meet the requirements of 10 CFR 72.122(h). The B&W 15 x 15 fuel assemblies should not fail if the dynamic impact loads are below 147 g for end drops and 101 g for horizontal drops. As can be seen from the preceding paragraphs, the NUTECH loadings are substantially below these levels.

In all cases, NUTECH used the ANSYS finite element code to model the drop cases for the DSC and TC cask components. For the vertical drop, an axisymetric load and an axisymetric geometry were modeled, using an equivalent 75 g static load. For the horizontal and corner drop cases, NUTECH modeled an axisymetric structure with non-axisymetric loading. 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.

The distribution of impact force for horizontal and corner drop cases was approximated by cosine functions, which in turn were approximated by the Fourier series. NUTECH calculated the depth of concrete penetration by the dropped cask using the modified Petry formula (Reference 31), which predicted a smaller penetration depth than Reference 29. Had NUTECH used the deeper crush depth as predicted by Reference 29, the impact force would have been distributed over a larger area of the TC than NUTECH actually used to develop the Fourier series coefficients. Hence the calculations provided in the appendices of the TR are based on conservative assumptions. The computer analyses use these assumptions.

The finite element analysis calculations which NUTECH made modeled the DSC inside of the cask for the end and corner drops. See Figures 8.2-6 and 8.2-7 of the TR. Note that the DSC shell and upper and lower cover plates as well as the cask top cover plate and cask bottom cover plate were all modeled using one element through the thickness. Consequently, as described in Section 3.3.4.2.1 of this SER, the computer code only calculated membrane stresses and did not calculate any bending stresses except in the structural shell of the TC and at the outside diameter of the top and bottom cover plates of the TC. NUTECH "estimated" bending stresses by referring the membrane stresses calculated at the centroid of the element to an outer face and/or node of the element.

The results of these analyses are reported in Table 8.2-7 of the TR. The NRC staff has summarized the results and included the findings of the staff review in Table 3.7. Discrepancies between the TR results and staff results can be attributed to two principal causes. Some results reported in the TR have been superseded by additional calculations that NUTECH performed following submittal of the Revision 1 of the TR and consequently are not shown in the TR. Another source of discrepancy is due to the difference in thickness of the DSC cover plates as calculated and as specified on the drawings. The staff increased the stresses listed in the computer output listing by a ratio of the squares of the end plate thicknesses. The results in Table 3.7 show that the stresses for all components for all drop orientations are lower than the ASME Code allowable stresses for Service Level D. The NRC staff evaluated the material properties for the worst case temperature reported by NUTECH, i.e., load case D1. Consequently, the allowable stresses are slightly lower than NUTECH used. In all cases, the calculated stresses are lower than the allowable stresses.

Two analyses were carried out to verify the design adequacy of the spacer disc. A finite element analysis of one half of a spacer disc, symmetrically loaded with 75 times the vertical static load, was performed. Also, a stress and buckling stability analysis of the entire disc was performed using ANSYS. This analysis assumed the disc was supported in-plane around the circumference of the disc and out-of-plane at the four support rod locations. Again, the load consisted of 75 times the dead weight. As Table 3.7 of this SER shows, the spacer disc satisfies the ASME Code allowable stresses for both vertical and horizontal drop orientations. The NRC staff verified that the eigenvalue buckling solution is 1.8 times the load for horizontal load cases. No buckling analysis was performed for the vertical drop case.

The guide sleeves were checked for bending plus membrane stress when loaded horizontally by 75 times self weight and simply supported between spacer discs. The resulting stress intensity was only 2 ksi, far below the allowable of 63.5 ksi.

The four support rods running the length of the basket were also checked for stress as well as critical buckling during a vertical drop. For the drop accident, NUTECH postulated the load for each rod to be one quarter

Table 3.7

DSC DROP ACCIDENT LOADS
Service Level D

DSC Component	Stress Type	Stress (ksi) Vertical NUTECH	Vertical NRC	Horizontal NUTECH	Horizontal NRC	Corner NUTECH	Corner NRC	Allowable*
DSC Shell	Pri. Memb.	6.2	33.2	9.2	17.6	12.7	18.2	43.4
	Memb + Bend	19.3	29.2	12.4	24.7	28.6	28.8	63.5
Inner Top Cover	Pri. Memb.	0	0	14.	16.9	17.6	17.6	43.4
	Memb + Bend	33.8	34.0	15.9	18.7	8.2	12.0	63.5
Outer Top Cover	Pri. Memb.	0	0	9.5	9.5	10.2	10.2	43.4
	Memb + Bend	20.1	29.0	14.6	21.0	6.2	14.7	63.5
Bottom Cover	Pri. Memb.	0	0	9.5	9.5	34.7	39.4	43.4
	Memb + Bend	19.1	30.5	14.6	21.0	23.2	43.8	63.5
Spacer Disc	Pri. Memb.	1.	24.4	36.4	37.3	-	-	43.4
	Memb + Bend	22.4	27.9	26.1	47.6	-	-	63.5
Support Rods	Primary	32.6	32.6			32.6	32.6	49.9
Top End Struct. Weld	Primary (shear)	5.	5.	9.5	9.5	9.5	9.5	43.4
Bottom End Struct. Weld	Primary (shear)	4.1	4.1	9.5	9.5	9.5	9.5	43.4

* Allowables taken at worst case temperature, i.e., for Case D1, T=513°F shell temperature.

of: the dead weight of eight spacer discs, the weight of the guide sleeves, and the self weight of one rod. The primary axial stress was only 32.6 ksi compared with an allowable of 49.9 ksi. Also, the critical buckling load was found to be 180 ksi, well above the actual load. Based on the above evaluation, the NRC staff concurs that the support rod design is satisfactory.

<u>DSC load combinations</u>

Table 8.2-9b of the TR summarized the enveloping load combination stress results for the DSC accident conditions. Table 3.2-5a of the TR defined the load cases for each load combination. The stress intensities in the DSC at various critical locations were evaluated by combining the dead load, accident pressure load, and the worst drop orientation load. Table 3.8 of this SER uses material allowables for Service Level D for the worst thermal condition reported in the TR. These allowables are somewhat lower than the TR used; however, it may be seen that even with these lower allowable stresses, the DSC components meet the ASME Code requirements.

It should be noted that NUTECH elected to use Service Level D for accident case allowable stresses. While the NRC staff concurs with this decision, it must be coupled with the NUTECH operating controls and limits as proposed in Section 10 of the TR. Following a cask drop of fifteen inches or greater, the DSC must be retrieved, and the DSC and the internals must be inspected for damage. The 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.

<u>DSC fatigue evaluation</u>

Section NB-3222.4a of Section III of the ASME Code (Reference 12) 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, NUTECH evaluated each of the six conditions defined by the ASME Code. The NRC staff

Table 3.8

DSC ENVELOPING LOAD COMBINATION
RESULTS FOR ACCIDENT LOADS
Service Level D

DSC Component	Stress Type	Controlling Load Combination	Stress (ksi) Calculated NUTECH	NRC	Allowable*
DSC Shell	Pri. Memb. Memb + Bend	D2	11.9 25.9	35.0 49.8	43.4 63.5
Inner Top Cover	Pri. Memb. Memb + Bend	D2	14.0 57.5	18.3 57.6	43.4 63.5
Outer Top Cover	Pri. Memb. Memb + Bend	D2	9.5 43.5	10.4 52.5	43.4 63.5
Bottom Plate	Pri. Memb. Memb + Bend	D2	9.5 28.4	40.6 49.0	43.4 63.5
Spacer Disc	Pri. Memb. Memb + Bend	D2	36.4 26.4	37.8 48.2	43.4 63.5
Guide Sleeve	Memb + Bend	D2	2.	2.0	63.5
Support Rods	Pri. Memb.	D2	32.6	32.6	49.9
Top End Structural Weld	Primary (shear)	D2	15.5	15.5	43.4
Bottom End Struc. Weld	Primary (shear)	D2	12.0	12.0	43.4

* Allowables taken at worst case temperature, i.e., Case D1, T=513^{o}F shell temperature.

evaluated NUTECH's analysis and concurs with the finding that the service loading of the DSC meets all conditions, and therefore does not require a separate analysis for cyclic service.

3.3.4.3 DSC Support Assembly Analysis

A linear elastic structural analysis was performed using the STRUDL finite element computer program to determine the deflections, forces and stresses under normal, off-normal and accident loading conditions. Three load combinations were performed to determine the worst resultant stresses and the end forces. The boundary conditions used in the finite element mathematical model do not reflect the boundary condition shown on the drawings in the TR. The drawings indicate that the ends of the T-section guide rails are welded to the access opening sleeve, whereas analytically they were modeled as free ends instead of fixed ends. The staff discussed this discrepancy with the vendor's contractor and concluded that this is a conservative approach for the loading condition listed above. Therefore, the staff accepts the finite element mathematical modeling technique for the analysis of the support assembly.

3.3.4.3.1 DSC Support Assembly Normal Operating Condition

The normal operating condition loads consist of the dead weight of the support assembly, the dead weight of the DSC, the DSC operational handling loads and temperature loads.

DSC support assembly dead weight analysis

The staff checked and concurs with the results due to the dead weight of the support assembly and the weight of the DSC from the STRUDL computer output. The worst stresses are listed in Table 8.1-8 of the TR, forces at the end connections are listed in Table 8.1-9, and the maximum vertical deflection is listed in TR Table 8.1-9a. The tabulated values of deflection and stresses meet the AISC allowable limits for normal conditions.

DSC support assembly operational handling analysis

The normal operating handling load considered was a 20,000 pound load applied axially to the rails. This models the normal condition of loading the DSC into the HSM with a coefficient of friction of 0.25. The staff checked the results from the STRUDL computer output. The worst stresses are listed in Table 8.1-8 of the TR. Forces at the end connections are listed in Table 8.1-9, the maximum vertical deflection was listed in TR Table 8.1-9a. The tabulated values of deflection and stresses meet the AISC allowable limits for normal conditions.

DSC support assembly thermal analysis

After reviewing the drawings in the TR and discussing the analytical model with the vendor's contractor, the staff agrees that no thermal stresses will be induced into the support assembly system. To permit free thermal expansion, slotted bolt holes are used at the connections. The bolts will be installed "snug tight" in accordance with AISC requirements with lock nuts added to ensure that the bolts remain in place; then the friction in the bolted assembly can be overcome by the thermal expansion of the members during normal heatup conditions.

3.3.4.3.2 DSC Support Assembly Off-Normal Event

Section 8.2 of the TR discusses off-normal events as they relate to the support assembly. The off-normal event conservatively considered was a jammed condition where the hydraulic ram exerted a force equal to the weight of DSC and was applied axially to the rails of the support assembly. Results were combined with the results due to the dead weight of the support assembly. The combination from the STRUDL output was checked by the staff. The worst stresses are listed in Table 8.2-11 of the TR and the maximum end loads are listed in Table 8.2-12. The tabulated stress values meet the AISC normal allowable limits. The staff checked the tabulated results against the computer output and concurs with the results as shown in TR Table 8.2-11. However, the qualification of the weld joints between the rails and the embedded access opening sleeve was not documented in the TR. The staff performed independent calculations for these weld joints and found them to be acceptable.

3.3.4.3.3 DSC Support Assembly Accident Analysis

The only loading that the DSC support assembly experiences during an accident analysis is the loading combination associated with a seismic event. Hand calculations were performed to determine the lowest frequency of the support structure. The staff review concurs with the frequency as calculated by NUTECH's contractor. The corresponding values of accelerations at 18.3 Hz are 0.40 g in the vertical and 0.60 g in the horizontal directions; however, 0.48 g acceleration was used in both horizontal orthogonal directions for the finite element computer analysis. The staff reviewed the load combinations from the computer input list, and found none of them reflect the accident load combination as shown in Table 8.2-11 of the TR. The tabulated stresses were taken from load combination 21 of the computer runs which is the combination of the load cases 1, 3, 7, 27, 28. They are: the dead weight of the support assembly, the normal axial handling load, and the transverse seismic (vertical Y) added to the X and directions for seismic (horizontal). The staff has not confirmed that this produces the worst load combination. However, conservatisms built into the model, such as boundary conditions of the structure, inclusion of the normal axial handling load, simultaneous application of the seismic forces in all three orthogonal directions and summation of the results absolutely should lead to an acceptable combination. The stresses are tabulated in TR Table 8.2-11. The calculated stresses of this accident load combination are lower than the accident allowable stress of 1.5 times the normal allowable stresses at 600^{0} as shown in TR Table 8.2-11. They are also lower than the accident allowable stress of 1.33 times for normal allowable stresses as prescribed by AISC steel construction manual.

The staff compared the tabulated stress and end load combinations from TR Tables 8.2-11 and 8.2-12 against the computer output and finds them acceptable.

Qualification of the embedded support connection to the HSM for the DSC support assembly

The licensee submitted additional detail drawings of the embedded support connections, but did not submit any qualification calculations. The staff evaluated the drawings and performed some hand calculations to determine that under the reaction forces at the boundary connections of the DSC support assembly, the embedded support connections are acceptable.

DSC support assembly load combinations

Three load combinations were considered. Load combination one consists of the DSC plus the support assembly dead weight, plus the DSC handling loads for a typical normal operating load case. Load combination two includes the dead weight of the support structure plus DSC handling loads in the jammed condition, representing an off-normal loading. The third load combination includes the total dead weight plus design basis seismic loads for an accident event. These results were compared to AISC code allowables and they are within the allowable limits.

DSC seismic restraint analysis

The DSC seismic restraint is located inside the HSM access opening. The restraint and its attachment were designed for a lateral force equal to the mass of the DSC times the horizontal acceleration times an impact factor of 1.5. The staff performed an independent review including hand calculations and concludes that the DSC seismic restraint is acceptable under the loading condition described above.

3.3.4.4 Transfer Cask

3.3.4.4.1 TC Normal Operating Conditions

The transfer cask was evaluated for the three normal operating conditions of: (1) dead weight load, (2) thermal loads, and (3) normal operation handling loads. Table 3.9 summarizes all the stress analysis results for the normal operating conditions. The summary table shows stresses for each of the TC components for each of the three loads as

Table 3.9

TRANSFER CASK STRESS ANALYSIS RESULTS FOR NORMAL LOADS
Service Levels A and B Allowables

Cask Component	Stress Type	Stress (ksi) Dead Weight		Thermal**		Normal Handling		*Allowable
		NUTECH	NRC	NUTECH	NRC	NUTECH	NRC	
Cask Shell	Pri Memb	0.7	0.7	NA	NA	0.5	-	21.7
	Memb + Bend	0.8	0.5	NA	NA	30.3	-	32.6
	Pri + Sec.	0.4		20.3	12.2	35.6	48.4	65.1
Top Cover Plate	Pri Memb	0.2	-	NA	NA	-	-	21.7
	Memb + Bend	0.6	-	NA	NA	6.3	-	32.6
	Pri + Sec.	0.5	-	7.4	9.2	-	-	65.1
Bottom Cover Plate	Pri Memb	0.2	-	NA	NA	-	-	18.7
	Memb + Bend	1.3	-	NA	NA	14.2	-	28.
	Pri + Sec.	1.4	-	5.3	-	-	-	56.1
Top Ring	Pri Memb	.2	-	NA	NA	-	-	20.3
	Memb + Bend	.1	-	NA	NA	-	-	30.5
	Pri + Sec.	.5	-	4.5	-	-	-	60.9
Bottom Ring	Pri Memb	.4	-	NA	NA	-	-	20.3
	Memb + Bend	.3	-	NA	NA	-	-	30.5
	Pri + Sec.	.6	-	14.9	-	-	-	60.9

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

analyzed by NUTECH. The NRC staff verified selected components and has recorded them adjacent to the NUTECH stress levels. The ASME Code allowable stresses for the various materials were taken at 400°F for Service Levels A and B. All calculated stresses are below allowable levels.

Dead weight loads for the TC

The dead weight loads were evaluated for the TC in a vertical orientation, suspended from the lifting trunnions, as well as a horizontal orientation supported by the pillow blocks of the TC support skid. See Figure 1.3-4 of the TR for a sketch of the skid. All stress levels are one to two orders of magnitude lower than allowables.

Thermal loads for the TC

Section 8.1.1.9.C of the TR describes the thermal analysis performed by NUTECH to verify that the thermal stresses in the TC are below allowable stresses for Service Levels A and B. These service levels are the only ones that NUTECH was required to evaluate according to the ASME Code for Class 2 components (Reference 17). Table 3.2-5b of the TR defines the temperature at which specific load cases were evaluated, i.e., an ambient temperature of 100°F for normal conditions, and an ambient temperature of 125°F for off-normal conditions. The NRC staff reviewed the computer analyses for the thermal case and confirmed that only one run was made for the 125°F case, although it was not possible to confirm that the temperature distribution for the model was correct. No information was provided for a temperature distribution. Also the NRC staff noted that the thermal stresses reported by NUTECH in TR Tables 8.1-10a and 8.1-10b show identical thermal stresses for normal and off-normal. The staff checked the computer output and recorded the stresses as shown in the summary table of this SER. In all cases the staff confirmed that the thermal stresses are below allowables.

Operational handling loads for TC

As described in the dead weight load section above, there are two normal operating 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 18) and the latter is governed by the ASME Code.

The ANSI code is concerned with critical loads and consequently only addresses the lifting trunnion design and the TC shell in the vicinity of the lifting trunnion. Table 3.9 of this SER summarizes the results of stress analysis for the TC shell and top and bottom cover plates. All results for the normal handling case are satisfactory for Service Level A.

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 18) and NUREG-0612 (Reference 32). Critical loads, as defined by N14.6, are defined as 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 and tilting trunnions shall be considered as a special lifting device for the DSC. Because its design does not provide a dual-load path, the design criteria requires 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 32 does not quantify the magnitude of this dynamic load, ANSI NOG-1-1983 (Reference 33) does specify 15%, which was used by NUTECH. Therefore the NUTECH assumption is appropriate.

Table 3.10 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 Table support loads. The local stresses in the TC at the intersection of the trunnion sleeve and the shell stiffener insert (see Figure C.1-1 of the TR) are calculated by using the method of the Welding Research Council, WRC-297 (Reference 34). The local stresses at the intersection of the shell and the shell stiffener insert are also evaluated by the same method. Summary Table 3.10 shows that all stresses are less than the allowables for both the ANSI-

Table 3.10

SUMMARY OF STRESS ANALYSIS FOR LIFTING TRUNNION ASSEMBLIES, WELD REGIONS AND CASK SHELL FOR LIFTING CASES

	Stress (ksi)											
	Critical Handling Loads per ANSI N14.6						On-site Transfer per ASME III Class 2					
Component Location	Stress Intensity			Allowable			Stress intensity			Allowable*		
Trunnion Lift Pin	5.9			13.5			NA			NA		
Trunnion Rest Pin	6.3			13.5			3.6			43.8		
1.5" Sleeve @ Insert	5.7			9.0			4.1			32.6		
Weld @ Rest Pin/ Sleeve	Plane 1	Plane 2		Plane 1	Plane 2		Plane 1	Plane 2		Plane 1	Plane 2	
	5.0	7.0		9.0	8.0		5.1	6.8		45	45	
Weld @ Sleeve/ Insert	Pl 1	Pl 2	Pl 3	Pl 1	Pl 2	Pl 3	Pl 1	Pl 2	Pl 3	Pl 1	Pl 2	Pl 3
	5.	3.8	4.	7.	9.0	5.6	5.	3.8	4.	32.6	45.	32.6
Cask Stiffener Plate ASME	Vert	Tilt	Horiz	All Cases			DL± Vert	DL± Lat	DL± Comb	All cases		
	22.6	22.8	19.1	32.6			22.3	41.0	31.3	65.1		
Cask Shell @ Stiff. Plt. ASME	Critical Lift						On-Site Trans.					
	17.4			32.6			48.4			65.1		

*Service Level A

and the ASME-governed load conditions. All stresses shown are the result of the NRC staff calculations.

Table 3.11 shows the results for the tilting trunnion assemblies. Comparisons between the NUTECH-derived and NRC staff-derived stresses show that all the stresses are lower than the allowable except for the tilting trunnion shell to sleeve intersection. The discrepancy arises due to the NRC staff using material allowables evaluated at 400°F. Table 8.2-13 of the TR also shows material allowables evaluated at 400°F. The staff considers that this temperature may be overly conservative, if Table 8.1-14 of the TR is consulted. There, the maximum exterior cask temperature noted by NUTECH was 248°F. If the NRC staff considers the maximum exterior temperature of the TC to be 300°F (still conservative), then the material allowable would be 67.5 ksi. With this adjustment in allowable stress, all calculated stresses are below the allowables.

3.3.4.4.2 TC Off-Normal Operating Conditions

The only off-normal operating condition considered by NUTECH was for an ambient temperature of 125°F. Since NUTECH reported the same stresses in Tables 8.1-10a and 8.1-10b of the TR, and actually only evaluated thermal stresses for 125°F, the results are the same. Table 3.9 of this SER shows these results. They are all satisfactory.

TC load combinations for normal and off-normal conditions

Table 3.2-5b of the TR defines the different load combinations for normal and off-normal events. These conditions correspond to Service Levels A and B of the ASME Code. Altogether there are five Level A conditions and two Level B conditions; however, NUTECH does not present data for all the cases, so Table 3.12 of this SER has combined the conditions as follows. NUTECH only evaluated the thermal case for an ambient temperature of 125°F. Consequently there is no difference between their load cases A4 and B1, and similarly for A5 and B2. In all cases the allowable stresses were evaluated for a material temperature of 400°F, a conservative value. As shown in Table 3.12, all the stresses are lower than the allowables.

Table 3.11

SUMMARY OF STRESS ANALYSIS FOR TILTING
TRUNNION ASSEMBLIES, WELD REGIONS AND CASK SHELL

For On-Site Transportation Cases
Per ASME III Class 2

Component, Location	Stress Intensity (ksi) NUTECH	Stress Intensity (ksi) NRC	Allowable (ksi)*
Trunnion/Sleeve Intersection	5.6	9.	18.7
Sleeve/Shell Intersection	9.3	8.4	21.7
Trunnion/Sleeve Weld	12.6	12.4	18.7
Sleeve/Shell Weld	9.5	7.2	21.7
Shell/Sleeve Intersection	67	65.4	65.1 67.5**
Shell membrane stress	5.2	5.2	28.4

* Allowable stresses taken at $400^{o}F$
** Allowable stresses taken at $300^{o}F$

Table 3.12

TRANSFER CASK LOAD COMBINATIONS
FOR NORMAL OPERATING CONDITIONS
Service Levels A and B

Cask Component	Stress Type	Load Combination	Calculated Stress (ksi) NUTECH	Calculated Stress (ksi) NRC	Allowable Stress (ksi)
Cask	Pri Memb	A4/B1	1.2	1.2	21.7
Shell	Memb + Bend	A4/B1	31.1	30.8	32.6
	Pri + Sec.	A4/B1	56.3	61.0	65.1
Top	Pri Memb	A2/B5	0.2	0.2	21.7
Cover	Memb + Bend	A4/B1	6.9	6.9	32.6
Plate	Pri + Sec.	A4/B1	7.9	9.7	65.1
Bottom	Pri Memb	A1	0.2	0.2	18.7
Cover	Memb + Bend	A4/B1	15.5	15.5	28.
Plate	Pri + Sec.	A4/B1	6.7	6.7	56.1
Top	Pri Memb	A1	0.2	0.2	20.3
Ring	Memb + Bend	A3	0.1	0.1	30.5
	Pri + Sec.	A1	5.0	5.0	60.9
Bottom	Pri Memb	A3	0.4	0.4	20.3
Ring	Memb + Bend	A3	0.3	0.3	30.5
	Pri + Sec.	A3	15.5	15.5	60.9

No load combinations for either case B1 or B2 were presented by NUTECH. The TR distinguished case B1 from A4 and case B2 from A5 by indicating a higher ambient temperature (125°F). However, NUTECH only calculated stresses associated with the 125°F temperature.

3.3.4.4.3 TC Accident Conditions

Section 8.2 of the TR defines the accident conditions that affect the transfer cask. These conditions are: (1) earthquake, and (2) accidental drop of the TC with the DSC inside. NUTECH also considered a third case, as defined on page 8.2-6 and 8.2-7 of the TR; however, this case was not incorporated in TR Table 3.2-5b, and the results were never incorporated into the enveloping load combination Table 8.2-14 in the TR.

The unincorporated case is for design basis winds. NUTECH postulated a pressure of 595 pounds per square foot (psf) pressure acting on the surface of the TC when supported by the transport trailer. This was based on a maximum wind pressure of 397 psf. NUTECH showed that if the height to the top of the cask is 146 inches, and the track of the transport vehicle is 132 inches, there is a safety factor of 1.5 against overturning. Shell stresses were also evaluated and found to be 3.8 ksi, well below the 26 ksi allowable for Service Level C. The 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 TR.

<u>TC seismic condition</u>

NUTECH evaluated 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 NUTECH showed that the loaded TC would not overturn during an earthquake, provided the loaded TC weighed 190 kips and experienced a horizontal acceleration of 0.4g. Since the seismic criteria calls for 0.25 g at ground level, even when both orthogonal directions are summed by the SRSS and the resultant 0.35 g is used to calculate the stability, the NRC staff calculated a safety factor of 1.18 against overturning.

The second case postulated by NUTECH was for a seismic event occurring during the normal transport of the TC loaded on the trailer. NUTECH stated that this case is enveloped by the handling case of $\pm$0.5g acting in the vertical, axial and transverse directions simultaneously. On page 8.2-21 of the TR, the statement is made that the calculated stress intensities for normal transport case are 17.9 ksi for the cask structural shell and 2.0 ksi for the trunnions, and furthermore that these were "conservatively used as the maximum seismic stresses in the load combination results" in Tables 8.2-13 and 8.2-14 of the TR. These tables do not reflect this statement, since Table 8.2-14 shows a shell stress of 31.1 ksi for load combination C1 which is dead weight loads, transportation handling loads, and seismic. If NUTECH had used 17.9 ksi for seismic as well as handling, they would have recorded at least 35.8 ksi. The staff evaluated this load condition and arrived at 31.2 ksi. This stress is lower than the allowable, as are the other stresses for the seismic case as shown in Table 3.13 of this SER.

TC load combinations for Service Level C accident conditions

Table 3.13 of this SER shows the results of two load combinations, as defined in Table 3.2-5b of the TR. The only difference between cases C1 and C2 is in the calculation of handling loads, i.e., during actual transport with ± 0.5 g acting in all three directions and during the transfer of the DSC into or out of the HSM. NUTECH does not present any results for the latter case, but it is clear that the higher loading case occurs during actual over-the-road transport between the spent fuel pool and the HSM pad. The NRC staff also included the stress intensities resulting from the DBT winds in this single load combination. As may be seen from Table 3.13, all the stresses are below the allowable levels for Service Level C conditions.

Cask drop accident

Section 3.3.4.2.3 (DSC Accident Conditions) of this SER presents a detailed discussion of the cask drop accidents postulated by NUTECH. This discussion includes the basis for the selection of the parameters and the assumptions used for the ANSYS finite element models. Because the previous discussion covered the DSC as well as the TC, it will not be repeated here.

Table 3.14 summarizes the results of all five drop orientations postulated by NUTECH. All the structural components of the TC are reported including: the cask shell, the top and bottom rings, and the top and bottom cover plates. The cask liner, bolts for the top cover plate, and the bottom sheet steel plates are also included for completeness. The temperature chosen by NUTECH to evaluate the material properties is 400°F. Because the outside surface of the TC does not exceed 248°F, this material temperature

Table 3.13

TRANSFER CASK STRESS ANALYSIS RESULTS
FOR ACCIDENT LOADS
Service Level C** Allowables

		Stress (ksi)				
Cask Component	Stress Type	Handling	Seismic	DBT Wind	Load Comb C1***	Allowables*
Cask Shell	Pri Memb	0.5	.5	3.8	5.5	26.
	Memb + Bend	30.3	0.4		31.2	39.
Top Cover Plate	Pri Memb	-	.2	0.5	.7	26.
	Memb + Bend	6.3	.3		7.2	39.
Bottom Cover Plate	Pri Memb	-	.2	0.5	.9	22.4
	Memb + Bend	14.2	.3		15.8	33.7
Top Ring	Pri Memb	-	.2	-	.4	24.3
	Memb + Bend	-	-		-	36.5
Bottom Ring	Pri Memb	-	.2	-	.6	24.3
	Memb + Bend	-	-		-	36.5

* Allowables taken at 400°F

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

*** The C1 load combination includes deadweight, seismic, handling loads, and DBT wind loads.

Table 3.14

TRANSFER CASK DROP ACCIDENT LOADS
Service Level D Allowables

Cask Component	Stress Type	Stress (ksi) Vertical Top Drop NUTECH	Vertical Top Drop NRC	Vertical Bottom Drop NUTECH	Vertical Bottom Drop NRC	Horizontal Drop With DW NUTECH	Horizontal Drop With DW NRC	Corner Top NUTECH	Corner Top NRC	Corner Bottom NUTECH	Corner Bottom NRC	Allowables*
Cask Shell	Pri. Memb	9.6	30.1	8.7	8.7	3.8	22.7	3.2	7.6	4.6	8.5	49.
	Memb + Bend	10.2	33.6	-	-	15.5	21.9	6.6	7.5	13.9	11.3	70.
Cask Liner	Pri. Memb	19.3	12.3	12.9	12.9	9.3	12.7	4.2	7.4	8.8	18.2	44.9
	Memb + Bend			-	11.4	-	17.4	7.4	5.8	25.7	28.9	64.9
Top Ring	Pri. Memb	25.2	24.2	-	-	12.2	17.3	2.1	7.5	-	-	48.7
	Memb + Bend	-	46.4	-	-	-	22.6	2.9	12.6	-	-	73.1
Top 3" Cover	Pri. Memb	24.2	20.3	-	-	5.8	7.7	2.7	11.7	-	-	49.
	Memb + Bend	-	22.5	3.7	3.7	-	8.0	14.1	14.1	-	-	70.
Bottom 2" Cover	Pri. Memb	-	-	22.9	5.8	5.8	6.4	-	-	-	33.1	44.9
	Memb + Bend	14.4	14.4	10.2	10.2	-	11.6	-	-	33.1	28.6	64.9
Bottom Ring	Pri. Memb	-	-	14.0	26.7	12.2	12.2	-	-	9.7	10.7	48.7
	Memb + Bend	-	-	-	-	-	25.9	-	-	4.6	33.9	73.1
Bottom 1/4" PL	Pri. Memb	-		11.1	11.1	5.8	6.4	-	-			44.9
	Memb + Bend	11.1	11.1	-	-	-	11.6	-	-	14.5	14.5	64.4
Bolts for Top Cover	Ave. Tension	-	-	-	-	-	-	27.1	29.7	-	-	77.0

* Allowables taken at 400^{o}F

is conservative. None of the stresses reported in the SER summary Table 3.14 exceed the allowables for Service Level D Conditions.

It is interesting to note that the ANSYS models predict that the stresses will exceed the yield stress for all major structural components except the top cover plate. Thus the previous discussion concerning the selection of a 7% critical damping value is partially justifiable, by virtue of stress levels in excess of the yield stress. (See Sections 2.4 and 3.3.4.2.3 of the SER). As discussed in the structure analysis of the DSC, any drop height higher than fifteen inches shall require the retrieval and inspection of the DSC and its internals, in keeping with the guidelines of the ASME Code when using Service Level D allowables.

In docketed responses to NRC staff's questions, NUTECH presented results of a fourth accident condition, namely design basis tornado (DBT) generated missiles. The two missiles considered are those suggested in NUREG-0800 (Reference 24), a 3967 pound automobile, and a 276 pound 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. The results are given in Table 3.15.

TC load combinations for Service Level D accident conditions

Table 8.2-9b of the TR summarizes the enveloping load combination stress results for the TC drop accident. Table 3.2-5b of the TR defines the load cases for each load combination. In Revision 1 of the TR, only three cases were postulated to envelop all Service Level D conditions. They were: (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 3.15 of this SER shows the results and the material allowables at 400°F for the various materials specified in the drawings. These allowables are in most cases somewhat higher than given in the TR, but they represent the values for the specified materials. In docketed responses to staff questions, NUTECH summarized the results of DBT winds and DBT-generated missiles. The results of these three additional accident cases are also shown in Table 3.15. The results of these cases need to be incorporated in Revision 2 of the reference TR. In all cases the actual stress intensities are lower than

Table 3.15

TRANSFER CASK LOAD COMBINATIONS
FOR ACCIDENT CONDITIONS
Service Level D

Cask Component	Stress Type	Stress (ksi) Case D1 (Vert) NUTECH	NRC	Case D2 (Corner) NUTECH	NRC	Case D3 (Horiz) NUTECH	NRC	DBT Wind* NUTECH	Massive Missile* NUTECH	Pen. Resist. Missile* NUTECH	Allowable** ksi
Cask	Pri. Memb	10.3	30.8	5.3	9.2	4.5	23.4	0.9	6.4	4.9	49.
Shell	Memb + Bend	11.0	34.1	14.7	11.8	16.3	22.4	2.9	20.5	30.3	70.
Top	Pri. Memb	25.4	24.4	2.3	7.5	12.4	17.3	NA	NA	NA	48.7
Ring	Memb + Bend	25.4	46.5	3.0	12.6	.1	22.6	NA	NA	NA	73.1
Top	Pri. Memb	24.4	20.3	2.9	11.7	6.0	7.7	0.	0.	0.	49.
Cover	Memb + Bend	24.4	22.5	14.7	14.7	.6	8.0	0.4	19.7	13.2	70.
Bottom	Pri. Memb	14.4	26.7	.4	-	12.6	12.2	NA	NA	NA	48.7
Ring	Memb + Bend	.3	26.7	.3	-	.3	25.9	NA	NA	NA	73.1
Bottom	Pri. Memb	23.1	5.8	.2	33.1	6.0	6.4	0.	0.	0.	44.9
Cover	Memb + Bend	15.7	10.2	34.4	28.6	1.3	11.6	0.3	17.5	22.2	64.4

* Data obtained from responses to NRC staff questions. This information needs to be incorporated in a Revision 2 to Reference 1.

** Service Level D Allowables

the allowables. Thus the TC meets the ASME Code for Service Level D conditions.

<u>TC fatigue evaluation</u>

Section C.4.2 of the TR presents an evaluation of the loading cycles of the TC to show that the six criteria associated with NC-3219.2 of the ASME Code are met. The NRC staff evaluated Section C.4.2 and concurs with NUTECH that all six ASME criteria are met; however, the margin is very small for the sixth criteria for mechanical loads. Using the WRC Bulletin No. 297 (Reference 34), the NRC staff calculated local stresses in the cask shell to be 48.4 ksi. These local stresses are due to normal mechanical handling loads. For the 5000 stress cycles selected by NUTECH, the allowable stress, Sa, is only 50. ksi. Thus even a small deviation in service cycles, material specification, or load could result in a situation where a licensee would be required to evaluate the cyclic operation according to Section NC-3219.2 of the ASME Code.

3.3.4.5 HSM Loading and Unloading

The actions and equipment associated with loading the DSC into the HSM are addressed in Section 5.1.1.6 of the TR. Unloading is addressed in Section 5.1.1.8 of the TR. An off-normal situation of a jammed DSC occurring during loading or unloading is addressed in Section 8.1.2.1 of the TR.

The TR states that approval of the procedure descriptions in Section 5 is not sought, that the included descriptions are for information and illustrate the feasibility and suitability of the prepared system. The TR states that actual proposed procedures would necessarily be the subject of a site-specific application. Equipment identified as required for the loading and unloading operations are: a trailer to hold and position the transfer cask which includes a skid positioning system and jacks for vertical position adjustment; a "porta-crane" to remove and/or replace the HSM door, the TC top cover plate, and the TC bottom ram access port cover; cover plate lifting cables; a cask restraint system to secure the TC to the HSM; an optical alignment system to align the TC with the HSM; a hydraulic ram system to push the DSC; and the trailer prime mover. In addition, tools and equipment would be required for removing and securing cover plate bolts/nuts and welding the HSM door in place (or cutting the welds for unloading). The TR does not include defined designs for any of the loading and unloading equipment.

The NRC staff reviewed the TR and concurs with the descriptive material of DSC loading into the HSM and unloading procedures. However, the staff considers that the design of the following have safety implications and therefore, since adequately defined designs are not included in the TR, such designs must be included in site-specific applications to use the NUTECH NUHOMS-24P system:

1. TC transfer skid and trailer, due to the potential for overturning and exceeding the limits on cask drop used in the accident condition analyses; the need to provide a stable base during DSC transfer operations; and the interfaces with the HSM, TC, ram system, and the cask restraint system.

2. Hydraulic ram system, due to the need to prevent excessive force on the DSC, provide a stable and linear motion, and interfaces with the TC, cask restraint system and/or trailer/skid and/or HSM.

3. Cask restraint system, due to the need to provide a secure mating of the TC with the HSM during DSC transfer and interfaces with the TC, HSM, trailer/skid and/or hydraulic ram system.

The staff reviewed the identification of normal, off-normal, and accident situations involving the TC to HSM and HSM to TC DSC loading and unloading operations and equipment and considers that they are adequate with regard to the DSC, TC, and HSM designs submitted in the TR. Additional off-normal and accident conditions may be appropriate to the equipment whose designs were not included (as noted above). These would include: determination of actual potential forces on the TC, DSC, and fuel rods if the actual trailer/skid design may permit a greater equivalent drop than used in the TR analyses; determination of the actual cask restraint system could produce overstresses on the attachment points on the HSM, TC, and any other connections; determination of the actual forces which might be exerted on the DSC by the hydraulic ram, as in a jammed condition or at either end of its travel; examination of the potential for failure of the ram to disengage from the DSC; and examination of the possibilities for TC movement relative to the HSM during DSC transfer.

3.3.4.6 Fuel Assemblies and Rods

10 CFR 72.130 briefly discusses the criteria for decommissioning of the ISFSI. Implicit in either decommissioning or in inspection for possible damage following a drop accident or a DSC containment leak is the ability of operators to remove the fuel assemblies from the DSC. 10 CFR 72.126 discusses the criteria for radiological protection including exposure control in Subpart (a) and effluent and direct radiation monitoring in Subpart (c) that must be followed during these operations. Normal and accident conditions are discussed below.

3.3.4.6.1 Normal Operating Conditions

Decommissioning, after completion of the storage period under normal conditions, is the only time when it would become necessary to remove the fuel assemblies from the DSC. The only possible problem that could be postulated as a result of the long-term storage in the horizontal condition is the sagging of the fuel rods due to creep, such that the fuel assemblies could not be removed from the DSC basket assembly.

An analysis of the potential creep and sag of the fuel rods was conducted. The fuel temperature decay was assumed to follow the ORIGEN-2 prediction for 10-year old fuel within the NUHOMS facility. The creep equation of M. Peehs et al. (Reference 35) was first used to determine whether creep of the fuel rods due to internal pressure could occur. The creep of the fuel rods for the total storage period was found to be less than 1%. This permitted the sag of fuel rods between grids to be calculated, since creep could be discounted. The sag was calculated using the standard beam equations for a tubular cross-section linearly loaded. The maximum sag was found to be 0.020 inch, which should not impede the removal of the fuel assemblies from the DSC.

3.3.4.6.2 Accident Conditions

Section 8.2.5.4 of the TR discusses recovery from a drop accident. Section 10.3.2.9 of the TR discusses fuel assembly retrieval and inspection following a cask accident. "Recovery" implies the removal of the spent fuel assemblies (SFA) from the DSC, i.e., it must be possible to easily extract the SFAs from the guide sleeves of the fuel basket. This is required by 10 CFR 72.122(l) and 72.126(a)(5). Both sections of the TR specify that for a cask drop of less than 15 inches, no inspection is required. However, for drop heights of 15 inches or greater, the transfer cask must be returned to the plant's fuel building where the DSC will be cut open and inspected for damage.

As noted in Section 3.3.4.6, radiological protection of the workers must be provided to assure that an aerosol of oxidized fuel particulate is not inhaled during inspection and removal operations. Fuel particulate can form if the spent fuel oxidizes at elevated temperature, due to air ingress into the DSC and the availability of failed cladding that could expose fuel to that air. The TR states that this work will be performed under the site's standard health physics guidelines for handling potentially contaminated equipment. These procedures <u>may</u> require personnel to work using respirators or supplied air. However, the staff finds that these precautions <u>must</u> be taken when the DSC is opened to protect the health of the operations personnel.

NUTECH used the finite element code, ANSYS, to predict the maximum elastic deflection of the spacer disc ligaments for the 75 g horizontal drop. They also postulated a three hinge collapse mechanism for plastic deformation. These deformations were 0.050 and 0.022 inches respectively. By summing these two deformations, an estimate of potential interference or binding between the guide sleeve and the SFA was predicted to be 0.072 inches, which is much less than the clearance available. Therefore, there should be no possibility of binding for the worst assumed case.

The NRC staff also compared NUTECH's maximum deceleration level (75 g) with a minimum predicted deceleration level required to yield B&W 15x15 fuel assemblies. Reference 27 predicts that 101 g deceleration is necessary to cause yielding of the fuel rods for a horizontal drop. The same reference also considered what vertical deceleration would be necessary to cause axial buckling. The level was 147 g, considerably higher than the 75 g level used by NUTECH in their design criteria. Thus there is considerable margin both with regard to the minimum deceleration levels required to cause yielding or buckling of the fuel rods as well as the clearance available between the fuel rods and the guide sleeve. The NRC staff concurs with NUTECH's statement that the SFAs could be extracted from the fuel basket following an accidental drop involving 75 g or less deceleration.

4.0 THERMAL EVALUATION

4.1 SUMMARY AND CONCLUSIONS

The staff has reviewed the thermal features of the NUHOMS-24P design and finds that they conform to appropriate sections of 10 CFR 72 and are acceptable.

4.2 DESCRIPTION OF REVIEW

4.2.1 Applicable Parts of 10 CFR 72

The thermal analysis was reviewed for conformance to 10 CFR 72 Subpart F. For normal and accident conditions 10 CFR 72.122(h) requires that the fuel cladding be protected against degradation and gross rupture. Sections 10 CFR 72.122(b,c) require that the system design provide protection against environmental conditions, natural phenomena and fires.

4.2.2 Review Procedure

4.2.2.1 Design Description

The NUHOMS system provides for the horizontal storage of irradiated fuel in a dry, shielded canister (DSC), which is placed in a concrete horizontal storage module (HSM). 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 DSC is constructed from stainless steel with an outside diameter of 67.25 inches, a wall thickness of 0.625 inches and a length of 186 inches. Within the DSC, there is a stainless steel basket consisting of twenty-four square cells. An intact PWR spent fuel assembly is loaded into each cell for a total of twenty four 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 DSC is removed from the DSC by natural draft convection and radiation. Air enters at the bottom of the HSM, flows around the canister and exits through the flow channels in the top shield slab. 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 NUHOMS system utilizes a transfer cask (TC), 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 TC.

4.2.2.2 Acceptance Criteria

Temperature limits for dry storage were developed by I.S. Levy, et al, in Reference 36. The NRC staff has reviewed and accepted the temperature limits developed in Reference 36. These limits are in the form of a family of generic limit curves of recommended maximum allowable initial cladding temperature as a function of cladding hoop stress. Fuel cooling time at the beginning of dry storage is a parameter. Based on the results presented in Reference 36, NUTECH derived a long term fuel cladding temperature storage limit of 340^{o}C. This limit was derived by applying the methodology of Reference 36 for a range of rod fill pressures (up to 480 psig), burnups, (up to 40,000 MWD/MTU) and ten years or less cooling time. Since it is possible to exceed the fuel performance limits of Reference 36 (while meeting the 340^{o} criteria) for higher burnup fuel, higher initial rod fill pressure fuel and/or fuel with cooling time greater than ten years, additional restrictions are required. To be stored in NUHOMS-24P dry storage a fuel assembly must have the following characteristics:

1. Maximum burnup less than 40,000 MWD/MTU
2. Maximum initial fill gas pressure less than 480 psig
3. Generated less than 660 watts of decay heat at ten years cooling time.

This limit is more conservative than the 380^{o}C maximum temperature used for the NUHOMS-07P (Reference 22) design, and is acceptable for normal operating conditions. Meeting these acceptance criteria assures that the requirements of 10 CFR 72.122(h) are satisfied.

Reference 37 establishes that no rods have failed in inert gas exposures up to 570^{o}C, and rods forced to failure required temperatures from 765 to 800^{o}C to produce ruptures. An accident temperature limit of 570^{o} is the acceptance criteria for accidents based on the above evaluation.

The thermal analysis review addresses the correctness of the reported concrete temperatures, and also the thermal input for stress analysis. Acceptability of the concrete temperatures relative to ACI-349-80 is addressed in Section 3 of this SER.

4.2.2.3 Review Method

The TR thermal analysis was reviewed for completeness, applicability of the methods used, adequacy of the key assumptions and correct application of the methods. The NUTECH thermal analysis was performed primarily with the HEATING-6 (Reference 38) 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 that 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 NUHOMS-24P design compared to the NUHOMS-07P design, 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 NUTECH analysis.

4.2.2.4 Key Design Information and Assumptions

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

1. The total heat generation rate for each fuel assembly is less than or equal to 660 Watts. This value is based on ORIGEN calculations and data published in the literature. All heat is assumed to be generated in the fuel region.

2. Each dry storage canister contains 24 intact PWR assemblies.

3. A factor of 1.08 to account for axial power peaking in the fuel during operation was assumed for thermal analysis inside of the DSC.

4.3 DISCUSSION OF RESULTS

The following discussion covers the analytical methods used by NUTECH for normal, off-normal, and accident conditions for the HSM, DSC, and TC, which were evaluated by the NRC staff. It also covers independent analyses that were performed by the staff.

4.3.1 Analytical Methods Used by NUTECH

The TR thermal analysis was done for the horizontal storage module, the dry shielded canister in the horizontal storage module and the dry shielded canister in the transfer cask. The HEATING-6 computer program was used to perform the major portion of the thermal analysis. HEATING-6 solves steady state and/or transient heat conduction problems in one, two or three dimensional Cartesian or cylindrical coordinates.

Air temperatures within the HSM were first established by a natural circulation cooling analysis. Steady-state circulation flow will occur when the buoyancy forces are balanced by friction and form loss forces. Flow areas and loss factors were designed to allow sufficient flow to maintain the desired temperature difference between the inlet and outlet air temperature. An independent analysis, including determination of friction and form losses, was performed by the staff to confirm the NUTECH results.

Thermal analysis of the HSM is performed to obtain the temperature of the outside surface of the DSC and the temperature distribution of the concrete module, given a heat flux across the canister surface corresponding to the spent fuel heat generation rate. Once this temperature is established, detailed analysis of the temperature distribution within the canister is done. A thermal analysis of the canister within a hypothetical TC is done to determine the peak fuel clad temperatures during normal and off-normal situations. The vacuum drying operation and loss of liquid shielding accident are also analyzed.

A two dimensional Cartesian model is used to represent the HSM for HEATING-6 analysis. The HSM is assumed to be infinitely long with the axial average heat flux determined over the DSC length. Only one-half of the module is modeled in HEATING-6, since symmetry exists about the vertical centerline. Both a single free standing unit and a 2 x 10 array of HSMs were considered. The model includes the 3 feet thick concrete ceiling, concrete side walls and the floor. The external surfaces of the side walls are assumed to be adiabatic for interior walls centered in a group of modules, or to be exposed to ambient conditions for exterior walls or for modules with no DSC stored in adjacent locations. The floor was taken as seven feet of concrete with a constant temperature at the bottom.

The DSC located within the module is modeled as a cylindrical shell represented as a series of 40 small rectangular slabs. The total surface area of these slabs is equal to the surface area of the canister. Heat transfer by convection and radiation is considered in the air gap between the canister and the interior surface of the module. Convection heat transfer at the outer surface of the module ceiling is included, as is solar heat loading on the outer surface of the module ceiling. The heat source consists of 24 PWR assemblies, each with an assumed heat generation rate of 660 Watts.

Temperatures within the DSC are determined using a second HEATING-6 model. A two dimensional Cartesian model is used to represent the DSC and the internal helium, stainless steel sleeves and fuel regions. The surface heat flux is based on the 144 inch active fuel length and a 1.08 axial peaking factor. All of the heat is conservatively assumed to be generated in the fuel regions. The regions representing the DSC wall are at fixed temperatures determined from the HSM HEATING-6 analysis. An effective thermal conductivity was determined for the fuel regions based on experimental results at E-MAD (Reference 39). These results were shown to be in agreement with the Wooten-Epstein correlation which has been previously used for TC thermal analysis.

Temperature profiles for the DSC within the transfer cask were determined from the steady-state heat conduction solution for a composite cylinder with combined radiation and convection heat transfer at the outer surface of the TC. Radiation, conduction, and convection were modeled in the air gap between the DSC and the TC.

4.3.2 HSM and Internals

4.3.2.1 Normal Operating Conditions

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 0°F representing "minimum normal conditions," (2) entering air at 70°F representing "normal conditions," and (3) entering air at 100°F representing "maximum normal conditions." The method of calculating concrete temperatures is conservative and acceptable. Satisfaction of the limiting condition for operation of a 60°F maximum air temperature rise on exit from the HSM gives a reasonable degree of assurance that adequate cooling is achieved.

Temperature gradients through the walls and roof were determined from the HEATING-6 results. These are acceptable temperature gradients for use in the reinforced concrete structural analysis.

4.3.2.2 Off-Normal Conditions

The off-normal conditions considered were an inlet temperature of -40°F representing extreme winter minimum and 125°F representing extreme summer maximum. The concrete temperature on the inside surface of the HSM reaches a maximum of 215°F for the extreme condition of 125°F ambient temperature. The results are acceptable for use in structural and concrete integrity evaluations.

4.3.2.3 Accident Conditions

The total blockage of all air inlets and exits was analyzed as the accident case. Adiabatic heatup of the various components was assumed, with the HSM providing the slowest heatup rate. Adiabatic heating starting at the 125°F inlet temperature condition is the limiting case for maximum concrete and fuel clad temperatures. The resulting concrete temperatures are reasonable and acceptable for use in the thermal loads analysis. Since it is assumed that the blockage will be cleared within 48 hours, heatup was calculated over this period.

4.3.3 DSC and Internals

4.3.3.1 Normal Operating Conditions

The normal operating conditions at 70°F and 100°F ambient air inlet temperature were analyzed for the DSC and internals. HEATING-6 input and output for the 70°F case and the corresponding HSM run were reviewed. No errors were detected. Trends and magnitude of the resulting temperature distributions are reasonable. Maximum fuel cladding temperatures were calculated and found to be less than the 340°C limit for the 70°F ambient temperature case. Maximum fuel cladding temperature was 349°C for the 100°F "maximum normal condition."

4.3.3.2 Off-Normal Conditions

The off-normal condition considered was the 125°F ambient inlet air temperature. HEATING-6 calculations were performed which yielded a maximum fuel clad temperature of 353°C compared to the acceptance criterion of

570°C. Results from this case were conservatively used to determine thermal loadings for purposes of the staff review.

4.3.3.3 Accident Conditions

Temperature distribution within the DSC was determined for the case of all air inlets and exits blocked for a 48-hour 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. Maximum fuel cladding temperature was calculated to be 403°C, which is below the 570°C accident limit.

4.3.4 Transfer Cask and Fuel

During loading, evacuation and transport to the HSM, the DSC is located within the TC. In this case, the inside surface temperature of the transfer cask was determined by calculating the steady-state temperature distribution through the cask which was modeled as a series of cylindrical annular regions. The surface temperature of the DSC was then determined from the conduction, convection and radiation heat losses from the canister to the cask. Two cases were considered: the top half of the DSC which is not in contact with the TC, and the bottom half which is assumed to contact the TC over its entire surface. This models the situation where the DSC and TC are in the horizontal position for transport.

Two normal, two off-normal and one accident conditions were analyzed. Normal minimum and maximum ambient air temperatures of 0°F and 100°F were analyzed, along with -40°F minimum and +125°F maximum ambient air temperature off-normal cases. The accident condition analyzed was the loss of liquid neutron shield. The vacuum drying operation with the evacuated DSC cavity was also analyzed since this is a limiting condition. Maximum fuel clad temperature was 421°C for the loss of liquid neutron shield accident, which is within the acceptance criteria of 570°C. For the evacuated DSC, the maximum fuel cladding temperature was 410°C. When the DSC is not evacuated, the maximum temperature will be significantly lower due to the higher effective thermal conductivity within the DSC. Since the evacuated condition is short term, the acceptance criterion of 570°C is satisfied.

5.0 CONFINEMENT BARRIERS AND SYSTEMS

5.1 SUMMARY AND CONCLUSIONS

The staff has reviewed the features of the NUHOMS-24P design which provide confinement of radioactive material and, specifically, protection of the fuel rod cladding. The review was directed at two aspects of the design: (1) the mechanical integrity of the DSC and (2) the long term behavior of cladding in an inert helium atmosphere.

As a result of this review, the staff concludes that the NUHOMS design conforms to applicable parts of 10 CFR 72.122(h). Confinement is assured by a radiographic inspection of the longitudinal full penetration weld and the bottom circumferential weld, radiographic inspection of the two welds for the bottom plug, and helium leak testing and dye penetrant testing of the welds for the top lead plug and top plate, respectively. The acceptance leak rate for helium leak testing is less than or equal to 10^{-4} atm - cc/sec. The less rigorous dye test procedure used for the top end plate can be considered acceptable due to the helium leak testing of the inner weld, and due to the fact that two seals are used instead of one, as for the longitudinal weld. Radiographic inspection of the top plug welds is not feasible due to the fact that irradiated fuel will already be installed before the tests can be made.

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 or sag of the fuel cladding. It was found to be 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% for postulated fuel oxidation of defective fuel rods. The staff concludes that the NUHOMS design has provided sufficient means to assure that the fuel cladding is protected against degradation.

5.2 DESCRIPTION OF REVIEW

5.2.1 Applicable Parts of 10 CFR 72

Paragraph (1) of Section 72.122(h) is pertinent to storage of spent fuel in NUHOMS. 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." Paragraphs (2) and (3) of that section relate to underwater storage of fuel and to the off-gas and ventilation systems, respectively, and are not applicable to this review. Paragraphs (4) and (5) deal with monitoring and handling and retrievability operations, respectively, and are addressed elsewhere in this document.

5.2.2 Review Procedure

5.2.2.1 Design Description

The NUHOMS design provides protection of the fuel cladding by storing fuel assemblies in an inert atmosphere of helium. The helium atmosphere is first established after the fuel is loaded into the DSC. The loaded DSC is welded closed, and the weld tested with the dye penetrant method, drained of water by pressurizing the cavity with helium, and evacuated. A vacuum of 3 Torr is drawn on the DSC cavity for not less than 30 minutes. This stable vacuum pressure of 3 Torr will result in an inventory of oxidizing gases in the cavity of less than 0.25 volume %. Then the DSC is back-filled with helium to an unspecified pressure for purposes of helium leak-testing of the primary weld.

After the end weld is checked for leaks, the DSC is again evacuated and backfilled with helium at 2.5 $\pm$ 2.5 psig. The evacuation lines are sealed and the top end cover is welded to the DSC. The field welds and the shop welds on the bottom and along the longitudinal seam are expected to maintain the internal helium atmosphere intact for the full time of storage of the DSC in the HSM. No device (e.g., gauge) is made part of the system for verifying the maintenance of the helium atmosphere.

5.2.2.2 Acceptance Criteria

The confinement barriers and systems design will be considered acceptable if the TR shows that: (1) there is a high likelihood that the DSC internal helium atmosphere will remain intact; (2) there is no long term cladding degradation mechanism in a helium atmosphere which could cause significant degradation or gross ruptures; and (3) 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.

5.2.2.3 Review Method

The NRC staff review of the TR 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 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 inert atmosphere. Based on the literature search, calculations were performed of 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 NUHOMS storage conditions. The staff also evaluated the possible long term creep and sag of the spent fuel under these storage conditions since creep or sag could impact on removal of the fuel from storage.

5.2.2.4 Key Assumptions

The assumptions made in review of the TR regarding confinement systems are listed below.

1. The diffusion rate of helium through the DSC is no greater than 10^{-8} g-moles/year at nominal design conditions and as much as 10^{-5} g-moles/year at accident conditions, as stated by the applicant.

2. The values used for various properties of the zircaloy cladding in the analysis of diffusion controlled cavity growth (DCCG) and the DCCG mathematical model lead to a very conservative estimate of degradation.

3. The fuel cladding is protected at steady state temperatures of up to 340°C and on short term transients up to 570°C if in an inert atmosphere.

5.3 DISCUSSION OF RESULTS

The following evaluation covers DSC integrity, potential for long term fuel rod failure, potential cladding creep or sag, and potential oxidation of fuel during cask dry-out or off-normal behavior.

5.3.1 DSC Integrity

In the review of the structural analysis of the DSC, the staff found the design acceptable.

The commitment to design and fabricate the DSC's bottom circumferential and longitudinal welds to the ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsections NB and NF for Class 1 components provides assurance of leak-tightness at these locations.

The top end plate welds are made in the field. The TR states that end plate welds are to be ultrasonically tested, or tested by dye penetrant method in accordance with the ASME Code as stated above. The dye penetrant method of testing reveals information about the weld surface only, hence a weld tested by this method does not yield as much information as the radiographic method. However, the top end welds cannot be radiographed because irradiated fuel will be in place before these tests can be performed. The staff finds this procedure acceptable because the primary welds are first leak-tested by a helium detector and because the two top end plates represent a double seal.

Since the DSC contains no penetrations for sampling or gauges, there are no diffusion or leakage paths for helium other than the welds and the primary metal. Presuming the weld integrity to be equivalent to that of the parent metal, the staff also concludes that diffusion is not a potential mechanism to permit escape of helium and ingress of oxygen.

The staff concludes that DSC design and fabrication methods will result in a high likelihood that the internal DSC helium atmosphere will remain intact over its storage lifetime.

5.3.2 Potential for Long Term Fuel Rod Failure

Calculations were performed on the potential failure of the cladding by the diffusion controlled cavity growth mechanism, which is the only mechanism postulated to occur under the NUHOMS storage conditions. The method used is described in Appendix A of the SER for the initial NUHOMS TR (Reference 42). Following the earlier assumptions, a constant ambient temperature of 70°F was used in the analysis. The temperature dependence of grain boundary decohesion is established using the temperature decay curve provided in the current TR in Figure 8.1-28. Since the data terminates at about 10 years from the beginning of storage, it was conservatively assumed that the temperature would remain constant thereafter, that is, for the remaining ten years.

Since the values of all parameters in the equation given in Appendix A of the initial SER, except for the exponential term involving the temperature decay, were identical, the calculation reduces to a comparison of the integrals of the two exponential terms. The difference was found to be insignificant. Therefore, the area of decohesion at the end of the twenty-year storage life is the same as that found previously, less than 4 percent. Hence, the requirements of 10 CFR 72 Section 72.122(h) are met.

5.3.3 Potential Cladding Creep or Sag

Cladding creep or sag could impact on the removal of fuel from storage. The potential for cladding creep was analyzed first, using the creep equations of Peehs et al. (Reference 35). The temperature profile was conservatively broken down into five or ten year constant temperature periods to estimate the cladding creep. For stresses of 80 or 100 MPa, the creep was found to be less than 1 percent. The sag of the cladding was then calculated using a standard linearly loaded beam formula. If no credit for inertia is taken for the fuel itself, the maximum sag was found to be 0.015 inches. If the fuel also resists bending, then the maximum sag was found to be 0.006 inches. For this analysis, the inter-grid distance was assumed to be 24 inches. For an inter-grid spacing of 26 inches, the maximum sag was found to be 0.020 inches. Since the space available between the fuel rods and the DSC basket is much greater than the 0.020 inches, it is not anticipated that sag would impede the removal of the fuel assemblies.

5.3.4 Potential Oxidation of Fuel During Cask Dry-Out or Off-Normal Behavior

The NUTECH thermal analysis, with which the staff concurs, indicates that cask dry-out or off-normal behavior could involve a temperature excursion of up to 375°C in 48 hours. It was conservatively assumed that air was present for the entire time period. The temperature profile given in Figure 8.2-12 of the TR was divided into eight six-hour periods. For each time segment, the oxidation rate was determined. Oxidation front velocity data were taken from Einziger and Cook (Reference 40) and Kohli et al. (Reference 41). The maximum length of fuel oxidized was found to be 2.1 inches for fuel rods containing defects. The cladding strain was estimated to be much less than 1 percent, so that defect extension or fuel powdering is not anticipated. However, as noted previously, radiological precautions must be taken to protect personnel during operations in which fuel could be exposed.

6.0 SHIELDING EVALUATION

6.1 SUMMARY AND CONCLUSIONS

The methods used for designing the NUHOMS-24P shielding, and the resultant shielding design, are similar to the design for the NUHOMS-07P system containing seven irradiated fuel assemblies which has been reviewed in Reference 42. The neutron and gamma ray design basis source strengths are slightly higher than the smaller capacity design; however, the basic shielding design of the NUHOMS system readily accommodates the slightly higher source strengths. The results of experimental measurements from a real cask similar to the NUHOMS-24P design also provides a benchmark for the shielding design methods.

The NUHOMS shielding design conforms to the ALARA requirements of 10 CFR 72 and to acceptable shielding methods and practices. The staff concludes, based on the TR analysis, that the shielding is designed to ensure that the surface dose rates satisfy the criteria established in the TR subject to the following conditions:

1. No more than twenty-four (24) fuel assemblies meeting the specifications discussed in Chapter 12 of this report are contained in the DSC, or

2. The maximum neutron source strength per DSC is $<3.715\times10^{9}$ neutrons/sec, and the maximum gamma ray source strength per DSC is $<3.85\times10^{16}$ MeV/sec (1.11×10^{17} gammas/sec).

6.2 DESCRIPTION OF REVIEW

6.2.1 Applicable Parts of 10 CFR 72

The applicable part of 10 CFR 72 regarding the shielding evaluation of the NUHOMS-24P is the requirement of 10 CFR Part 72.3 related to ensuring that occupational exposures to radiation are as low as reasonably achievable, and 10 CFR Part 72.126 relating to criteria for radiological protection.

6.2.2 Review Procedure

6.2.2.1 Design Description

The principal design criterion for the NUHOMS-24P module is to limit the average external surface contact dose (gamma ray and neutron) to site workers to less than 20 mrem/hr. The design criteria during handling and transfer operations is to limit contact dose to less than 200 mrem/hr.

6.2.2.2 Acceptance Criteria

The shielding design is acceptable if the shielding evaluation results provide reasonable assurance that the design criteria indicated above are satisfied in the NUHOMS-24P system design.

6.2.2.3 Review Method

The TR shielding analysis was reviewed. Independent or confirmatory calculations were not performed. Rather, an assessment of the appropriateness of the shielding methods was made. Checks of the results for consistency with the similar NUHOMS-07P system were made as well as checks for self-consistency of the results.

6.2.2.4 Key Assumptions and Computer Codes

The major assumption in the shielding design is the source strength specification for the fuel to be stored, which is described in Section 2.2 of this report.

Two computer codes were used in the shielding analysis reported in the TR. ANISN, a one-dimensional discrete ordinates code, was used to estimate the neutron and gamma-ray dose rates at the outer HSM wall, the DSC top and bottom cover plate surfaces, and the TC outer surfaces. The ANISN calculations used the CASK cross section library, which includes 22 neutron energy groups and 18 gamma ray energy groups. QAD-CGGP, a three dimensional point kernel shielding code, was used for the gamma ray shielding analysis of the HSM door, the DSC and cask end sections, the DSC-cask annular gap and the HSM air vent penetrations.

6.3 DISCUSSION OF RESULTS

The following evaluation covers source specifications, HSM dose rates, and cask-DSC dose rates.

Source Specifications

The neutron and gamma radiation sources include the design basis irradiated fuel and activated portions of the fuel assemblies. The shielding analysis includes both primary neutrons and gamma-rays from these sources as well as an approximation of the secondary gamma rays from interactions of neutrons with the DSC and shielding materials. A more rigorous estimate of secondary gamma rays is included in the ANISN calculations, while in the QAD-CGGP calculations, secondary gamma rays have been approximated by increasing the primary gamma ray source strength. This approximation was justified by comparison with experimental data to confirm that the calculational results are giving conservative results.

The shielding is designed for a neutron source strength of 3.715×10^9 neutrons per second and a gamma ray source strength of 3.85×10^{16} MeV per second. Any combination of fuel irradiation time, burnup, specific power, enrichment, post irradiation time, and selection of assemblies to be loaded into a DSC is acceptable for storage in the HSM if the neutron and gamma ray source strengths do not exceed these criteria. The design basis is derived from a burnup analysis of 4 weight percent ^{235}U initial enrichment PWR fuel irradiated to an average fuel burnup of 40,000 MWd/MTHM at a specific power of 37.5 MW/MTHM, and a post irradiation time of ten (10) years. Irradiated fuel assemblies that meet these criteria are bounded by the neutron and gamma ray sources used in the shielding analysis.

The neutron and gamma ray source energy spectrum used for the shielding analysis were derived from an ORIGEN burnup calculation and are reported in TR Tables 7.2-1 and 7.2-2, respectively.

HSM Dose Rates

A dose rate of 7 mrem/hr for the HSM top and lateral surfaces was calculated using the ANISN discrete ordinates transport code and a cylindrical model of the DSC/HSM. This contact surface dose rate is less than the design criteria of 20 mrem/hr. Dose rates of 45 and 24 mrem/hr were calculated at the HSM door and 2 meters from the door, respectively, using a combination of results from an ANISN slab calculation modeling the ends of the DSC and a QAD calculation of the shielding effectiveness of the HSM door. These dose rates are less than the design criteria contact dose rates of 200 mrem/hr for workers performing operations.

Dose rates at the air ventilation inlets and outlets were calculated using the QAD-CGGP code and a manual albedo method to account for radiation streaming through the air ducts. Dose rates of 96 mrem/hr and 63 mrem/hr were calculated at the center of the air inlet and outlet ducts, respectively. A dose rate of 3600 mrem/hr at the air outlet duct was calculated for the accident condition of the shielding cap being removed.

Cask-DSC Dose Rates

Maximum cask surface dose rates were calculated to be 168 mrem/hr on the radial surfaces of the cask using a cylindrical model of the cask-DSC in the ANISN code. A maximum dose rate of 191 mrem/hr was calculated in the cask-DSC annulus. These dose rates are less than or equal to the design criteria dose rates of 200 mrem/hr.

The NRC staff concludes that, based on the material supplied in the TR, the NUHOMS-24P design meets the design criteria as stated in the TR.

7.0 CRITICALITY EVALUATION/BURNUP

7.1 SUMMARY AND CONCLUSIONS

Criticality safety cannot be assured for the NUHOMS-24P system under the conditions that have previously been considered for licensing of independent spent fuel storage installations using a dry storage concept, i.e., criticality safety is assured assuming loading of system with unirradiated fuel of maximum initial enrichment with optimal interstitial water density. Additional measures have been considered to provide assurance of nuclear criticality safety for the NUHOMS-24P system, including:

1. evaluation of fissile isotope concentrations and stable fission product absorbers in irradiated fuel (i.e., burnup credit)

2. modification of operational procedures to ensure fuel loading in a moderator solution with sufficient soluble poisons to assure nuclear criticality safety

Since consideration of the generic issue of the use of allowance for burnup credit in the safety evaluation of Independent Spent Fuel Storage Installations has not yet been completed by the Nuclear Regulatory Commission, allowance for burnup credit has not been accepted by NRC staff as a basis for the safety evaluation of the NUHOMS-24P system. However, nuclear criticality safety can be assured in the NUHOMS-24P design if the DSC is filled with borated water ($\geq$ 1810 ppm boron) during loading and unloading operations and if the irradiated fuel assemblies are loaded with the DSC submerged in a borated-water PWR spent fuel pool. The maximum effective reactivity under these conditions with optimal moderator density and 4% unirradiated fuel has been determined to be < 0.98.

Since unirradiated fuel will not be loaded into the DSC, there will be a reactivity safety margin realized, although it has not been quantitatively evaluated in this safety evaluation because of the outstanding research issues. Thus, it is concluded, based on the analysis presented in the TR and response to questions, that the NUHOMS-24P system is designed to provide assurance of nuclear criticality safety. The NUHOMS-24P system is determined to be in compliance with 10 CFR 72.124 as long as the following conditions are met:

1. Irradiated fuel initial enrichment equivalent[a] $\leq$ 1.45 weight percent ^{235}U;

2. DSC filled with borated water ($\geq$ 1810 ppm boron) and submerged in borated-water PWR spent fuel pool during loading and unloading operations;

3. The irradiated fuel assemblies are not more reactive than the design basis 15x15 rod fuel assemblies;

4. Borated water is drained from the DSC within 50 hours of being removed from the spent fuel pool;

7.2 DESCRIPTION OF REVIEW

7.2.1 Applicable Parts of 10 CFR 72

The applicable part of 10 CFR 72 regarding nuclear criticality safety is the requirement of 10 CFR 72.124.

a The initial enrichment equivalent of an irradiated fuel assembly is the ^{235}U enrichment of unirradiated fuel assemblies which would exhibit the same reactivity as the irradiated fuel assembly.

7.2.2 Review Procedure

7.2.2.1 Design Description

The NUHOMS-24P DSC is designed to provide nuclear criticality safety during wet loading and unloading operations. After fuel loading and DSC drying, the irradiated fuel assemblies are not moderated and therefore criticality safety is assured for subsequent operations and configurations.

The moderator density conditions are an important factor for criticality safety during fuel loading into the DSC and if removal of fuel assemblies from the DSC is necessary for any reason. The DSC is initially filled with borated water ($\geq$ 1810 ppm boron) prior to placement in the spent fuel pool, which is also borated. The loaded DSC is removed from the fuel pool and the DSC cavity is subsequently dried and backfilled with helium as part of the DSC closure operation. Flow paths are provided in the DSC design to ensure that the DSC draining or refilling process is a controlled and determinate process.

During transfer and storage of the canistered spent fuel, ingress of water into the helium-filled DSC is precluded by its welded seals and its presence in the TC and HSM, respectively. To ensure criticality safety, the NRC staff also limits its approval of the use of this design to storage on flood-free sites, eliminating water intrusion into the DSC as a credible event.

Further, for loading or unloading of the DSC, any licensee shall have in place fuel selection and verification, fuel identification and verification, borated water measurement and verification, and fuel handling procedures.

7.2.2.2 Acceptance Criteria

The requirement of 10 CFR 72.73 is determined to be satisfied if the 95% probability/95% confidence (95/95) effective multiplication factor for the NUHOMS-24P design is demonstrated to be less than 0.95, and the 95/95 effective multiplication factor for unirradiated fuel, which is not to be stored, is demonstrated to be less than 0.98.

7.2.2.3 Review Method

The criticality analysis presented in the TR and supplementary response to questions was reviewed. Independent and confirmatory calculations were also performed to verify important sensitivities in the criticality analysis.

7.2.2.4 Key Factors/Assumptions

Key factors and assumptions in the criticality safety analysis were:

<u>factors</u>

1. The maximum initial fuel enrichment is 4.0 wt. % ^{235}U; note, however, that unirradiated fuel will not be loaded in the DSC.

2. The DSC is filled with borated water ($\geq$ 1810 ppm boron) and submerged in a borated-water spent fuel pool ($\geq$ 1810 ppm boron) during loading and unloading operation,

3. Only irradiated fuel assemblies with an initial enrichment equivalent $\leq$ 1.45 wt. % ^{235}U will be loaded in the DSC,

4. DSC draining is accomplished within 50 hours of removal from the spent fuel pool.

<u>assumptions</u>

1. Fuel assemblies to be stored are no more reactive than the design basis 15x15 rod array,[b]

2. The DSC can not be filled with unborated water or borated water with less than 1810 ppm boron,

b The 15x15 rod array was determined to be the most reactive of several fuel assemblies evaluated in the TR.

3. No accident can occur which could alter the mechanical configuration of the stored array of irradiated fuel assemblies.

7.3 DISCUSSION OF RESULTS

7.3.1 Analytical Methods

The criticality safety analysis presented in the TR was performed using the Criticality Safety Analysis Sequence No. 2 (CSAS2) included in the SCALE-3[c] package of codes. The CSAS2 sequence and the 123GROUPGMTH master cross-section library included in the SCALE-3 system were used in calculating the effective neutron multiplication factor, k_{eff}, for the design basis configuration and the several evaluations of sensitivities to design parameters. The CSAS2 analysis sequence used two cross-section processing codes (NITAWL and BONAMI), and a three dimensional Monte-Carlo code (KENO-V) for calculating the multiplication factor for the DSC fuel assembly arrays.

The NRC staff performed independent calculations for this safety evaluation report using the NITAWL and KENO-V codes from the same SCALE-3 safety evaluation system. The geometry modeling of the fuel assemblies and DSC internals were independently developed for these calculations.

7.3.2 Design Basis Calculations

Misloading of unirradiated fuel was determined to be the worst case nuclear criticality condition that can reasonably be conceived for the operation of the NUHOMS-24P system. Analyses were performed to confirm an adequate criticality safety margin for this worst case configuration.

The criticality safety analysis presented in the TR shows that the loading of 24 unqualified unirradiated fuel assemblies with an enrichment of 4 wt. % ^{235}U would result in a (95/95) k_{eff} of 0.887. If the draining of

c SCALE-3: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, Oak Ridge National Laboratory, Revision 3, December 1984.

the DSC were not accomplished prior to the onset of boiling and if optimal moderator density conditions were realized, the worst case (95/95) k_{eff} is determined to be 0.979. The staff's evaluation confirms these results.

Note that in this instance, attainment of a condition of optimal moderation for unirradiated fuel is a non-mechanistic assumption. Unirradiated fuel does not constitute a significant heat source to reduce water density to an optimum, as might be postulated for irradiated fuel.

For design basis irradiated spent fuel with a 0.66/kW/assembly decay heat rate, the heatup rate for the moderator in the DSC cavity is no greater than 2^{o}F/hr following removal from the spent fuel pool. Thus for a nominal fuel pool temperature of 100^{o}F, the boiling point of the DSC moderator would not be reached for 50 hours. It is also recognized that the reactivity of irradiated fuel assemblies is less than unirradiated fuel assemblies, although the reactivity of irradiated fuel assemblies is not evaluated in this report.

In the event that the DSC must be unloaded following the second backfilling with helium gas, the irradiated fuel assemblies may have reached a temperature in excess of 600^{o}F and the DSC may have reached a temperature in excess of 200^{o}F (Reference 1). Before safe unloading can be accomplished, the DSC must be reflooded with borated water. The staff has considered this case.

Based on data from references in footnotes d, e, f, and g, the solubility of boric acid in water at 100^{o}F is approximately 13,000 ppm boron. This is well in excess of the 1810 ppm solution being injected into the cask. The solubility of boric acid in water increases with increasing temperature. Therefore, the raised solution temperature due to heat transfer from the fuel would only further increase the solubility of boric acid in water. A bounding

d WCAP-1570, January, 1961, D.E. Byrnes and W.E. Foster.
e Boric Acid Properties Data, U.S. Borax Research Corporation, Anaheim, California.
f Supplement to Mellor's Comprehensive Treatise on Inorganic and Theoretical Chemistry, Volume V, Boron, Part A: Boron-Oxygen Compounds.
g Boron, Metallo-Boron Compounds and Boranes, edited by Roy M. Adams, Interscience Publishers.

conservative heat transfer calculation indicated that only about 0.3% of the water inventory could be removed by steaming in the first hour after the water is introduced to the fuel array. This represents an insignificant increase in the boric acid concentration. In conclusion, there is a very large margin between the boric acid concentration and the solubility limit of boric acid in the temperature range of interest and a reduction of the boric acid concentration is not possible for this scenario.

Further, as discussed above, any licensee shall have in place site-specific procedures for fuel selection and verification; fuel identification and verification; borated water measurement and verification; and fuel handling. The staff concludes that such procedures are necessary to assure that spent fuel is "...handled, stored and transported in a manner providing a sufficient factor of safety to require at least two unlikely independent and concurrent changes in conditions before a criticality accident is possible."[h] The staff's intent here is to identify those site-specific procedures that shall be implemented to assure the staff's factors and assumptions set forth in Section 7.2.2.4 for this criticality analysis are validated.

On the basis of the analysis presented in the TR, the supplementary analysis presented in response to questions, and the operating controls and limits, it is concluded that the NUHOMS-24P system is designed to be maintained in a subcritical configuration and to prevent a nuclear criticality accident in compliance with 10 CFR 72.124.

h ANSI/ANS 8.17-1984 "Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors," p. 2. Endorsed with modification by Regulatory Guide 3.58.

8.0 OPERATING PROCEDURES

8.1 SUMMARY AND CONCLUSIONS

The staff reviewed proposed operations described in Sections 5 and 9* of the TR. Portions of Sections 1 (1.3.1), 3 (3.1.2) and 4 (4.2.3, 4.5, and 4.7) of the TR contain summaries of operating procedures and were also reviewed.

Operations described in the TR are intended to serve as an example only and are not submitted for approval. Therefore, this review is limited to evaluating the feasibility of accomplishing the various activities. Approval of operations by the staff must await submittal of a site-specific application.

The staff concludes from its review that the operating sequence and steps proposed in the TR are feasible. If a site-specific applicant develops his own detailed operating procedures from the TR descriptions, there is no reason to believe they could not be made to meet the NRC's regulatory requirements. However, since NUHOMS is a new system that has not been built and tested, approval of site-specific procedures will be contingent upon successful demonstration of most "first-of-a-kind" features.

8.2 DESCRIPTION OF REVIEW

8.2.1 Applicable Regulations

The regulations used in the review of the TR included appropriate parts of 10 CFR 20 under the heading of "PERMISSIBLE DOSES, LEVELS, AND CONCENTRATIONS," and those paragraphs of Subparts E and F of 10 CFR 72 related to potential operational accidents (e.g., cask drop), off-normal events, and radiological doses.

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* Section 9.6, "Decommissioning Plan," of the TR is reviewed in Section 11 of this SER.

8.2.2 Review Procedure

8.2.2.1 Design Description

Section 5 of the TR presents a generic description of the handling, transfer and storage operations for NUHOMS. The operations considered unique to this system include:

1. Water filling of DSC and sealing of Cask/DSC annulus prior to lowering cask into spent fuel pool

2. DSC top lead cover welding of inner seal and weld inspection

3. DSC evacuation and helium backfill

4. DSC top cover plate welding and weld inspection

5. Draining of cask/DSC annulus and placing TC top cover

6. Transfer of fuel across the site on a specially designed vehicle in the TC which is built specifically for this use and is able to unload the DSC at the HSM

7. Positioning and aligning the TC with respect to the HSM opening while it sits on the transfer vehicle

8. Pushing the DSC into the HSM from the TC cavity

9. Reversing the order of loading the DSC into the HSM in order to be able either (1) to retrieve the spent fuel from the DSC on-site or (2) to ship the loaded DSC off-site.

8.2.2.2 Review Criteria

Since all operations are generic and no approval is sought, acceptance criteria are not applicable to this review. The review criteria for suitability of the operating procedures are based on: (1) the identification of appropriate steps for the protection of operating personnel, the public and the equipment, and (2) the feasibility of performing the operations.

8.2.2.3 Review Method

The sequence of operations and the step-by-step procedures proposed in the TR for the handling, transfer and storage of spent fuel were reviewed to determine if any portion of the proposed system might not function as planned. The reviewers used engineering judgment and past experience in a review of all proposed steps to reach a determination of feasibility. For those situations in which accidents might occur, a judgment was made of whether the results reported in the TR were reasonable or, lacking results, whether mitigating measures were available that could be implemented on a site-specific basis.

In the review of NUHOMS operations, special attention was given to the following issues.

1. Are inspection procedures and records normally available to determine the characteristics and the mechanical and structural integrity of fuel assemblies prior to loading them into a DSC?

2. Is the DSC able to withstand some reasonable combinations of loads, including various drops at normal TC transfer and placement heights, when used with the TC and transfer vehicles, while still maintaining its mechanical integrity, including retrievability of the fuel?

3. Are the dose rates, distances, and worker residence times during the DSC top welding operations reasonable and do they result in acceptably low exposures?

4. Are the dose rates, distances, and worker residence times for loading the DSC into the HSM reasonable, and do they result in acceptably low exposures?

5. Are the dose rates, distance of personnel from the DSC in the HSM, and personnel residence time during normal operation, off-normal events and

accidents reasonable and do they result in dose rates below levels specified by regulations?

6. Are the alignment dimensional tolerances between the HSM and the TC achievable and can the DSC be easily retrieved from the HSM after 20 years of storage?

7. Are the dose rates, distances, and worker residence times for the removal of fuel from the DSC reasonable and do they result in acceptably low exposures?

8.2.2.4 Key Assumption

It is assumed that approval of operating procedures will be given only on a site-specific basis.

8.3 DISCUSSION OF RESULTS

Although NUTECH is not seeking approval of the generic operations outlined in Section 5 of its TR, the NRC staff has evaluated Section 5, as well as the above issues. Based on engineering judgment and past experience with nuclear plant equipment and general level of personnel capability, the staff believes that the appropriate steps have been identified for the protection of operating personnel, the public and the equipment, and that the proposed operations are feasible.

The radiological guidelines for handling potentially contaminated equipment that apply to the removal of fuel from the DSC must include the requirement that personnel use respirators or supplied air to protect the health of the operations personnel.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

9.1 SUMMARY AND CONCLUSIONS

The staff has reviewed the proposed acceptance tests and maintenance programs for the storage of spent fuel in NUHOMS. Most of these activities are site-specific or are included as a part of the Codes of Design and Construction, which the vendor has committed to via its quality assurance program. The TR does specify the following generic requirements that must be met by the system:

1. The dose rates at the end of the DSC shield plug and at the surface of the HSM after the DSC is first inserted are restricted to specified values consistent with ALARA principles.

2. The maximum rise in air temperature from the HSM inlet to the HSM outlet after the DSC is initially loaded into the HSM is limited to a predetermined figure of 60°F. At this value the maximum fuel cladding temperature is predicted to remain below 340°C.

3. Daily inspections (surveillance) of the HSM air inlets and outlets is required to ensure that airflow is not interrupted. An annual inspection of the HSM internals is also recommended to identify potential airflow blockage and material degradation. The results of such inspections may require corrective action, which could be classified in the category of maintenance.

The staff finds that these generic activities, when augmented by a complete set of site-specific acceptance tests and maintenance programs, will provide for safe operation of the TC, the DSC, and the HSM with one exception. The staff requires the same level of pre-operational testing for the 24P design as was required for the 07P design. See Table 9.2-1 of Reference 22. Table 9.2-1 of Reference 22 outlines three sets of pre-operational tests for the NUHOMS-07P system. Two of the three are incorporated in Table 9.2-1 of the TR. The set missing involves the heat transfer, temperatures and air flow in various configurations for a DSC inserted in the HSM. The NRC staff requires the licensee to perform this set of tests for a DSC loaded with a heat source other than radioactive material, prior to spent fuel loading.

9.2 DESCRIPTION OF REVIEW

The review was performed by grouping the proposed test and maintenance activities into the following phases:

1. Design, procurement and fabrication of components.

2. Site commissioning: construction and installation of the system leading to start-up, including pre-operational testing.

3. Operational.

The tests and maintenance activities proposed in the TR for each phase were evaluated for completeness. Those activities that will be the subject of a site-specific application were not reviewed in detail. Those proposed generically were reviewed to determine whether they provide for safe operation of the three components which are important to safety.

9.3 DISCUSSION OF RESULTS

The material relating to acceptance tests and maintenance programs is very sparse and widely scattered throughout the TR. Table 3.3-4 of the TR identifies three major components as being important to safety: (1) the transfer cask, (2) the dry shielded canister, and (3) the horizontal storage module. These three components have codes of design and codes of construction associated with them. The vendor has formally committed to these codes of design and construction as an integral part of the NUHOMS 24P system.

9.3.1 Acceptance Tests

Although acceptance tests during spent fuel handling, transfer and storage are for the most part considered to be site-specific, subsection 10.3.2 of the TR does establish limiting conditions on certain critical parameters prior to the time that passive storage begins. Dose rates at the end of the DSC shield plug and at the surface of the HSM after the DSC is first inserted are restricted to specified values consistent with ALARA principles. The maximum rise in air temperature from the HSM inlet to the HSM outlet after the DSC is initially loaded into the HSM is also limited to a predetermined figure of $60^{o}F$. At this value, the maximum fuel cladding temperature is predicted to remain below $340^{o}C$.

Acceptance tests are not presented as such in the TR. They are primarily implied as a part of the codes of design and construction. Examples of these codes are given below:

1. Code of Design for DSC: ASME Code Section III, Division 1, Subsection NB.

2. Code of Design for TC: ASME Code Section III, Division 1, Subsection NC.

3. Code of Design for TC lifting trunnions: ANSI N14.6.

4. Code of Design for HSM: ACI 349-85.

5. Code of Design for DSC Supports: AISC Code 8th Edition.

6. Code of Construction for DSC: same as design code.

7. Code of Construction for TC: same as design code.

8. Code of Construction for TC trunnions: ASME Code Section III, Division 1, Subsection NC.

9. Code of Construction for HSM: ACI 318-83.

10. Code of Construction for DSC supports: same as design code.

Section 10 of the TR has various operational controls and limits for the performance of the system prior to service as well as immediately following emplacement of a DSC into an HSM. Examples of these controls and limits are:

1. Fuel characteristics
2. Dose rates for the on-site TC
3. Weld inspection standards for the DSC
4. Vacuum pressure required for drying the DSC following seal welding
5. Helium leak testing of the DSC and content of helium following backfilling
6. Dose rates for HSM
7. Surveillance of the HSM air inlets and outlets.

Section 9.2 of the TR refers to a pre-operational testing program that was specified for the NUHOMS-07P design. Although this program is relevant to the 24P design, the TR does not present any data or results. Thus the claim made in the TR that pre-operational testing of the 07P design "will provide sufficient data to demonstrate that the analytical methods described in this report provide conservative thermal and radiological results," is premature. The NRC staff therefore requires the same level of pre-operational testing for the 24P design as was required for the 07P design. (See Section 9.2 and Table 9.2-1 of Reference 22.) This is necessary to confirm the validity of the analytical methods with regard to the thermal hydraulic calculations.

The calculated maximum fuel cladding temperature, assuming 70°F ambient conditions, is 339°C, just 1°C below the limit of 340°C. The staff therefore requires that acceptance testing be performed in the same manner as for the 07P design, to confirm the validity of this design. Such testing must confirm that the air temperature rise from inlet to outlet is less than 60°F for a fully loaded (15.8 kW) DSC.

In general, the generic activities, when augmented by a complete set of site-specific acceptance tests, will provide for the safe operation of the DSC, TC and HSM. However, the validity of the analytical thermal hydraulic model must be confirmed by acceptance testing. The staff accepts the TR sections that refer to acceptance tests (Chapters 3, 4, 5, 9 and 10) except for Section 9.2.

9.3.2 Maintenance Program

Maintenance of NUHOMS is addressed in the TR in Sections 4.5 and 5.1.3. Section 4.5 of the TR covers the routine and annual inspection considered necessary for the TC. It includes: (1) visual inspection of such items as the cask exterior for cracks, damaged bearing surfaces, leakage from the neutron shield fittings, (2) visual inspection of all threaded parts for wear and burrs, (3) check of quick-connect fittings, and (4) visual inspection of interior surfaces of the cask. It also includes an annual inspection of the neutron shield pressure relief system and the cask lifting points and cask lifting yoke.

Sections 5.1.3 and 10.3.3.1 of the TR discuss the surveillance requirements of the HSM air inlets and outlets. Basically, however, the TR maintains that:

1. Maintenance of the system in order to assure continuous operation is not required since the system is totally passive once the spent fuel is in long term storage. However, daily inspection (surveillance) of the HSM air inlets and outlets is required to ensure that airflow is not interrupted. An inspection of the HSM internals at intervals of five and ten years after initial storage is also required for at least one HSM per installation to identify potential material degradation. The detailed procedures to be used during such inspections, which must address criteria for determining the effect of degradation, are site-specific. The results of such inspections may require corrective action, which could be classified in the category of maintenance.

2. Maintenance of the fuel handling and transfer equipment is site-specific. The major components involved are the transfer cask, transfer trailer and skid, cask restraint system, and hydraulic ram. (Note: the devices used for lifting heavy loads while the DSC is in the reactor or spent fuel pool building are assumed to be covered under a 10 CFR Part 50 license).

The NRC reviewers found the information on maintenance of equipment and procedures supplied in the TR to be adequate.

In summary, the TR treats acceptance testing and maintenance in the following ways:

1. Pre-operational acceptance testing of the system is site-specific.

2. Acceptance testing of components that are important-to-safety (the DSC, DSC internals and HSM) is subject to industry codes and standards, NUTECH's quality assurance program (as applicable), a site-specific applicant's quality assurance program (as applicable) and various procurement specifications, the last two items being site-specific.

3. Generic limiting conditions for operation are applied in the TR which, if not met, require corrective action.

4. Surveillance of the HSM exterior during the passive storage phase is required, which may result in maintenance activities if the NUHOMS performance is jeopardized. As noted above, detailed surveillance procedures are site-specific.

5. Maintenance of equipment used in handling and transfer of spent fuel is a site-specific requirement.

With one exception, the staff finds that this treatment is acceptable and that the generic activities, when augmented by a complete set of site-specific items, will provide for safe operation of the TC, DSC and the HSM when applied to a site-specific situation. Special attention needs to be given to establishing criteria which define when corrective actions are required. The single exception is that the pre-operational test requirements of the 24P design need to be modified to reflect the level of testing required for the 07P design.

10.0 RADIOLOGICAL PROTECTION

10.1 ON-SITE RADIOLOGICAL PROTECTION

10.1.1 Summary and Conclusions

The shielding, confinement, and handling design features of the NUHOMS-24P conform to the on-site radiological protection requirements of 10 CFR 20, and are considered acceptable for the set of conditions assumed in this review. The NUHOMS-24P design and operational procedures are also consistent with the objective of maintaining occupational exposures as low as reasonably achievable (ALARA). Detailed discussions of access control, surveillance, and other operational aspects affecting on-site exposure are deferred to the site-specific license application.

10.1.2 Description of Review

10.1.2.1 Applicable Parts of 10 CFR 72

Part 72.24 of 10 CFR requires 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 reasonably achievable.

Part 72.126(a) of 10 CFR requires 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.

Part 20.101(a) of 10 CFR 20 states that any individual in a restricted area shall not receive in any period of one calendar quarter from radioactive material and other sources of radiation a total occupational dose in excess of 1.25 rems to the whole body. Part 20.101(b) states that, under certain conditions, the quarterly dose limit to the whole body is 3 rems in any calendar quarter.

Guidance for ALARA considerations is also provided in NRC Regulatory Guides 8.8 and 8.10 (References 13 and 14, respectively).

10.1.2.2 Review Procedure

10.1.2.2.1 Design Description

The main radiation protection features of the NUHOMS-24P design include (1) radiation shielding; (2) radioactive material confinement; (3) prevention of external surface contamination; and (4) site access control. Access to the site of the NUHOMS-24P array, although not specifically addressed in the TR, would be restricted by a periphery fence to comply with 10 CFR 72.106(b) restricted area requirements. The details of the access control features are site-specific, and would be described in the applicant's site license application.

The shielding features of the NUHOMS-24P are discussed in Section 7.3.2 and Appendix A of the TR. 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 contact dose rate to site workers to below an average of 20 mrem/hr

2. A lead 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 contact dose rate to 200 mrem/hr or less

4. Placing external shielding blocks over the HSM air outlets

5. Use of an internal shielding slab and wall around the HSM air inlet opening

6. Filling of the DSC cavity with borated water and the DSC-transfer cask annulus with demineralized water

7. 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 NUHOMS-24P control the release of gaseous or particulate radionuclides and are described in Section 3.3.2 of the TR. These features include:

1. The cladding of the stored fuel assemblies

2. The DSC containment pressure boundary

3. The inner and outer seal welds of the DSC

4. The DSC shielded end plugs and cover plates.

The DSC has been designed as a weld-sealed confinement pressure vessel with no mechanical or electrical penetrations. All the DSC pressure boundary welds are inspected according to the appropriate articles of the ASME Boiler and Pressure Vessel Code to ensure that the weld metal is as sound as the parent metal.

10.1.2.2.2 Acceptance Criteria

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

10.1.2.2.3 Review Method

The calculational methods used in the estimation of on-site doses are described in detail in the TR. These methods focused on the use of the ANISN, QAD-CGGP, SKYSHINE-II, and MICROSHIELD radiation transport codes, as well as manual albedo calculations, to calculate exposure rates around the DSC in a TC and an HSM. Independent or confirmatory calculations of these exposure rate calculations were not made. Rather, the calculational methods and results presented in the TR were reviewed for completeness, correctness, and internal consistency. The dose rate results were then used with estimated distance and occupancy rate data to assess the individual and collective on-site doses.

10.1.2.2.4 Key Assumptions

Radiation doses to on-site workers are not calculated in the TR. Rather, a summary of the operational procedures which lead to occupational exposures is presented, as are the number of personnel required, the estimated time for completion of each operation, and the average source-to-subject distance. Dose rate estimates for the specific areas to be occupied during these operations are not presented directly, but can be estimated from the exposure rate data, which are presented. The TR notes that the operations and labor estimates are provided only as an example, since a collective dose calculation of this type is required for a site-specific license application. The TR also states that the dose rates for the NUHOMS-24P system are similar to those of the NUHOMS-07P system, although the number of specific operations and the time required for their completion, as listed in Table 7.4-1 of the respective TRs, differ significantly.

The following method was used in this review to estimate the on-site dose. It was assumed that the working area dose rates around the surfaces of the TC or HSM are similar to those presented in Table 7.4-1 of the NUHOMS-07P TR, and that the labor-hour requirements for specific operations are similar to those listed in Table 7.4-1 of the NUHOMS-24P TR. For operations which were not listed in the NUHOMS-07P TR, the dose rates which are used were assessed in a recent site-specific application.

Both collective and maximally exposed individual dose results are presented for the phase of ISFSI operations from loading of the DSC to insertion into the HSM. The assessment does not include the dose to on-site workers not directly involved in ISFSI operations, which is highly dependent on site-specific factors.

10.1.3 Discussion of Results

The following evaluation covers ALARA considerations, radiation protection design features, and on-site dose assessments.

10.1.3.1 ALARA Considerations

The design of the NUHOMS-24P exhibits several features that are specifically directed toward ensuring that occupational doses are in accordance with the ALARA guidance given in Regulatory Guide 8.8, in addition to satisfying the requirements of 10 CFR 20. In addition to the radiation protection design features discussed below, specific considerations include administrative programs such as access control, the application of maximum acceptable dose rates related to access requirements, and provisions for shielding based on demonstrably conservative assumptions. Other considerations are identified in Section 7.1 of the TR.

10.1.3.2 Radiation Protection Design Features of the NUHOMS-24P

There are several radiation protection design features of the NUHOMS-24P described in Sections 7.1.2 and 7.3 of the TR. The principal radiation protection design features include provisions for shielding, confinement, and contamination control.

Shielding includes many features designed to reduce direct and scattered radiation exposure. The specific features were listed in Section 10.1.2.2.1.

The DSC has been designed as a weld-sealed confinement pressure vessel with no mechanical or electrical penetrations. All the DSC pressure boundary welds are inspected according to the appropriate articles of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB. These criteria ensure that the weld metal is as sound as the parent metal.

Contamination of the DSC exterior and transfer cask interior surfaces is controlled by placing demineralized water in the TC during loading operations, then sealing the DSC/cask annulus with a donut-shaped inflatable rubber tube.

10.1.3.3 On-Site Dose Assessment

Radiation doses to on-site workers were not calculated in the TR. Rather, a summary of the operational procedures which lead to occupational exposures is presented, as are the number of personnel required, the estimated time for completion, and the average source-to-subject distance. Dose rate estimates for the specific areas to be occupied during these operations are not presented. The TR notes that the operations and labor estimates are provided only as an example, since a collective dose calculation of this type is required for a site-specific license application. The TR also states that the dose rates for the NUHOMS-24P system are similar to those of the NUHOMS-07P system.

In the NUHOMS-07P TR, the estimated collective dose for the loading, transfer, and insertion of one DSC was about 0.26 person-rem, while the maximum individual dose for these operations was about 125 mrem. Using the assumptions stated in Section 10.1.2.2.4, the collective dose associated with the loading, transfer, and insertion of one DSC would be about 1.4 person-rem, and the maximum individual dose would be about 610 mrem. This dose rate could require the use of multiple worker crews, depending on the number of transfers in a given year.

A detailed assessment of operator doses and the possible provision of management or administrative controls to meet ALARA criteria is deferred to a site-specific license application.

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 deferred to site-specific applications.

10.2 OFF-SITE RADIOLOGICAL PROTECTION

10.2.1 Summary and Conclusions

The shielding and confinement design features of the NUHOMS-24P conform to the off-site radiological protection requirements of 10 CFR 72 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 effluent streams from the NUHOMS-24P. Off-site dose is, therefore, due strictly to direct and scattered radiation, the intensity of which is a function of distance. Site-specific factors such as the number of HSMs in the storage array, the distance and direction of the nearest boundary of the controlled zone, 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 NUHOMS-24P at a specific site.

10.2.2 Description of Review

10.2.2.1 Applicable Regulations

Sections 72.24(l) and (m) of 10 CFR require, in part, that a safety assessment be performed on the potential dose or dose commitment to an individual located outside the controlled area as a result of radioactivity releases caused by accidents or natural phenomena events.

Section 72.104(a) of 10 CFR requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual located beyond the controlled area shall not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 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 NUHOMS-24P operations, and (3) any other radiation from uranium fuel cycle operations within the region.

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 5 rem to the whole body or any organ from any design basis accident.

10.2.2.2 Review Procedure

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 and are described in Section 3.3.2 of the TR. The radiation shielding design features limit the direct radiation exposure rate and are described and analyzed in Section 7.3.2 and Appendix A of the TR. Additionally, Section 7.4 provides a dose-versus-distance curve from the shield analysis results.

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 TR. In particular, an accident resulting in the loss of both air outlet shielding blocks is analyzed in Section 8.2.1, while an instantaneous release of 30 percent of fission gas inventory is assessed in Section 8.2.8. Other accidents are assessed in Section 8.2 (e.g., floods, tornadoes, earthquakes, accidental cask drop, blockage of air inlets and outlets, etc.), but the TR concludes that none of these other accidents represent credible sources of off-site dose consequences. The NRC staff accepts this conclusion.

This evaluation focuses on the off-site doses resulting from normal operations and from the two postulated accident events which can have off-site dose consequences. These doses are assessed for compliance with 10 CFR 72. The minimum distance selected for the evaluation of compliance with this section is 200 m, which is a reasonable approximation of the minimum distance to the nearest residence beyond the 100 m controlled area required by 10 CFR 72.106.

10.2.2.3 Acceptance Criteria

Off-site radiological protection features of the NUHOMS-24P system are deemed acceptable if it can be shown that design and operational considerations which are not site-specific result in off-site dose consequences which are in compliance with the applicable sections of 10 CFR 72, and that these doses to off-site individuals are as low as reasonably achievable.

10.2.2.4 Review Method

The review for off-site radiological protection mainly involved a detailed evaluation of the methods applied and the results obtained in the applicable TR sections, supplemented by additional information provided by NUTECH 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 the ANISN, SKYSHINE-II, and MICROSHIELD computer codes, which were used to calculate gamma-ray and neutron dose equivalent rates at various locations in and around the HSM, and to generate a dose-versus-distance curve. The dose rates predicted by this curve for an off-site distance of 200 m was used to assess the general level of compliance with the minimum 100 m criterion of 10 CFR 72.104(a).

The accident analyses provided in Section 8.2 of the TR were evaluated for technical soundness, and the results of the DSC leakage event, which provides the highest off-site dose, were verified by independent calculation. The dose consequences were assessed at 200 m and 300 m. The former distance is used as a reasonable estimate for the distance to the nearest resident. The 300-m distance is used in the TR and is used here for comparison purposes.

10.2.2.5 Key Assumptions

The assessment of off-site dose from normal operations assumes the following:

1. The recipient of the dose resides at a distance of 200 m or 300 m from a two-by-ten array of NUHOMS-24P modules, which are filled with design basis spent fuel.

2. An occupancy factor of unity is assumed, and no credit is taken for attenuation in building materials.

3. The dose rate as a function of distance from a filled NUHOMS-24P is as illustrated in Figure 7.4-1 of the TR.

The consequences of the loss of shielding blocks event assume the following:

1. The air outlets on a single HSM or all air outlets on a 2x10 array of HSMs lose their shielding blocks and remain unshielded for a period of 7 days.

2. The resultant dose rate at the surface of the air outlet is 3600 mrem/hr, and decreases with distance according to the results presented in Table 8.2-2 of the TR.

3. The recipient of the dose is present for the entire duration of the recovery at a distance of 200 m or 300 m.

The consequence assessment of the DSC leakage event assumes the following:

1. The fraction of the noble gas (assumed to consist entirely of Kr-85) inventory which is released is either 0.1, as recommended by NUREG-0575 (Reference 43), or 0.3, as recommended by Regulatory Guide 1.25 (Reference 44), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in

the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

2. The release is short-term (i.e., assumed to last from 0 to 8 hours).

3. Short-term atmospheric dispersion factors were obtained from Regulatory Guide 1.4 (Reference 45), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", and assumes Class F stability, 1 m/sec wind speed, and ground-level release.

4. External dose conversion factors were obtained from Regulatory Guide 1.109 (Reference 46), "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I". The inhalation dose conversion factor for I-129 was taken from NUREG-0172 (Reference 47), "Age-Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake."

5. The distance from the release point to the receptor is 200 or 300 m.

10.2.3 Discussion of Results

The evaluation covers both normal operating and accident conditions.

10.2.3.1 Normal Operating Conditions

The dose to an off-site individual residing at a distance of 200 m from a filled NUHOMS-24P array is conservatively estimated as 110 mrem/yr. At 300 m, the dose is estimated as 32 mrem/yr. Since the assessment methodology conservatively assumed peak irradiated fuel, minimum post-irradiation time, full-time occupancy in the direction of maximum off-site dose, and no attenuation by building materials, it is likely that off-site doses to a "real" individual would be significantly lower, perhaps by a factor of five or more. Although site-specific factors (e.g., distance and direction of the nearest off-site residence, fuel conditions, contribution of off-site dose from reactor plant effluents, etc.) must be carefully considered, it is likely that normal operation of an NUHOMS-24P would comply with the requirements of 10 CFR 72.

10.2.3.2 Accident Conditions

The TR evaluated the dose to an off-site individual at several distances as a result of a loss of air outlet shielding block accident. Based on the TR evaluation, the dose to an individual at a distance of 200 m or 300 m is computed as approximately 1.3 mrem or 0.6 mrem, respectively, for a single HSM, or 19 mrem and 9 mrem for a 2x10 array of affected HSMs. These doses are well below the limits prescribed by 10 CFR 72.106 (b).

The following accident dose consequence results have been calculated for an offsite individual at distances of 200 m or 300 m. This assessment uses the method of Regulatory Guide 1.25 (i.e., 30% of fission gas inventory released), dispersion factors from Regulatory Guide 1.4, and dose factors from Regulatory Guide 1.109. For comparison, doses for a 10% fission gas release are also presented. These results are as follows:

Dose Equivalent (rem)

	10% Fission Gas Release		30% Fission Gas Release	
Organ	200 m	300 m	200 m	300 m
Whole Body	0.21	0.094	0.62	0.28
Skin	3.1	1.4	9.3	4.2
Thyroid	1.2	0.56	3.7	1.7

With the exception of skin dose at 200 m from 30% fission gas release, these doses are all within the 5 rem limit for whole body or any organ prescribed by 10 CFR 72.106(b). The DSC leakage event should be further assessed for site-specific applications. It should also be noted that, as indicated in the TR, 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 standard.

11.0 DECOMMISSIONING

11.1 SUMMARY AND CONCLUSIONS

The applicant has summarized decommissioning considerations for the NUHOMS-24P. The TR takes the position that the basic design of the NUHOMS-24P recognizes the need to decommission at the end of its useful life, and that a decommissioning plan would be developed based on site-specific factors. The TR also states that the DSC is designed to interface with a transportation system planned to transport canistered intact fuel assemblies (i.e., filled DSCs) to either a monitored retrieval storage facility (MRS) or a geologic repository. Once the DSCs have been removed, only small amounts of residual contamination are expected to remain in the HSM passages, thereby facilitating easy decommissioning. This position is based mainly on the fact that external contamination of the DSC is limited by its confinement features and through the contamination control procedures used during DSC fuel loading.

The staff finds that the proposed design and procedures are in conformance with the intent of 10 CFR 72.130, but withholds formal approval pending review of a site-specific case.

11.2 DESCRIPTION OF REVIEW

11.2.1 Applicable Parts of 10 CFR 72

Part 72.130 of 10 CFR provides criteria for decommissioning. It requires that considerations for decommissioning be included in the design of an ISFSI, and that provisions (1) facilitate the decontamination of structures and equipment, (2) minimize the quantity of radioactive wastes and contaminated equipment, and (3) facilitate removal of radioactive wastes and contaminated materials at the time of decommissioning.

Part 72.30 of 10 CFR defines the need for a decommissioning plan, which includes financing. Such a plan, however, is only appropriate to a site-specific situation, and 10 CFR 72.30 is therefore considered not applicable to this review.

11.2.2 Review Procedure

11.2.2.1 Design Description

The three primary components reviewed against 10 CFR 72.130 are the TC, the DSC and the HSM. Contamination levels on the external surface of the DSC are minimized by the use of uncontaminated water in the cask/DSC annulus during fuel pool loading operations. This prevents contaminated fuel pool water from contacting the DSC exterior.

Based on the proposed procedures described in Section 3.3.2.1, 4.4.1, and 5.1.1 of the TR, the contamination levels of the DSC will be determined by taking surface swipes of the upper one foot of the DSC exterior while the DSC is in the transfer cask prior to making the first closure weld. This swipe will be used as a representative sample of the DSC body. If the specified limits are exceeded, the annular space between the DSC and TC will be flushed with demineralized water until the contamination levels are within these limits. By minimizing DSC contamination, the potential for the internal surfaces of the HSM are kept to a minimum.

The current design of the 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.

A detailed decommissioning plan, which would consider such factors as decommissioning options available, likely further use of the site, environmental impact, and available waste transportation and disposal capabilities, would be developed on a site-specific basis.

11.2.2.2 Acceptance Criteria

Although 10 CFR 72.130 does not provide specific criteria for acceptance, the licensee is required to design the ISFSI for decommissioning. Therefore, the NUHOMS-24P design has been reviewed against good nuclear engineering practices which include (1) means to control the spread of contamination, and (2) a design that facilitates decontamination.

11.2.2.3 Review Method

Decommissioning considerations are addressed in a general manner in Section 3.5 of the TR. Other applicable descriptions in the TR include (1) Sections 3.3.2 and 4.2.3.1, which describe the confinement features of the DSC, (2) Sections 4.4.1 and 5.1.1, which describe methods used to limit external contamination of the DSC, and (3) Section 3.3.7.1, which provides guidelines for external surface contamination limits. These sections were reviewed to assess the adequacy of the proposed design in meeting the acceptance criteria.

11.2.2.4 Key Assumptions

It has been assumed for the purpose of this review that:

1. There is no credible chain of events that would cause the DSC confinement to fail, resulting in contamination of the HSM passages

2. Contamination of the external surfaces of the DSC and the internal surfaces of the HSM can be maintained below applicable surface contamination limits. The TR uses the following surface removable contamination limits as a guide:

Beta-gamma emitters:	10^{-4} uCi/cm^2
Alpha emitters:	10^{-5} uCi/cm^2

11.2.3 Discussion of Results

The material presented in Section 3.5 of the TR addresses decommissioning of the HSM. This section claims that the DSC is designed to interface with a transportation system capable of transporting intact canistered assemblies to either a monitored retrievable storage (MRS) facility or a geologic repository. However, no evidence to support this statement was provided in the TR. The NRC staff has not evaluated this aspect of the NUHOMS-24P system. For purposes of decommissioning, the NRC staff has assumed that the spent fuel must be removed from the DSC by a cutting operation. For personnel protection, see Section 8.3 of this SER.

When the fuel must be removed from the DSC, the internal surface of the DSC will be contaminated and may be slightly activated. After the interior is cleaned to remove loose contamination, the DSC can be disposed of as low-level waste, or possibly even as scrap. Decommissioning of the transfer cask is considered a site-specific issue.

The current design of the 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. A detailed decommissioning plan, which would consider such factors as decommissioning options available, likely further use of the site, environmental impact, and available waste transportation and disposal capabilities, would be developed on a site-specific basis.

The primary reason for requiring a clean exterior surface of the DSC is to reduce the total amount of activity available as a source of potential contamination for the HSM interior. The DSC surface contamination limits can be converted to a maximum total activity of roughly 15 microcuries of beta-gamma emitters, and 1.5 microcuries of alpha emitters. If the DSC exterior is initially below the contamination guidelines, contamination of the HSM interior will be much lower than these values.

The applicant has also claimed that failure of the DSC and release of radionuclides is not feasible under normal, off-normal, or accident conditions. Therefore, the contamination levels of the HSM are limited to levels which are much less than the initial DSC surface levels. This would probably allow the demolition and disposal of the HSM by conventional methods.

The staff concludes that adequate attention has been paid to decommissioning in the design of the NUHOMS-24P, considering the current state of knowledge. It will be necessary to review each site-specific application before determining whether demolition and removal of the HSM can be performed by conventional methods. The staff also notes that decommissioning of the DSCs, TC, and other equipment, as well as preparation of a decommissioning plan, are matters properly addressed in a site-specific application.

12.0 OPERATING CONTROLS AND LIMITS

12.1 SUMMARY AND CONCLUSIONS

Although operating controls and limits are normally reviewed as part of an application for a site-specific license, the staff has reviewed the set of generic operating controls and limits found in Chapter 10 of the TR. These controls and limits are summarized and expanded upon by the NRC staff in Table 12-1 of this SER. Operating controls and limits as stated in Table 12-1 of this SER are found acceptable.

12.2 DESCRIPTION OF REVIEW

12.2.1 Applicable Parts of 10 CFR 72

10 CFR 72.44 defines the requirements for operating limits and controls. That section only applies to specific licenses, not to reviews and approvals of topical reports. However, to the extent that operating controls and limits in a topical report are referenced in an application for a license, they require approval by the NRC.

12.2.2 Review Procedure

The staff has reviewed Sections 3, 7, 8, and 10 of the TR with special attention given to those parts which form the basis for a set of generic operating controls and limits. The criteria for and results of the safety analyses provided in the first three above-mentioned sections were used to review the limiting conditions proposed in Section 10 of the TR.

12.3 DISCUSSION OF RESULTS

Section 10.3 of the TR presents one fuel specification, nine limiting conditions for operation, one surveillance requirement, and two limiting conditions for transfer cask operation. The TR identifies these controls and limits as being generic and necessary for safe operation. The NRC staff has reviewed this set of operating controls and limits and added some additional conditions based on the overall evaluation of the TR. The

Table 12-1 Summary of Specifications from Section 10 of NUHOMS TR

Topic	Specification		TR Reference
Fuel Specifications (nominal)	Type	PWR Fuel	10.3.1.1 (3.1.1.1)
	Fuel Cladding	Zircaloy-clad fuel with no known or suspected cladding damage	10.3.1.1
	Burnup	≤ 40,000 MWd/MT	10.3.1.1 (3.1.1.1)
	Post Irradiation Time	≥ 10 years	10.3.1.1 (3.1.1.1)
	Initial (Beginning of Life) Enrichment	≤ 4.0 wt % U-235	10.3.1.1 (3.1.1.1)
	Initial Enrichment equivalent of stored assemblies	≤ 1.45 wt % U-235	3.3.4.1
	Weight Per Distance Between Any Adjacent Spacers, Per Assembly	≤ 109.00 kg	Table 3.1-2
	Distance Between Spacers	≤ 0.574 m	Table 3.1-2
	Maximum initial rod fill gas pressure	≤ 480 psig	8.2.9.1
	Any fuel not specifically filling the above requirements for burnup and post irradiation time may still be stored in the NUHOMS system, if all the following requirements are met:		
	Decay Power Per Assembly	≤ 0.66 kw at 10 years post irradiation time	
	Neutron Source Per DSC	≤ 3.715×10^9 n/sec/DSC, with spectrum bounded by Table 3.1-4	Table 3.1-1
	Gamma Source Per DSC	≤ 3.85×10^{16} MeV/sec/DSC with spectrum bounded by that shown in Table 3.1-4	Table 3.1-1

Table 12-1 Summary of Specifications from Section 10 of NUHOMS TR (cont'd)

Topic	Specification		TR Reference
DSC Vacuum Pressure During Drying	Vacuum Pressure Time at Pressure:	3 torr Not less than 30 minutes	10.3.2.1
DSC Helium Backfill Pressure	Helium backfill pressure (stable for 30 minutes filling)	2.5 psig ± 2.5 psig	10.3.2.2
DSC Moderator During Loading and Unloading	Boron concentration	≥ 1810 ppm	NRC 10-11 Feb. 1989, page 4
Time Limit to Complete DSC Draining After Removal From Spent Fuel Pool	Time	≤ 50 hours	NRC 10-11 Feb. 1989, page 3
DSC Helium Leakage Rate Test of Primary Weld	Leakage rate of primary weld 10^{-4} atm - cc/sec		10.3.2.3
DSC Dye Penetrant Test of Secondary Weld	Acceptance standards for liquid penetrant examination ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NB-5350 (1983) Liquid Penetrant Acceptance Standards		10.3.2.4
Dose Rate at End of DSC Lead Shield Plug	Dose Rates at the following locations:		10.3.2.5
	Center of Lead Shield Plug with water in cavity of DSC	100 mrem/hr	
	Center of DSC Top Cover Plate with Temporary Shield in place	200 mrem/hr	
Location of HSM	The HSM shall be located on a flood-free site.		SER Section 7.3.2
Surface Dose Rates on the HSM While the DSC is in Storage	Surface dose rates at the following locations:		10.3.2.6
	1) Outside of HSM door on centerline of DSC	100 mrem/hr	
	2) Center of air inlets	200 mrem/hr	
	3) Center of air outlet shielding caps	125 mrem/hr	
	4) Exterior side walls	20 mrem/hr	

Table 12-1 Summary of Specifications from Section 10 of NUHOMS TR (cont'd)

Topic	Specification		TR Reference
Maximum Air Temperature Rise from HSM Inlet to Outlet	Maximum air temperature 60^{0}F measured 24 hours after DSC emplacement.		10.3.2.7
Alignment of Cask and HSM for DSC Transfer Operation	The cask must be aligned with respect to the HSM so that the longitudinal centerline of the DSC in the cask is within $\pm$ 1/8" of its true position when the DSC rests on the HSM.		10.3.2.8
DSC Retrieval and Inspection	The DSC must be retrieved and inspected subsequent to any cask drop of 15 inches or greater.		3.3.4.2.3 of SER
Surveillance of the HSM Air Inlets and Outlets	Normal visual inspection	Every 24 hours	10.3.3.1
Maximum Surface Dose Rate on Transfer Cask	Transfer cask lid	$\leq$ 250 mrem/hr	10.3.4.1
	Body of TC with neutron shield filled with water	$\leq$ 250 mrem/hr	
	Bottom of transfer cask with bottom cover plate installed	$\leq$ 250 mrem/hr	
Transfer Route Selection	The surface within an eight foot proximity of the transfer trailer roadway shall be at the same elevation to ensure that the potential drop height of 80 inches is not exceeded.		10.3.4.2

complete list of operating controls and limits is given in Table 12-1 of this SER.

The requirements which were provided comprise a set of controls and limits for use with the proposed design. They will have to be augmented by additional specifications or revised to accommodate site-specific issues, but they do serve as a basis for review as a minimum set of requirements.

12.3.1 Fuel Specification

The fuel specification of Section 10.3.1.1 of the TR restricts the type of fuel acceptable for storage in the proposed design to ensure that peak fuel rod temperatures, radiation source terms, neutron multiplication factor, and stress on the DSC and its internals are below specified design limits.

12.3.2 Limiting Conditions for Operation

The nine limiting conditions for operation (LCO) are acceptable as proposed (see Section 10.3.2 of the TR).

12.3.3 Surveillance Requirements

The surveillance requirements are acceptable.

12.3.4 Limiting Conditions for Operation for TC Containing Fuel

The two limiting conditions for operation are acceptable as proposed in Section 10.3.4 of the TR.

13.0 QUALITY ASSURANCE

13.1 DISCUSSION

NUTECH's quality assurance (QA) program is addressed in the NUHOMS-24P topical report by incorporation of Section 11 of Reference 22. The QA program describes how NUTECH ensures the quality of the ISFSI described in the topical report. NUTECH is expected to be responsible for final design, specifications, procurement, fabrication, assembly, delivery, and preoperational testing associated with the dry storage canister and transfer cask. NUTECH may also have responsibility as consultant, supplier, installer, and/or on-site engineer for the horizontal storage module. The staff reviewed the QA program description against the acceptance criteria in Reference 48, the "Standard Review Plan for Quality Assurance Programs for an Independent Spent Fuel Storage Installation (ISFSI)."

The staff found that NUTECH's QA program described in the topical report adequately addresses the QA functions appropriate for these responsibilities, that the commitments meet the requirements of Subpart G of 10 CFR Part 72, and that the QA program is acceptable. The topical report can be referenced without further QA review in a license application to receive and store spent fuel under 10 CFR Part 72, provided the applicant applies its NRC-approved QA program that meets the requirements of Appendix B to 10 CFR Part 50 to the design, construction, and use of the spent fuel storage installation.

13.2 CONCLUSION

The staff concludes that the QA program described in the NUHOMS-24P topical report is acceptable as an appropriate reference for partial fulfillment of QA program requirements for ISFSI license applications.

14.0 SUMMARY EVALUATION

14.1 SUITABILITY AS REFERENCE

The NRC staff finds the TR to be suitable as a reference for ISFSI license applications except as specifically noted in Table 14.1. This evaluation is based on the extent of compliance with 10 CFR 72 using guidance as provided in NRC Regulatory Guides 3.48 and 3.60.

The NRC staff approves use of the TR as reference or for direct incorporation in license application documents with the exceptions noted in Table 14.1. Any modifications beyond those specified in Table 14.1 in material taken from or referenced in the TR, or any use of material in the TR by incorporation or reference which is outside the limitations, assumptions, conditions, or other context as stated in the TR or this SER requires full explanation, calculations, and/or descriptions, and is subject to further review by the NRC.

14.2 SATISFACTION OF LICENSE APPLICATION REQUIREMENTS

The NRC staff finds that the TR can partially fulfill ISFSI license application documentation requirements stated in 10 CFR 72 when used by clear and specific reference.

TABLE 14.1 EXCEPTIONS TO USE OF TR AS REFERENCE IN LICENSE APPLICATION DOCUMENTATION

License Application Documentation	10 CFR Reference	Subject	Limitations on Use of TR Sections as References
License Application	72.22	General & Financial Info.	Not a reference
Safety Analysis Report	72.24(a)	Site Description & Safety Assessment	Background only
	72.24(b)	Description & Discussion of Structures	
		- Design	For DSC, TC, HSM (less Found. Anal) only
		- Operating Characteristics	For DSC, TC, HSM only
		- Unusual/Novel Design Features	Not a reference
		- Principal Safety Considerations	DSC, TC, HSM (less Found. Anal) only
	72.24(c)	Design Sufficient to Support 72.31 findings	For DSC, TC, HSM (less Found. Anal) only, except as indicated below.
	72.122(b)(1)	Design for compatibility w/site characteristics & environment	Not complete. SAR to validate that satis. for site & environment.
	72.122(b)(2)	Design for Natural Phenomena for site.	Not complete. SAR to validate that site phenom. within design envelope.
	72.122(b)(3)	Capability to determine intensity of Natural Phenomena	Not a reference
	72.122(c)	Protection vs Fires & Explosion	Background only
	72.122(d)	Sharing of Structures, Systems & Components	Background only
	72.122(e)	Proximity of Sites	Background only
	/ 72.122(f) \ \ 72.44(c)(3) /	Testing & Installation of Systems & Components	Background only. Satisfaction of surveillance requirements left to SAR.
	72.122(g)	Emergency Capability	Background only
	72.122(h)(3)	Ventilation and off-gas systems	Background only
	/ 72.122(i) \ \ 72.44(c)(3) /	Instrumentation & Control systems	Background only. Satisfaction of surveillance requirements left to SAR.
	72.122(j)	Control room or area	Not a reference
	72.122(k)	Utility services	Background only
	72.126(a)	Exposure control	Background only
	72.126(b)	Rad alarm systems	Background only
	72.126(c)	Effluent & Direct Rad Monitoring	Background only
	72.126(d)	Effluent Control	Background only
	72.128(b)	Waste Treatment	Background only
	72.130	Criteria for Decommissioning	Background only
	72.24(c)(3)	Applicable Codes and Standards	Not a reference

License Application Documentation	10 CFR Reference	Subject	Limitations on Use of TR Sections as References
SAR (cont'd)	72.24(d)	Structures, Systems & Components Important to Safety	For DSC, TC, HSM (less Found. Anal)
	72.24(e)	Satisfaction of 10 CFR 20 occupational radiation exposure limits	Background only
	72.24(f)	Design & Operating mode features to maintain low waste volume	Background only
	72.24(g)	Probable License Conditions and Tech Specs	Design background only
	72.24(h)	Plan for Conduct of Operations	Not a reference
	72.24(i)	Resolution of Safety Questions	Background only
	72.24(j)	Applicant's Tech Qualifications	Not a reference
	72.24(k)	Description of Emergency Plans	Not a reference
	72.24(l)	Equipment to Maintain Control over Gas and Liquid Effluent	Not a reference
	72.24(m)	Individual dose/dose commitment outside controlled area	Not complete. SAR to address actual site conditions.
	72.24(n)	Description of QA program	Not complete. SAR to address or integrate w/applicant QA program.
	72.24(o)	Description of Detailed Security Measures for Physical Protection	Not a reference
	72.24(p)	Descr of Program for Preoperational testing & initial operations	Not complete. SAR to address or integrate w/applicant's program.
	72.24(q)	Description of Decommissioning Plan	Background only
Decommissioning Plan	72.30	Decommissioning Plan	Background only
Emergency Plan	72.32	Emergency Plan	Background only
Environmental Report	72.34	Environmental Report	Background only
Quality Assurance Program	72. Subpart G	Quality Assurance Program	Not complete. SAR to address or integrate w/applicant QA program.
Physical Security Plan	72.180	Physical Security Plan	Not a reference
Design for Physical Protection	72.182	Design for Physical Protection	Not a reference
Safeguards Contingency Plan	72.184	Safeguards Contingency Plan	Not a reference

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