SAFETY EVALUATION REPORT
FOR THE
PRAIRIE ISLAND
INDEPENDENT SPENT FUEL STORAGE INSTALLATION

SPECIAL NUCLEAR MATERIAL LICENSE NO. 2506

LICENSE AMENDMENT REQUEST

DOCKET NO. 72-10

U. S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
August 2010

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Docket No. 72-10

SPECIAL MATERIAL LICENSE NO. 2506
AMENDMENT REQUEST TO MODIFY THE TN-40 CASK DESIGN FOR USE
AT THE PRAIRIE ISLAND INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)
DESIGNATED: TN-40HT

BACKGROUND – SUMMARY

By application dated March 28, 2008 (ML081400652 – package, ADAMS Accession Number), as supplemented June 26, 2008 (ML081830521), August 29, 2008 (ML082421114), June 26, 2009 (ML091910507), September 28, 2009 (ML092720211), January 18, 2010 (ML100210197), May 4, 2010 (ML101270112), and July 27, 2010 (ML102090138), Nuclear Management Company, LLC (NMC)[1] requested approval of an amendment to Special Nuclear Material (SNM) License No. 2506 and the license Technical Specifications (TS) for the Prairie Island ISFSI in accordance with 10 CFR Part 72. The applicant proposes in this license amendment request (LAR) to modify the TN-40 cask for storage of higher initially enriched and higher burnup fuel and to reformat the TS.

The proposed new cask design, the TN-40HT, is similar to the TN-40 design with the exception of the following: 1) dose rate limits; 2) criteria for fuel selection; 3) helium backfill pressures; 4) difference in lid-bolt torque; and 5) fuel basket design. The fuel basket design is similar to the TN-68. The applicant asserts that the differences in the design modifications would minimally impact the existing fuel loading and transfer operations.

Requests for additional information (RAIs) were issued on March 24, 2009 (ML090840020); with two enclosures: non-proprietary (ML090840028) and proprietary (ML090840038). The applicant requested an audience with the NRC after receipt of RAIs. Public meetings with the applicant occurred on May 7, 2009 and June 22, 2009; ADAMS accession numbers for the meeting summaries are ML091670052 and ML092030400, respectively. Responses to the RAIs (ML091910507) elicited a second RAI (ML093310291, dated November 25, 2009). Inadequate and/or insufficient responses received by the NRC (ML100210197) prompted the decision to suspend the technical review. Staff engaged in telephone conferences with the applicant to clarify the three outstanding issues: 1) equivalency for non-ASTM materials used in the TN-40HT cask; 2) adoption of NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," classification criteria; and 3) leak testing the entire confinement boundary (including the base metal) during fabrication. ADAMS accession numbers for the conversation records are ML100290300, dated January 27, 2010; ML100470801, dated February 1, 2010; and ML100920077, dated April 1, 2010. The applicant requested a public meeting. The meeting was noticed on March 16, 2010 (ML100750740) and held on April 20. The meeting summary (ML101160130) outlines the approach the applicant proposed to resolve the outstanding issues. The applicant submitted RAI responses on May 4, 2010 (ML101270112). As a courtesy, the preliminary safety evaluation report (SER, this document) and draft copies of the proposed license, technical specifications, and notice of issuance were transmitted electronically to the licensee for editorial review and comment on June 6, 2010. Subsequent to the follow-up conference call (ML101650445), NRC review was put on hold to permit extension of NSPM's

[1] On September 22, 2008, NMC transferred its operating authority to Northern States Power Company, a Minnesota corporation (NSPM), doing business as Xcel Energy. By letter dated September 3, 2008 (package, ML082240762), NSPM assumed responsibility for actions and commitments previously submitted by NMC.

review period. To clarify licensing aspects of the pending licensing amendment the licensee and NRC staff again engaged in dialogue (ML101810010).

Other licensing activities performed to process this LAR included the issuance of a finding of no significant impact (FONSI) and the notice of docketing for the LAR. The Notice of Availability of Environmental Assessment and FONSI for the Prairie Island ISFSI was published in the *Federal Register* on December 4, 2009 (74 FR 63798). Issuance of Federal Register Notice – Notice of Docketing for Amendment to Materials License No. SNM-2506 for the Prairie Island ISFSI was published in the *Federal Register* on May 4, 2010 (75 FR 23820). The NRC staff evaluated the LAR for SNM License No. 2506 for the TN-40HT and documented the security assessment review separately, as it contains sensitive information that cannot be made publicly available. The security assessment should be reviewed prior to approval of any amendment to this application.

1 GENERAL DESCRIPTION

By application dated March 28, 2008, as supplemented June 26, and August 29, 2008; June 26, and September 28, 2009; January 18, May 4, and July 27, 2010, pursuant to 10 CFR 72.56, the Nuclear Management Company, LLC (NMC) requested approval of an amendment to Special Nuclear Material License No. 2506 and the license Technical Specifications (TS) for the Prairie Island ISFSI. This amendment proposes to modify the TN-40 cask for storage of higher initial enrichment and higher burnup fuel and to reformat the TS. TN-40HT is the designation of this modified cask.

1.1 Background

The current Prairie Island ISFSI TS limit the fuel that may be stored in a TN-40 cask to fuel that had an initial enrichment of ≤ 3.85 wt% U-235 and a burnup of ≤ 45 GWd/MTU. However, since 1990 the Prairie Island Nuclear Generating Plant (PINGP) has been operating with fuel that has had an initial enrichment greater than 3.85 wt% U-235 and has burned the fuel to burnup values greater than the 45 GWd/MTU limit.

To accommodate dry storage of fuel with higher initial enrichment and the increased decay heat associated with the increase in burnup, the TN-40 cask design has been modified. The modifications include an improved heat transfer basket design that relies upon slotted aluminum and neutron poison plates, thicker stainless steel fuel compartment walls, stainless steel support bars between the fuel compartment walls, and an increase in the neutron poison loading. The construction and materials of the modified basket are very similar to the previously approved basket design in 10 CFR Part 72.214, Certificate Number: 1027, "TN-68 dry cask." To maintain the loaded cask weight within the Auxiliary Building Crane capacity rating, the increase in weight due to the basket modifications was offset by reducing the thickness of the radial gamma shield, bottom gamma shield, and lid outer plate. To reduce the neutron dose associated with the increase in burnup, the thickness of the radial neutron shield was increased. The proposed changes to the TN-40 cask design are such that the TN-40HT cask will use the existing equipment for lifting, loading, and transporting the cask.

To improve plant operator usage of the Technical Specifications, this amendment request reformats the Technical Specifications to the format adopted for the PINGP Technical Specifications.

1.2 TN-40HT Dry Cask Description

The TN-40HT cask is very similar in design to the TN-40 cask. The major difference is that the TN-40HT cask is designed to store higher initial enrichment and higher burnup fuel. In order to accomplish this, the heat transfer capability of the basket design was enhanced. Additionally, to accommodate the enhanced basket, some minor changes were made to the cask body. The TN-40HT cask employs a slightly thinner lid, shield shell and cask bottom shield. However, the radial neutron shield thickness is increased to offset the higher neutron source of the high burnup fuel.

The TN-40HT cask accommodates up to 40, 14x14 pressurized water reactor (PWR) fuel assemblies with or without fuel inserts. It consists of the following components:

- A basket assembly which locates and supports the fuel assemblies, transfers heat to the cask body wall, and provides neutron absorption to satisfy nuclear criticality requirements.

- A containment vessel including a bolted closure lid and seals which provides radioactive material confinement and a cavity with an inert gas atmosphere.
- Gamma shielding surrounding the containment vessel.
- Radial neutron shielding surrounding the shield shell which provides additional radiation shielding. This neutron shielding is enclosed in an outer steel shell.
- A top neutron shield which is attached to the outer surface of the cask lid and provides additional neutron shielding.
- An overpressure system which monitors and maintains the pressure between the cask closure seals and provides a positive pressure differential across the inner seals so that any inner seal leak will result in in-leakage to the cask cavity.
- A protective cover which provides weather protection for the closure lid, top neutron shield and overpressure system.
- Sets of upper and lower trunnions which provide the means for lifting and rotating the cask.

The maximum allowable initial enrichment of the fuel to be stored in a TN-40HT cask is 5.0 wt% U-235. The maximum bundle average burnup, maximum decay heat, and minimum cooling time are 60 GWd/MTU, 0.80 kW/assembly, and 12 years, respectively. The cask is designed for a maximum heat load of 32 kW.

Section A1.3.2 of the applicant's SAR provides TN-40HT cask characteristics.

1.3 Proposed Technical Specifications Changes

This LAR proposes to revise the Prairie Island ISFSI Technical Specifications (TS) to incorporate operating restrictions associated with the TN-40HT dry cask system, e.g., fuel enrichment and burnup limits. In addition, the proposed changes include changing the format of the TS to be consistent with the current format adopted for the PINGP TS. As part of the reformatting effort and where practical, NUREG-1745 (Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance, ML011940387) was incorporated. Where the current licensing basis requirements can safely be applied to operation of a TN-40HT dry cask system, the proposed TS do not change the technical requirements. However, where operation of the TN-40HT system requires a more restrictive requirement and the more restrictive requirement will not unduly affect operations of a TN-40 cask system, the proposed TS are written such that the more restrictive requirement will apply to both the TN-40 and the TN-40HT dry cask systems. For those few circumstances where it is not possible to have a single requirement that is appropriate for both a TN-40 and TN-40HT system, the proposed TS are written such that the requirement for each cask design system is clear.

1.3.1 Applicable Regulatory Requirements/Criteria

Table 1 shows the regulations affected by the proposed modification as identified by the applicant. The evaluation findings in this SER do address these regulations and any applicable 10 CFR Part 72 section(s) and subsection(s).

Table 1 - Regulatory Safety

Regulation	Title	Affected by this Amendment
72.120	General consideration	No
72.122	Overall requirements	Yes
72.124	Criteria for nuclear criticality safety	Yes
72.126	Criteria for radiological protection	Yes
72.128	Criteria for spent fuel, high-level radioactive waste, and other radioactive waste storage and handling	Yes
72.130	Criteria for decommissioning	No

1.4 Identification of Agents and Contractors

Section A1.4 of the applicant's SAR it is stated that the agents and contractors associated with the TN-40HT casks are the same as those described in Section 1.4 for the TN-40 casks.

1.5 EVALUATION FINDINGS

The documentation submitted with the application fully supports positive findings for each of the regulatory requirements.

F1.1 The staff concludes that the information presented in this section of the applicant's SAR satisfies the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered the Regulation itself; Regulatory Guide 3.48 and accepted practices.

F1.2 Agents and contractors responsible for the design, construction, and operation of the installation have been identified.

2 SITE CHARACTERISTICS

The information that is independent of cask design include – geography and demography of site selected (site location, site description, population distribution and trends, and uses of nearby land and waters); nearby industrial, transportation and military facilities; meteorology (regional climatology, local meteorology – data source and topography; onsite meteorological measurement program; diffusion estimates – basis and calculations; hydrology – surface water and ground water; geology (basic geologic and seismic information – storage site geomorphology , geologic history of storage site and surrounding region, specific structural features of significance, large scale geologic map, plot plan and site investigations, geologic profiles, plan and profile drawings, local geologic features affecting site location, site groundwater conditions, geophysical surveys and studies, soil and rock properties, and analysis techniques); vibrating ground motion; surface faulting, stability of subsurface materials; and slope stability. Therefore, there was no change in the information provided.

2.1 EVALUATION FINDINGS

There are no evaluation findings; site characteristics were unaffected by the changes proposed in this amendment request.

3 OPERATION SYSTEMS

The review was oriented on functions and the compatibility of proposed systems with performance of those functions. The NRC did not, for the purposes of this license amendment request, review procedures. The review of the descriptions of functions constitutes the principal basis for assessing the assurance provided by the submitted documentation.

3.1 General Operating Functions

The fuel assemblies will be stored unconsolidated and dry in sealed storage casks. The casks will rest on a reinforced concrete pad, and provide safe storage by ensuring a reliable decay heat path from the spent fuel to the environment and by providing appropriate shielding and confinement of the fission product inventory. Storage of spent fuel in storage casks is a totally passive function, with no active systems required to function. Cooling of the casks is accomplished by radiant and convective cooling.

Each cask will be handled with a lifting yoke, the 125 ton capacity Auxiliary Building crane, a transport vehicle, or other appropriate equipment. The crane will lift the cask from the spent fuel pool, in the spent fuel pool enclosure, move the cask laterally through an access door, and lower the cask to ground level in the rail bay of the Auxiliary Building. The cask will then be picked up by the transport vehicle which will be pulled to the ISFSI by a tow vehicle. After the transport vehicle has been maneuvered to locate the cask in its storage position, the cask will be set down.

All the handling equipment to be used outside the Auxiliary Building will be designed according to appropriate commercial codes and standards, and will be operated, maintained, and inspected in accordance with the supplier’s recommendations. Documentation will be maintained to substantiate conformance with all applicable standards.

3.2 Spent Fuel and High-Level Waste Handling Systems

The handling of spent fuel within the Prairie Island Nuclear Generating Plant will be conducted in accordance with existing fuel handling procedures. Only undamaged fuel will be considered for storage in the TN-40HT casks.

In the TN-40HT casks, a DAMAGED FUEL ASSEMBLY is a Spent Nuclear Fuel Assembly that:

a. has visible deformation of the rods in the spent nuclear fuel assembly. Note: This is not referring to the uniform bowing that occurs in the reactor. This refers to bowing that significantly opens up the lattice spacing;
b. has individual fuel rods missing from the assembly. Note: The assembly is not a DAMAGED FUEL ASSEMBLY if a dummy rod that displaces a volume equal to, or greater than, the original fuel rod, is placed in the empty rod location;
c. has missing, displaced, or damaged structural components such that radiological and/or criticality safety is adversely affected (e.g., significantly changed rod pitch);
d. has missing, displaced, or damaged structural components such that the assembly cannot be handled by normal means (i.e., crane and grapple);
e. has reactor operating records (or other records) indicating that the spent nuclear fuel assembly contains cladding breaches; or
f. is no longer in the form of an intact fuel bundle (e.g., consists of, or contains, debris such as loose fuel pellets or rod segments).

Handling of the sealed casks outside of the Auxiliary Building in the process of emplacing them at the ISFSI will be done according to procedures that ensure that their safety functions and the power station capability for safe shutdown are not impaired. These operations for the TN-40HT casks are the same as for a TN-40 cask and are described in Section 5.4 (Spent Fuel Transport to ISFSI) of the SAR.

3.3 Other Operating Systems

Several systems are declared as independent of cask design in the Operating Systems section (A4.4) of the SAR. The systems included are: (1) loading and unloading systems, which includes function; major components and operating characteristics; and safety consideration and controls; (2) decontamination system; and (3) other systems (electrical systems; alarm system; fire protection system; and vacuum systems).

3.3.1 Storage System Operations

The TN-40HT was designed to use the same equipment as the TN-40 cask. As a consequence of this design objective, the operation of the TN-40 cask, as described in Section 5 of the SAR, is applicable to the operation of the TN-40HT cask.

The narrative of the storage system operation in Section 5.1.1 of the SAR is applicable to the operation of the TN-40HT casks. The information outlined in Section 5.1.2 is applicable to the TN-40HT casks except for the location of the radiation exposure determination for the TN-40HT cask which is located in Section A7 of the SAR. The information in Section 5.1.3.1 (Criticality Prevention) is applicable to the TN-40HT casks except for the location of the criticality discussion which located in Section A3.3.4. The information in Section 5.1.3.2 is applicable to the TN-40HT casks except for the location of the description of the transmitters which is located in Section A3.3.3. The information in Section 5.1.3.3, maintenance techniques, is applicable to the TN-40HT casks.

3.3.2 Operation Support Systems

The information in the section on spent fuel accountability program (Section 5.3); transport to ISFSI (Section 5.4); and transfer to transport cask are applicable to the TN-40HT casks.

3.3.3 Control Room and Control Area

The information in Section 5.2 of the SAR is applicable to the TN-40HT casks.

3.3.4 Analytical Sampling

The information in Chapter 7 of the SAR is applicable to the TN-40HT casks. The analytical sampling operation specifies the types of samples and rate of sampling that are appropriate for the condition being monitored. Provisions for obtaining samples during off-normal conditions to ensure that prescribed limits have not been exceeded are specified. The SAR describes the facilities and equipment that will be available to perform the analyses. Disposition of laboratory wastes is also described.

3.3.5 Shipping Cask Repair and Maintenance

Storage cask repair and maintenance performance is described in Section A5.1.3.3 of the SAR. For utility supplies and systems, the TN-40HT storage casks are passive devices – no utility services are needed for operation of the casks.

3.4 EVALUATION FINDINGS

F3.1 The SAR includes acceptable descriptions and discussions of the projected operating characteristics and safety considerations, in compliance with 10 CFR 72.24(b).

F3.2 The SAR provides reasonable assurance that the activities to be authorized by the license can be conducted without endangering the health and safety of the public and will be in compliance with the applicable regulations of 10 CFR 72.40(a)(13).

F3.3 The ISFSI is to be located on a site with existing facilities suitable and available for control of ISFSI operations under off-normal or accident conditions, whose use will not interfere with other operations on the site important to safety, in compliance with 10 CFR 72.40(a)(3) and 72.122(j).

4 SSC AND DESIGN CRITERIA EVALUATION

The objective of this part of the review is to ensure that the applicant acceptably defines: (1) the limiting characteristics of the spent fuel or other high-level radioactive waste materials to be stored, (2) the classification of structures, systems and components (SSCs) according to their importance to safety, and (3) the design criteria and design bases, including the external conditions during normal and off-normal operations, accident conditions, and natural phenomena events.

4.1 Materials to be Stored

The TN-40HT cask is designed to store 40 Westinghouse and Exxon 14x14 Pressurized Water Reactor (PWR) spent fuel assemblies with or without fuel inserts (thimble plug devices, TPDs or burnable poison rod assemblies, BPRAs). The maximum allowable initial enrichment is 5.0 wt% U-235. The maximum bundle average burnup, maximum decay heat, and minimum cooling time for the fuel assembly are 60 GWd/MTU, 0.80 kW/assembly (including heat from inserts), and 12 years, respectively. The cask is designed for a maximum heat load of 32 kW.

Reconstituted assemblies, (uranium, inert, or stainless steel rods replacing fuel rods), may also be stored in the cask. The decay heat of a reconstituted assembly with stainless steel rods is bounded by an intact assembly. However, irradiated stainless steel rods increase the gamma source term for a period of time after irradiation. This period is shorter than the 12 year minimum cooling time required and thus no additional cooling time is required for these reconstituted assemblies.

4.1.1 Spent Fuel

Section A3 of the applicant's SAR describes the necessary spent nuclear fuel characteristics important to the design and analytical calculations and acceptance tests. The analytical calculations include nuclear criticality safety, heat removal, shielding, etc. Fuel characteristics include reactor type, fuel configuration and vendor, enrichment, dimensions, weight, burnup, cooling time, type of cladding, assemblies to be stored per confinement vessel or pool facility, decay heat, fuel pin gas volume and temperature, condition (i.e., intact, undamaged), presence of control components, or other radioactive materials associated with fuel assemblies, and physical form of radionuclides.

4.2 Classification of Structures, Systems, and Components

Table A4.5-1 of the applicant's SAR lists the structures, systems, and components (SSCs). This information is presented below in Table 2; the SSCs are classified into two broad categories: important to safety or not important to safety according to NUREG/CR-6407 "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety." Section A4.5 of the SAR discusses important to safety components further categorized into Category A, B, or C (also according to NUREG/CR-6407). All confinement boundary components are classified as important to safety Category A. The gamma and neutron shielding are classified as important to safety Category B. The NRC review involves both important to safety or not important to safety categories; however, SSCs important to safety are reviewed in greater depth. Acceptance criteria for classification of SSCs important to safety are discussed in 10 CFR 72.3, 10 CFR 72.24(n), 10 CFR 72.144 (a) and (c).

Table 2 - Classification of TN-40HT Major Components (Table A4.5-1)

IMPORTANT TO SAFETY	NOT IMPORTANT TO SAFETY
Containment vessel including lid, flange, inner containment shell & bottom containment plate	Pressure monitoring system, & overpressure cover
Lid bolts	Protective cover, bolts, & seal
Lid vent and drain covers, & bolts	Paint on exterior of cask
Basket assembly including fuel compartments, poison plates, & structural plates	
Trunnions	
Basket rails	
Lid, vent & drain seals	
Radial neutron shield	
Cask body shield shell	
Cask body bottom	
Lid shield plate	
Top neutron shield including bolts	
Outer shell	

4.3 Design Criteria for SSCs Important to Safety

4.3.1 General

The regulatory requirements for design bases and general design criteria are given in 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.106 (a) and (c); 10 CFR 120 (a) and (b); 10 CFR 122 (a) through (l); 10 CFR 72.144; and 10 CFR 72.182 (a), and (b). The applicant identified design criteria and design bases for all SSCs determined to be important to safety. The basic design criteria for SSCs which are important to safety shall: maintain subcriticality, maintain confinement, ensure radiation rates and doses for workers and public do not exceed acceptable levels and remain as low as is reasonably achievable (ALARA), maintain retrievability, and provide for heat removal (as necessary to meet the above criteria). Acceptance criteria for the specific design criteria are discussed in detail in the chapters indicated for the specified discipline.

TN-40HT SAR Section A3 provides the Principal Cask Design Criteria. The following TN-40HT SAR sections identify design criteria and design bases for SSCs determined to be important to safety for the specific functions described below:

4.3.2 Structural

The sections in the applicant's SAR that identify the design bases and general design criteria for the structural evaluation are: "A3.1.2 General Operating Functions," "A3.2 Design Criteria for

Environmental Conditions and Natural Phenomena," "A3.4 Summary of Storage Cask Design Criteria," and "A3.5 ASME Code Alternatives."

4.3.3 Thermal

The section in the applicant's SAR that identifies the design bases and general design criteria for the thermal evaluation is A3.3.2.2, entitled "Heat Transfer Design."

4.3.4 Shielding

The section in the applicant's SAR that identifies the design bases and general design criteria for the shielding evaluation is Appendix A7A, entitled "TN-40HT Cask Dose Analysis."

4.3.5 Confinement

The section in the applicant's SAR that identifies the design bases and general design criteria for the confinement evaluation is A3.3.2.1, entitled "Confinement Barriers and Systems."

4.3.6 Criticality

The section in the applicant's SAR that identifies the design bases and general design criteria for the criticality evaluation is A3.3.4, entitled "Nuclear Criticality Safety."

4.3.7 Decommissioning Considerations

The applicant specifies in Table 1, entitled "Criteria Sections Affected by the TN-40 Design Modifications" of Enclosure 3 (L-PI-08-020), that the regulatory requirement for satisfying the criteria for decommissioning (10 CFR 72.130) was unaffected by the changes proposed in this amendment request.

4.3.8 Retrieval Capability

Retrieval capability is not specifically discussed in the SAR; but it is an inherent element of basket design, the loading procedures, and administrative controls (Technical Specifications) which ensure that neither the fuel assembly nor the basket get deformed or oxidized to such a degree (during Loading or Storage conditions) that a fuel assembly could not be reasonably retrieved.

4.3.9 Design Criteria for Other SSCs

Design criteria for other SSCs (i.e., those Not Important to Safety) are discussed in various sections of the SAR, including Section A3, entitled "Principal TN-40HT Cask Design Criteria." For instance, the design criteria for the protective weather cover are provided in SAR Section A3.2.4.

4.4 EVALUATION FINDINGS

Evaluation findings were prepared by the staff upon completion of the SAR review. The regulatory requirements identified in Section 4.3 and staff safety concerns have been properly addressed and factored into the design. The documentation submitted with the application fully supports positive findings for each of the regulatory requirements. The findings are as follows:

F4.1 The SAR and docketed materials adequately identify and characterize the spent fuel to be stored at the site in conformance with the requirements given in 10 CFR 72.2 (a)(1) and (a)(2), and 10 CFR 72.6 (b). The form of the spent fuel is acceptable if the fuel is solid fuel and not in liquid form, and meets the requirements given in 10 CFR 72.120 (b).

F4.2 The SAR and docketed materials adequately identify and characterize the high level radioactive waste as required by 10 CFR 72.3. The waste form is solid and not liquid as required by 10 CFR 72.120 (b).

F4.3 The structure, systems and components have been classified according to their function as important to safety or not important to safety, and meet the requirements given in 10 CFR 72.3 and 10 CFR 72.24 (n).

F4.4 The SAR and the docketed materials relating to the design bases and criteria meet the general requirements as given in 10 CFR 72.24 (c)(1), (c)(2), (c)(4), and (n); 10 CFR 122 (a), (b), (c), (d), (e), (f), (g), (h), (i), (j), (k), and (l); 10 CFR 72.144; and 10 CFR 72.182 (a), and (b).

F4.5 The SAR and docketed materials relating to design criteria for confinement SSCs, including applicable codes and standards, meet the requirements of 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.122 (a), (b), and (l); 10 CFR 72.128 (a); and 10 CFR 72.236 (b) and (e). Additionally, the confinement SSC design criteria meet the guidance provided in applicable parts of Regulatory Guides for protection against seismic and tornado events.

F4.6 The SAR and docketed materials meet the regulatory requirements for design bases and criteria for thermal consideration as given in 10 CFR 72.122 (a), (b)(1), (b)(2) and (b)(3), (c), (d), (f), (g), (h), and (i); and 10 CFR 72.128 (a)(4). The SAR meets the regulatory requirements for design criteria for fire protection given in Regulatory Guide 1.120.

F4.7 The SAR and docketed materials relating to the design bases and criteria for shielding and radiation protection meet the regulatory requirements in 10 CFR 72.24 (c)(1), (c)(2), and (c)(4);10 CFR 72.104 (a), (b), and (c); 10 CFR 72.104 (a) and (b); 10 CFR 72.126 (a)(1), (a)(5).

F4.8 The SAR and docketed materials relating to the design bases and criteria for criticality safety meet the regulatory requirements as given in 10 CFR 72.124 (a), (b), and (c).

F4.9 The SAR and docketed materials relating to the design bases and criteria for other SSCs not important to safety, but subject to NRC approval, meet the general regulatory requirements as given in 10 CFR 72.24 (a), (b), (c), (d), (e), (f), (g), (h), (l) and the appropriate requirements as given in Subparts E and F of 10 CFR 72.

5 INSTALLATION AND STRUCTURAL EVALUATION

The license amendment request introduces the TN-40HT cask as a modified version of the TN-40 cask system. The applicant notes that the criteria applicable to the facility are not affected by the request. It further states that the cask modifications and this amendment request do not change the loading and handling operations described in Section 5 of the Prairie Island ISFSI Safety Analysis Report. As a result, the installation and structural evaluation consider only the TN-40HT cask confinement structures, systems, and components (SSCs) including the containment vessel and fuel basket assembly. Other important to safety SSCs, such as trunnions, and SSCs not important to safety but subject to NRC approval are also considered. The important to safety reinforced concrete structures, such as storage pads, and other SSCs associated with the Prairie Island ISFSI are not re-evaluated, as their designs for the TN-40 casks continue to remain applicable for the TN-40HT cask system.

In the following the staff evaluates the structural performance of the TN-40HT dry cask storage system under normal, off-normal, and design basis accident conditions, and loads associated with environmental conditions and natural phenomena for meeting the requirements of 10 CFR Part 72.122, "Overall Requirements," and 10 CFR Part 72.128, "Criteria for Spent Fuel, High-Level Radioactive Waste, Reactor-Related Greater than Class C Waste, and Other Radioactive Waste Storage and Handling." The evaluation focuses primarily on effects of limited changes in design features and design-basis loadings on resulting structural margins for demonstrating acceptability of the cask system. This is in recognition that, with exception to the fuel basket, the cask structural details and corresponding design basis and acceptance criteria are essentially identical to those of the approved TN-40.

5.1 Structural Design Features and Design Criteria

5.1.1 Design Features

Section A1.3 of the SAR provides a general description of the TN-40HT system, which, as a modified version of the TN-40 dry cask, consists of two major structural components: the cask body and the fuel basket.

Drawings TN40HT-72-1 through -72-10 provide cask body design features. Table A1.3-1 lists major dimensions of the cask. With a slightly shorter overall height and varied thickness for the lid and bottom shield plates and radial gamma and neutron shield shells, all major dimensions for TN-40HT are identical to those of TN-40. This includes an identical 72-inch diameter by 163-inch high cask cavity lined up with a 1-1/2-inch thick cask inner shell as confinement boundary. The loaded cask weight is 242,400 lbs, which is about 0.7% heavier than the TN-40 at 240,700 lbs.

Drawing TN40HT-72-21 depicts design details of the fuel basket consisting of an assembly of 40 rectangular stainless steel tubes joined, through fusion welds, by a grillage of 1.75-inch wide by either 7/16-inch or 5/16-inch deep stainless steel bars. Sandwiched between the tube walls and above and below the bars are slotted neutron poison plates and aluminum plates assembled in an egg-crate construction. The neutron absorber plates, as non-structural components, provide criticality control and the aluminum plates, which work as intervening structures to resist the out-of-plane deformation of fuel tube walls, also provide the heat conduction paths from the fuel assemblies to the cask cavity wall.

5.1.2 Design Bases

Subpart F of 10 CFR Part 72 sets forth general design criteria for the design, fabrication, construction, testing, and performance of structures, systems, and components (SSCs) important to safety in an ISFSI. Section A3.2 of the SAR presents the design basis loading conditions imposed on the TN-40HT storage cask. The bases include the applicable codes and standards, individual loads as related to operating and environmental conditions and natural phenomenon events. They also define load combinations and stress allowables for the service conditions consistent with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code.

<u>Codes and Standards</u>. Section A4.2.3.3.1 of the SAR notes that the cask containment vessel (i.e., confinement boundary) is designed to the maximum practical extent as an ASME Class 1 component in accordance with the rules of the ASME Section III, Subsection NB. Section A4.2.3.3.3 states that the basket is designed, fabricated, and inspected in accordance with the ASME Code Subsection NG to the maximum practical extent. Section A3.5 summarizes ASME Code alternatives, including the basket fusion weld capacity as demonstrated by qualification and production testing. American National Standards Institute (ANSI) N14.6, "American National Standard for Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," are used for evaluating the cask lifting trunnions. Compared to the TN-40 cask, more individual loadings are included for load combination cases, which continue to meet the intent of the ANSI/American Nuclear Society (ANS) 57.9, "Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)," standard, or equivalent, and are acceptable.

<u>Site Environmental and Natural Phenomenon Loads</u>. Section A3.2 of the SAR presents the site environmental and natural phenomenon loads associated with tornado and wind loadings, water level (flood) design, seismic design, and snow and ice loadings used in the design-basis analysis of the TN-40HT system. Except for the 18-inch cask handling accident vertical drop, which is reviewed in Section 5.3.1 of this SER, the SAR uses approaches similar to those for the TN-40 to calculate mechanical and pressure loads for a slightly heavier yet shorter TN-40HT cask system. The resulting loads and their structural effects are seen to vary little from those for the TN-40.

<u>Load Combinations</u>. Tables A3.2-9 and A3.2-10 of the SAR list load combination cases for the respective normal and accident conditions in evaluating the structural performance of the TN-40HT. Compared to those for the TN-40, major case additions include the 18-inch cask end-drop deceleration g-load and fabrication stresses introduced by shrink-fitting the confinement inner shell to the cask body.

5.2 Acceptance Criteria

10 CFR 72.122 does not provide explicit structural acceptance criteria except to satisfy the safety requirements of that section. For the cask performance review, the structural integrity of the cask is deemed adequate if it can be demonstrated that the stresses and displacements induced by the loads noted in Section 5.3.1 of this SER are lower than the allowables for the cask components important to safety. SAR Tables A4.2-2, -3, and -4 summarize stress allowables for the containment vessel, non-confinement structures, and fuel basket, respectively. The stress allowables for the confinement closure bolts are also listed in Table A4.2-2. In general, the stress allowables are in accordance with the standards established by the ASME B&PV Code. Consequently, they conform to the NUREG-1567, "Standard Review

Plan for Spent Fuel Dry Storage Facilities," guidance on quality standards requirement of 10 CFR 72.122(a), and are acceptable.

5.3 Review

The staff reviewed the SAR for compliance with 10 CFR 72.122(a), which refers to quality standards that govern the characterization of materials, the establishment of stress allowables, and the design, as well as analysis methods that provide confidence in the capability of the structures, systems, and components to perform the required safety function. SAR Section A4.2.1 re-evaluates the ISFSI structure for deploying the TN-40HT casks for a maximum cask design basis weight of 250,000 lbs, which is heavier than that of the TN-40 design basis at 240,700 lbs. Considering a slightly reduced cask base diameter and recognizing the large margins that existed in the previous evaluation, however, the staff agrees with the SAR conclusion that more rigorous static and dynamic analyses with the TN-40HT are not warranted. Hence, the evaluations performed for the TN-40 continue to be applicable for demonstrating the structural capability of the ISFSI pad and its foundation. In the following, the SAR is reviewed for compliance with 10 CFR 72.122, which requires the cask be designed for protection against environmental conditions, natural phenomena, and postulated accidents.

5.3.1 Cask Loads

The SAR specifies the TN-40HT cask loads, which are similar to those used for the TN-40 evaluation. Table A3.2-2 continues to consider the bounding normal operating internal pressure of 100 psi for the cask since it envelopes all applicable pressure conditions, and is acceptable.

In Section A3.2.5.4.3, the trunnion loads are defined on the basis of ANSI N14.6, which requires that lifting devices be capable of lifting 6 times and 10 times the cask weight, without exceeding the respective yield and ultimate strengths of the material.

Table A4.2-1 summarizes the individual load cases analyzed for the cask body, lid bolts, and trunnions.

10 CFR 72.122(b)(1) provides that the cask be designed to accommodate effects of postulated accidents. The staff notes that the cask will not tip over per the current licensing basis. Section A4A.10 of the SAR performs a cask vertical drop analysis for a handling height 18 inches above the 3-foot thick ISFSI storage pad. The LS-DYNA finite element transient analysis follows essentially the NUREG/CR-6608, "Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet onto Concrete Pads," approach to modeling the cask body, concrete pad, and soil subgrade. The fuel and basket are considered as an elastic solid cylinder inside the cask walls. A design-basis end-drop deceleration of 50 g is then selected, which envelops the calculated peak decelerations, for both the cask body and fuel basket.

5.3.2 Review of Cask Body

Section A4A of the SAR presents structural analysis of the TN-40HT cask body for the normal operating conditions, accident conditions, and loads associated with environmental conditions and natural phenomena. Table A4A.3-1 summarizes the 10 individual loading cases, which include three extra cases, such as fabrication stress effects, more than those for the TN-40. The analysis described in Section A4.2.3.4 also considers the 50-g deceleration inertia force associated with the cask end-drop accident. Except for the hand calculation for trunnion local stresses due to a 3-g lifting load, the cask body stress performance is analyzed with ANSYS three-dimensional (3-D) finite element models.

Section A4A.3.2 of the SAR presents the finite element model details, including boundary and loading conditions for each individual loading case. Sections A4.2.3.4.1 and A4.2.3.4.2 describe the process of evaluating cask body stresses for load combinations. Tables A4.2-8 and A4.2-9 summarize nodal stresses for various cask body components under the normal and accident conditions, respectively. Table A4.2.10 presents the corresponding primary membrane, P_m, and membrane-plus-bending, $P_l + P_b$, stress intensities for the component sections deemed to be the most critical. These stresses are all less than the allowables, and are, therefore, acceptable.

Sections A3.2.2, A3.2.3, and A3.2.4 evaluate performance of the cask and conclude that its structural integrity is maintained for flood, seismic, and snow and ice loading conditions. Section A.3.2.1 evaluates effects of tornado and wind loadings, which include tornado missiles impact on the cask body. Similar to those for the TN-40, the evaluation concludes and the staff agrees that the cask body will perform adequately, and that neither the gamma shielding forging nor the confinement boundary will be punctured by the postulated missiles.

Section A4A.10 of the SAR evaluates the cask axial deceleration for the cask vertical drop from a handling height of 18 inches above the ISFSI storage pad. A maximum cask body deceleration of 44.1 g is calculated from a transient impact analysis. Tables A4.2.6 shows that effects of the design basis cask deceleration of 50 g, which bounds the 44.1-g deceleration, is combined with relevant loadings, including bolt preload, fabrication stress, and cask cavity internal or external pressure, for the cask body evaluation. As summarized in Table A4.2-9, since the calculated nodal stress intensities are far below the membrane stress allowable, the staff has reasonable assurance to agree with the SAR conclusion, which relies primarily on the nodal stress intensity values, that the cask body structural performance is acceptable for the hypothetical cask drop accident.

5.3.3 Review of Basket

Figures A4B.1-2 and 1-3 of the SAR depict the basket finite element model, which is represented by a 15-inch long axial basket section supported laterally by aluminum transition rails. The model conservatively ignores strength of the aluminum conducting plates and uses pipe elements to simulate use of fusion welds for joining the fuel compartment walls through the intervening spacer bars. Table A4B.1-1 lists the five loading cases associated with mechanical and thermal loading conditions. Section A4B.1.5.2.4 notes the ANSYS gap element implementation for analyzing the lateral loads exerted at the 0°, 30°, 45°, 60°, and 90° azimuth orientations. Section A4B.1.5.5 notes and the staff agrees that the fuel is supported by the cask bottom plate and, as such, the fuel inertia load need not be considered in the evaluation of the basket subject to the end-drop accident.

Table A4B.1-5 of the SAR summarizes maximum calculated basket stresses in the fuel compartments and the transition rails for the five lateral load orientations. Table A4B.1-6 compares the maximums to the stress allowables for the corresponding primary membrane, P_m, membrane-plus-bending, $P_l + P_b$, and thermal, Q, stress intensity categories for the fuel compartments and transition rail. Section A4B.1.5.7 notes that all maximum primary stress intensities, P_m and $P_l + P_b$, are below the membrane allowable, S_m, and all secondary stress intensities, $P_l + P_b + Q$ are less than $3S_m$. This demonstrates adequate structural performance of the basket for the design basis loading conditions.

5.3.4 Review of the Neutron Shield, Trunnions, and Other Components

Section A4A.7 of the SAR evaluates structural performance of the cask outer shell, as a neutron shield enclosure, using a finite element analysis. Table A4A.7-1 summarizes the maximum stress intensities with large stress factors of safety for the governing loading cases, including the case of the 25-psig internal pressure combined with a 3-g inertia load.

Section A3.2.5.4.3 of the SAR notes that the upper trunnions are evaluated for vertical lifting reaction applied on the lifting shoulders for the loads of 6 times and 10 times the maximum weight of a fully loaded cask. When the load is equal to 6 and 10 times the weight, the maximum tensile stresses are below the respective yield and ultimate strengths of the trunnion material, as required by ANSI N 14.6.

Section A4.A.4 of the SAR describes the lid bolt analysis, considering load conditions such as bolt preload, gasket seating load, cask internal pressure, and temperature load. Table A4.2-12 summarizes the results. The highest bolt stresses are much below the stress allowables for both normal and accident conditions. The shear and tensile stress interaction equation check values are 0.345 and 0.226 for the normal and accident conditions, respectively, which are less than unity and acceptable.

5.3.5 Fuel Assemblies

Section A4.2.3.8 of the SAR evaluates structural integrity of the fuel rod cladding for the cask handling end-drop accident. The applicant adapted a single-pin analytical model, developed by Pacific Northwest National Laboratories (PNNL) and also used in NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," to determine the strain ductility demand that a typical fuel pin may be subject to during an 18-in cask bottom-end drop. The model was constructed with shell elements for the fuel cladding, internal springs to prevent ovalization of the cladding, springs to represent spacer grids, a cask-to-fuel pin spring, and a cask-to-ground spring representing the deformation characteristics of the bottom plate of the cask. The applicant also included internal pressure to ensure that all realistic loadings were considered.

The results of the finite element analysis illustrate that the strain ductility demand of 0.76% is below the yield strain of 0.96% at 600 °F. This indicates that the fuel cladding remains elastic and subsequently fully intact with respect to the cask accident drop event.

Staff reviewed the applicant's methodology and results and finds that the implementation of the model and subsequent conclusions by the applicant provide a reasonable assurance of safety.

5.4 EVALUATION FINDINGS

On the basis of the evaluation, the staff concludes that the applicant has provided adequate installation and structural evaluations to meet regulatory requirements for major categories of structures, systems, and components (SSCs), including: (1) confinement structures, systems and components, (2) reinforced concrete structures, (3) other important to safety SSCs, and (4) other SSCs subject to NRC approval. The staff's evaluation findings include:

F5.1 The SAR and docketed materials relating to the description of confinement SSCs meet the requirements of 10 CFR 72.24 (a) and (b).

F5.2 The SAR and docketed materials relating to design criteria for confinement SSCs, including applicable codes and standards, meet the requirements of 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.122 (a), (b), and (l); 10 CFR 72.128 (a); and 10 CFR 72.236 (b) and (e). Additionally, the confinement SSC design criteria meet the guidance provided in applicable parts of Regulatory Guides for protection against seismic and tornado events.

F5.3 The SAR and docketed materials provide adequate analytical and/or test reports to ensure that structural integrity of the confinement SSCs meet the requirements of 10 CFR 72.24 (d)(1) and (d)(2), and 10 CFR 72.122 (b)(1), (b)(2), and (l).

F5.4 The SAR and docketed materials relating to the description of other SSCs important to safety meet the requirements of 10 CFR 72.24 (a) and (b).

F5.5 The SAR and docketed materials relating to design criteria for other important to safety SSCs, including applicable codes and standards, meet the requirements of 10 CFR 72.24 (c)(1), (c)(2), and (c)(4); 10 CFR 72.122 (a), (b); 10 CFR 72.128 (a); and 10 CFR 72.236 (b).

F5.6 The SAR and docketed materials provide adequate analytical and/or test reports to ensure that structural integrity of the other SSCs important to safety meet the requirements of 10 CFR 72.24 (d)(1) and (d)(2), and 10 CFR 72.122 (b)(1) and (b)(2).

6 THERMAL EVALUATION

The maximum enrichment and bundle average burnup of the TN-40HT design are 5.0 wt% U-235 and 60 GWd/MTU, respectively. In order to accommodate the higher burnup fuel and higher enrichment, the heat transfer capability of the basket had to be enhanced. To do this, the TN-40HT cask employs a slightly thinner lid, shield shell and cask bottom shield. However, the radial neutron shield thickness is increased to offset the higher neutron source of the high burnup fuel.

6.1 DECAY HEAT REMOVAL SYSTEMS

6.1.1 General Considerations

In the design of the TN-40HT, different modes of heat transfer are taken into consideration. For example, solar radiation occurs on the outer surface of the cask. In the case of the fire accident conditions, convection and radiation heat transfer are combined as these modes take place on the outer surface of the cask. During the post-fire cool down period, free convection and radiation occur. Within the lid-seal region model, radiation heat transfer is used to maximize the heat input from the fire via the protective cover to the vent and drain port seals located in the cask lid.

Section A3.3.2.2 provides discussion on the TN-40HT basket and the neutron shielding, which play vital roles in the heat transfer throughout this design. The basket consists of an assembly of 40 stainless steel fuel compartments sandwiched between aluminum and neutron poison plates. A fusion welding process welds 1.75 in. wide stainless steel bars to connect the compartments. Above and below the bars are slotted aluminum and neutron poison plates, which form an egg-crate structure. Stainless steel basket rails including aluminum inserts are bolted to the basket periphery to provide a conduction path from the basket to the cask cavity wall. This thermal design feature of the basket allows the heat from the fuel assemblies to be conducted along the basket structure to the basket rails and dissipated to the cask cavity wall.

A resin compound cast into long slender aluminum provides the neutron shielding containers placed around the cask shell and enclosed within a smooth outer shell. By butting against the adjacent shell surfaces, the aluminum containers provide a conduction path and allow decay heat to be conducted across the neutron shield.

In designing the TN-40HT cask, both the internal and external pressures are established. The design internal pressure for this cask is 22 psig, which bounded the normal and off-normal operating pressures. The design external pressure for this cask is 25 psig because it exceeds the anticipated maximum external pressure for any of the loading conditions considered above, including floodwater discussed in Section A3.2.2, and snow and ice in Section A3.2.4.

6.1.2 Dry Storage Systems

In evaluating the TN-40HT cask, the applicant provided the environmental temperatures for normal and off-normal conditions. The normal storage condition temperature applied here is 50 °F. Within the off-normal conditions, the maximum off-normal storage temperature is 100 °F. The applicant used 10 CFR Part 71 to dictate the insolation values for the maximum amount of solar radiation available for absorption on any surface. The SAR for this cask design ensures the site conditions are enveloped by the cask thermal analysis and these conditions are encapsulated in Section A3.3.

6.1.3 Pool Systems

Within this amendment, this section was not applicable in the analysis of this cask.

6.1.4 Dry Transfer Systems

In reviewing the TN-40HT cask, the staff performed an evaluation to confirm that the fuel cladding temperature did not exceed 400°C (752°F) for normal and vacuum drying conditions, and did not exceed the allowable limit of 570°C (1058°F) for off-normal and accident conditions. Listed below in Table 3 are the predicted temperatures for each condition along with their maximum temperatures, displaying that the fuel cladding did not exceed the respective temperature barriers.

Table 3 - Maximum Predicted Fuel Cladding Temperatures

TN – 40HT Fuel Clad	Vacuum Drying	Normal Conditions	Accident Conditions
Fuel Cladding Temperature	725 °F	680 °F	772 °F
Maximum Fuel Cladding Temperature	752 °F	752 °F	1058 °F

6.2 MATERIAL TEMPERATURE LIMITS

6.2.1 General Considerations

Section A3.3.2.2 of the applicant's SAR describes the thermal model used to evaluate the ability of the TN-40HT cask to transfer the heat generated by the spent fuel assemblies to the environment. The thermal properties of the materials are presented in Table A3.3-8.

The cask utilizes Helicoflex metallic seals, which have a maximum service temperature of 280 °C (536 °F). The cask uses a top neutron shield material with a maximum service temperature of 149 °C (300 °F). Due to the predicted temperatures not exceeding the maximum temperatures in the neutron shield, staff has reasonable assurance that the neutron shield material will perform as required. Table 4 below displays maximum predicted temperatures within the cask versus the maximum or limiting temperatures for normal conditions of storage (NCS).

Table 4 - Maximum Predicted Temperatures for O-Ring Seals and Neutron for NCS

Location	Predicted Temperature (°C)	Maximum Temperature (°C)
O-Ring Seals	85 (185 °F)	280 (536 °F)
Top Neutron Shield	88 (191 °F)	149 (300 °F)
Radial Neutron Shield	141 (285 °F)	149 (300 °F)

Section A3.3.2.2.1.2.2 defines the assumptions/boundary conditions employed for the analysis of a buried cask. The results of the buried cask accident show that if the cover of the cask is not removed within 1.85 hours, the neutron shield temperature exceeds the allowable limit of 300 °F. The fuel temperature exceeds the allowable limit of 1058 °F about 95.75 hours after the cask is buried completely. During the 95.75 hours, the seals will reach a temperature of 316 °F, well below the maximum service temperature of 536 °F.

6.2.2 Fuel Cladding

This table confirms that the maximum fuel cladding temperatures for normal conditions, vacuum drying conditions, and accident conditions do not exceed 400 °C (752 °F) for normal and vacuum drying conditions and 570 °C (1058 °F) for off-normal and accident conditions.

6.2.3 Special Thermal Criteria for Reinforced Concrete

Within this amendment, this section was not applicable in the analysis of this cask.

6.2.4 Extreme Low Temperatures

In Section A3.3.2.2.1.1.3, the lowest temperature used to design this cask was -40 °F. The analysis performed on the TN-40HT cask is in accordance with the lowest designed temperature captured in the SAR.

6.3 THERMAL LOADS AND ENVIRONMENTAL CONDITIONS

The maximum heat load of the TN-40HT cask design is 32 kW. The maximum decay heat based on the maximum heat load is 0.8 kW/assembly. The maximum bundle average burnup is 60 GWd/MTU. The values used for solar insolation were as follows: 400 gcal/cm^2 for the cask surface, curved and painted, 800 gcal/cm^2 for the cask surface, flat horizontal and painted, and 800 gcal/cm^2 for concrete, flat and horizontal.

6.4 ANALYTICAL METHODS, MODELS, AND CALCULATIONS

The SAR presented the analytical methods, the various thermal models, and calculations within Section A3.

6.4.1 Finite Element Models

ANSYS was used to design various models for the normal, off-normal, steady-state, and accident conditions. In designing the full-length cask model for normal and off-normal conditions, the SAR depicts the model as three-dimensional and represents a 90° symmetric section of the TN-40HT cask. This model includes the geometry and the material properties of the basket, basket rails, cask shells, cask lid, protective cover, cask bottom plates, radial neutron shield, and top neutron shield, as well as the concrete pad and supporting soil. The protective cover was modeled using SHELL57 elements. For conservatism, no heat transfer is considered between the protective cover and the upper surface of the cask lid in the full-length model to minimize the axial heat transfer. It was also assumed that no convection heat transfer occurred within the cask cavity. The effective thermal conductivity calculated for the homogenized fuel includes the radiation heat transfer. Conduction through the basket and cask components is modeled using SOLID70 elements. Further discussion of the full-length cask model can be found in Section A3.3.2.1.1.1.

The other finite element model used for normal, off-normal, steady-state, and accident conditions represents only the top portion of the TN-40HT cask. The model for the cask top includes the radiation between the top neutron shield and protective cover in order to determine the maximum temperatures of the seals and the top neutron shield resin.

The finite element models and their heat transfer processes using steady-state boundary conditions are located in Section A3.3.2.2.1.1.3. For accident conditions, the finite element models are located in Section A3.3.2.2.1.2.

6.4.2 Calculations

Some of the calculations used in performing the thermal analysis on the respective finite element models depicted in the SAR are presented in this section.

In determining the uniform heat flux at the bottom of the cask top sub-model, the following equation was used:

$$q''_{top} = \frac{q_{top}}{\pi/4D^2} \quad (1)$$

where q''_{top} is the uniform heat flux in Btu/hr-in^2, q_{top} is the heat flow (Btu/hr), and D is the diameter of the bottom of the cask top sub-model (in.).

In the cross-section model during the hypothetical fire accident conditions, the heat generating rate to be placed across the homogenized fuel is as follows:

$$\dot{q}'' = \frac{q}{a^2 L_a} \cdot PF \quad (2)$$

where $\dot{q}''$ is the decay heat load per assembly (Btu/hr), PF is the maximum peaking factor, L_a is the active fuel length (in.), and a is the width of the modeled fuel assembly (in.).

In calculating the total heat transfer coefficient to ambient via free convection and radiation, the equations are listed as follows:

$$H_t = h_r + h_c \quad (3)$$

where H_t is the total heat transfer coefficient, h_r is the radiation heat transfer coefficient, and h_c is the convective heat transfer coefficient.

$$h_r = \varepsilon * F_{12}[\sigma\,(T_1^2 - T_2^2)/(T_1 - T_2)] \quad (4)$$

where ε is the surface emissivity, F_{12} is the view factor from cask surface to ambient, σ is 0.1714 x 10^{-8} Btu/hr-ft²-°R^4 , T_1 is the cask surface temperature (°R), and T_2 is the ambient temperature (°R).

6.4.3 Analysis of the Finite Element Models

Staff reviewed the thermal analysis for NCS in the SAR. Staff reviewed the applicant's models and calculation options to assess the adequacy of the applicant's proposed design and to make certain that the fuel cladding temperature does not exceed values specified in the SAR. Additionally, staff examined the applicant's ability to apply the necessary boundary conditions, heat loads, and other conditions for each of the respective cases. After examining the aforementioned items, staff ran the input files provided by the applicant. Table 3, listed in Section 6.1.4 of the SER, displays the maximum predicted temperatures during vacuum drying and normal conditions versus the maximum or limiting temperatures of the fuel clad. Due to the predicted temperatures of the fuel cladding not exceeding the maximum temperatures, staff has reasonable assurance that the fuel cladding will maintain integrity in NCS and vacuum drying conditions.

The thermal analysis in the SAR for an accidental fire was reviewed to determine if any radioactive release could occur in violation of 10 CFR 72.106. A pool with a diameter of 180" and a fuel consumption rate of 0.15 in/min (selected from a Sandia National Labs report on open pool fires) was used for this accident fire scenario. The SAR assumed the cask was exposed to a 1475°F engulfing fire for 15 minutes. The analysis of the fire accident was based upon an analytic model of the cask where the thermal conductivity values for SA 516, Gr. 70, Al – 6063 and aluminum 1100 were used. During the fire, the maximum fuel cladding temperature is still well below the maximum allowable temperature.

Section A3.3.2.2.1.2.2 within the SAR defines the assumptions/boundary conditions employed for the analysis of a buried cask. The results of the buried cask accident show that if the cover of the cask is not removed within 1.85 hours, the neutron shield temperature exceeds the allowable limit of 300 °F. The fuel temperature exceeds the allowable limit of 1058 °F about 95.75 hours after the cask is buried completely. During the 95.75 hours, the seals will reach a temperature of 316 °F, well below the maximum service temperature of 536 °F.

6.5 PROTECTION FROM FIRE AND EXPLOSIONS

6.5.1 General Considerations

According to A3.3.6, no hydrocarbon fuel of any sort will be stored in the ISFSI. The quantity of fuel carried in the tow vehicle will be limited so that only a small fire of short duration would be possible. There are no other significant combustible sources within the ISFSI security fence. Due to the large thermal mass of the casks any minor fires in the vicinity of the ISFSI would raise the cask temperature by only a few degrees and are not expected to affect cask integrity.

6.5.2 Spent Fuel Casks

Within this amendment, this section was not applicable in the analysis of this cask.

6.5.3 SSCs Important to Safety

Within this amendment, this section was not applicable in the analysis of this cask.

6.5.4 Guidance for a Fire Protection Program

Within this amendment, this section was not applicable in the analysis of this cask.

6.6 EVALUATION FINDINGS

F6.1 SSCs important to safety are described in sufficient detail in Section 3 of the SAR to enable an evaluation of their heat removal effectiveness. Cask structures, systems, and components important to safety remain within their operating temperature ranges in accordance with 10 CFR 72.122.

F6.2 The TN-40HT is designed with a heat-removal capability having testability and reliability consistent with its importance to safety as required by 10 CFR 72.128.

7 SHIELDING EVALUATION

The staff reviewed the shielding evaluation for the TN-40HT dry cask storage system to determine if the cask design features are adequate to protect against direct radiation from the cask contents while in dry storage at the Prairie Island Independent Spent Fuel Storage Installation. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

7.1 Shielding Design Features and Design Criteria

The TN-40HT dry cask is a modified version of the TN-40 dry cask containing a fuel basket that accommodates up to 40 14x14 PWR fuel assemblies with or without fuel inserts. The TN-40HT cask is designed for the storage of high enrichment and high burnup fuel. The cask body has a slightly thinner lid, shield shell, and cask bottom shield. A gamma shield is provided around the walls and bottom of the containment vessel made of carbon steel. A lid shield plate is also attached to the inside of the containment lid.

The thickness of the radial neutron shield was increased to accommodate the increased neutron source as a result of the high burnup fuel. This neutron shield is enclosed in an outer steel shell. A neutron shield is also attached to the cask lid to provide additional shielding.

The TN-40HT cask is designed to accommodate up to 40 14x14 PWR assemblies with or without inserts. The maximum allowable initial enrichment is 5.0 wt% U-235. The maximum bundle average burnup, maximum decay heat, and minimum cooling time for the fuel assembly are 60 GWd/MTU, 0.80 kW/assembly (including heat from inserts), and 12 years, respectively. The maximum decay heat load that the cask is designed for is 32 kW.

The overall radiological protection design criteria are the regulatory dose requirements in 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), and maintaining occupational exposures as-low-as-reasonably-achievable (ALARA). The applicant analyzed the radiological effects of a design basis assembly on occupational personnel and individuals outside of the containment area.

7.2 Source Specification

A sensitivity analysis was performed for candidate assemblies to determine which assembly type was the most limiting, or design basis assembly. The design basis assembly was determined to be the Westinghouse Standard 14x14 assembly with a burnup, bundle average enrichment, and cooling time of 60 GWd/MTU, 3.4 wt% U-235, and 18 years, respectively. The radiological characteristics for the PWR spent fuel were generated using the SCALE SAS2H/ORIGEN-S depletion modules. Fuel with various combinations of burnup, enrichment, and cooling times can be stored in the TN-40HT cask as long as the combination results in decay heats and surface dose rates that are bounded by the design basis fuel.

The source terms for the design basis assemblies assumed that all assemblies included a fuel insert burnable poison rod assembly (BPRA) in the fuel region and a fuel insert thimble plug device (TPD) for the plenum and top end fitting. This method bounds the use of both types since the TPD does not extend into the fuel region but has a higher source term than the BPRA in the plenum and end fitting areas. The TPD is irradiated at an equivalent host assembly life burnup of 125 GWd/MTU and an initial enrichment of 3.85 wt% U-235. The BPRA is irradiated at an equivalent host assembly life burnup of 30 GWd/MTU and an initial enrichment of 3.85

wt% U-235. These were cooled at 16 years and 18 years, respectively.

As part of the source term analysis, the boron concentration was chosen to be 600 ppm for the first cycle, with the second cycle having 95% of this value. As part of the design basis qualification process, the effects from the change in boron concentration were evaluated.

The reactor moderator temperature can vary between 500-600°F. However, a moderator density corresponding to a temperature of 566°F was used in the analysis. As part of the design basis qualification process, a sensitivity analysis was performed to determine the effects from a change in moderator density resulting from changing moderator temperature.

7.2.1 Gamma Source

The gamma source terms are calculated for the burnup, enrichment, and cooling time combinations that yield the highest total dose rates at the 2 meter radial distance as described in the SAS2H model of the SAR. The hardware activation analysis considered the cobalt impurities in the assembly hardware. The activated hardware source terms are calculated using the hardware masses listed in The SAR and the appropriate activation ratios. Although cobalt impurities can vary, the applicant's assumed values are reasonable and acceptable.

7.2.2 Neutron Source

The neutron source terms are calculated for the burnup, enrichment, and cooling time combinations that yield the highest total dose rates at the 2 meter radial distance as described in the SAS2H model of the SAR. The neutron source in this analysis was represented by the Curium-244 (Cm-244) energy spectrum, which is the primary neutron source nuclide at cooling times beyond five years.

7.3 Confirmatory Analyses

The staff performed confirmatory source term evaluations using the SCALE 5.1 computer code with the SAS2H/ORIGEN-S isotopic depletion and decay sequence with the 44-group ENDF/B-V cross section library. Using irradiation parameter assumptions similar to the applicant's, the staff obtained bounding source terms that were similar to, or bounded by, those determined by the applicant. Scale 6.0 was also used for confirmatory modeling of the storage cask.

The SAR analysis provides reasonable assurance that the TN-40HT can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b).

7.4 Shielding Analysis

The radiation source is modeled as a homogenized fuel assembly within a discretely modeled fuel basket. The design characteristics, composition, and densities of the materials used in the shielding analysis are presented in Tables A7.2-1 thru A7.2-5 of the SAR. There are some conservative representations that differ from the actual design, as stated in Section A7A.4 of the SAR.

The SAR stated that localized areas of elevated dose rates should be anticipated due to streaming paths through drain and vents ports. The applicant stated that the streaming was due to handling operations involving draining water from the cask. Due to the port covers and top neutron shield, the streaming only produces a negligible effect on the offsite dose. During cask loading, procedures, worker training, and radiation protection barriers are put in place to minimize worker dose to ALARA. Furthermore, due to distance and the surrounding earth berm, the streaming path would have negligible effect on off-site dose.

7.4.1 Computer Programs

The applicant generated neutron and gamma source terms using the SCALE 4.4 computer code with the SAS2H/ORIGEN-S isotopic depletion and decay sequence with the 44-group ENDF/B-V cross section library. The design basis source terms were selected from burnup, enrichment, cooling time combinations that yield the highest dose rates at 2 meters from the cask midplane.

The applicant used MCNP for the bounding external dose rate calculations. The MCNP three-dimensional Monte Carlo particle transport code is a standard in the nuclear industry for performing neutron and photon shielding analyses. The staff agrees that the codes and cross-section data used by the applicant are appropriate for this particular application and fuel system.

The staff performed confirmatory source term evaluations using the SCALE 5.1 computer code with the SAS2H/ORIGEN-S isotopic depletion and decay sequence with the 44-group ENDF/B-V cross section library. Staff also used SCALE 6.0 to confirm dose rates listed in the SAR. The dose rates obtained by staff were slightly lower than those found in the SAR. This was determined to be attributed to the difference in the number of source locations used to model the system.

7.4.2 Flux-to-Dose-Rate Conversion

As listed in the SAR, the applicant uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose rate conversion factors to calculate dose rates, which is an acceptable methodology.

7.4.3 Normal Conditions

A single shielding model of the TN-40HT cask design was developed for both the normal and off-normal conditions of storage. During normal conditions of storage the cask is upright and placed on a concrete pad. During off-normal conditions, loading and transfer activities may be taking place. This model contains a discrete basket configuration with homogenized fuel assemblies positioned within the fuel compartments.

The gamma model was modified with distinct radial cask steel layers to calculate the primary gamma dose rates. The neutron model was utilized to calculate the neutron and secondary gamma dose rates.

The dose rates corresponding to normal and off-normal conditions are presented in Table A7A.2-1 of the SAR. These are dose rates for the external surface of the cask and at one (1) and two (2) meters from the cask. For areas where the neutron shield is present, the neutron dose rates are less than the gamma dose rates. However, in areas below and above the neutron shields, the neutron dose rates are considerably higher. The dose rates on the bottom of the cask are considered to be a factor during loading and transfer operations.

7.4.4 Accident Conditions

The model used for accident conditions of storage is similar to that used for normal conditions with the exception that all neutron shielding and the outer steel shell materials were removed and replaced by void space in the accident condition model. The accident condition is based on a cask drop scenario which results in the removal of all neutron shielding materials.

The dose rates corresponding to the accident conditions of storage are presented in Table A7A.2-1 of the SAR.

As stated in Section 7.4 of this SER, the dose rate estimates in areas where radiation streaming may occur are not addressed in the SAR.

7.5 Occupational Exposures

The analysis in Section A7A of the SAR used the design basis fuel to estimate occupational exposures corresponding to storage of the TN-40HT cask. Section A7A of the SAR presents the estimated occupational exposures that are based on dose rate calculations in Section A7 of the SAR. The staff's evaluation of the occupational exposures is located in Section 11 of this SER. Dose rates must meet the limits incorporated into the technical specifications for normal and off-normal conditions.

7.5.1 Off-site Dose Calculations

Section A7A of the SAR evaluates the offsite dose rates for two (2) 2x12 arrays of TN-40HT casks. Section A7A presents the calculated offsite annual doses for these arrays at distances of 10 to 1000 meters. These generic off-site calculations demonstrate that the TN-40HT cask system is capable of meeting the offsite dose criteria of 10 CFR 72.104(a).

Section 11 of this SER evaluates the overall off-site dose rates from the TN-40HT cask system. The actual dose to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, and atmospheric conditions. In addition, 10 CFR 72.104(a) includes doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of the applicant.

The applicant must also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and must demonstrate compliance with dose limits to individual members of the public as required by evaluation and measurements. Engineered features for radiological protection are considered important to safety and must be evaluated to determine the applicable quality assurance category.

7.5.2 Confirmatory Calculations

The staff performed confirmatory analysis of selected source terms used in the dose evaluation. The staff based its verification on design features specified in the SAR and modeling assumptions used in the analysis. Limiting fuel characteristics, and the burnup and cooling time, are included in the technical specifications (TS). The staff's calculated source term results were in reasonable agreement with the SAR values or were generally lower due to the applicant's conservative loading assumptions. Confirmatory dose rates obtained by staff were slightly lower than those found in the SAR. This was determined to be attributed to the difference in the number of source locations used to model the system.

7.6 EVALUATION FINDINGS

The staff determined that the SAR has adequately demonstrated that the TN-40HT cask system is designed to meet the criteria of 10 CFR 20.1201(a)(1), 10 CFR 72.24(e), 10 CFR 72.104(a), 10 CFR 72.106, 10 CFR 72.126(a)(6), and 10 CFR 72.128(a)(2).

F7.1 The design of the shielding system(s) of the ISFSI satisfies the criteria for radiological protection of 10 CFR 72.126(a)(6). The shielding design is evaluated in the Radiation Protection Design Features section of the SAR.

F7.2 The design of the ISFSI provides acceptable means for limiting occupational radiation exposures within the limits given in 10 CFR 20.1201 and for meeting the objective of maintaining exposures as low as is reasonably achievable, in compliance with 10 CFR 72.24(e). Occupational exposures are evaluated and listed in the Estimated Onsite Collective Dose Assessment section of the SAR.

F7.3 The design of the ISFSI provides acceptable means for limiting exposure of the public to direct and scattered radiation within the limits given in 10 CFR 72.104. This was evaluated in the Radiation Protection Design Features section of the SAR.

F7.4 The design of the ISFSI provides suitable shielding for radioactive protection under normal and accident conditions, in compliance with 10 CFR 72.128(a)(2). Items specific to shielding are evaluated in the Shielding design Features section of the SAR.

8 CRITICALITY EVALUATION

In its March 28, 2008 letter, NMC (now NSPM) discusses the criticality analysis for the TN-40HT storage cask as a modification of the TN-40 cask for the storage of higher enrichment and higher burnup Westinghouse and Exxon 14x14 PWR fuel.

8.1 Criticality Design Criteria and Features

Regulations require the package to be maintained subcritical in normal, off-normal and accident conditions. 10 CFR 72.124(b) lists the acceptable methods of criticality control.

The basket will include a permanently fixed neutron absorbing material. This material will either be a Borated-Aluminum alloy, Aluminum/B_4C metal matrix composite with a minimum absorber aerial density of 37.5 mg B-10/cm^2, or Boral with a minimum areal density of 45.0 mg B-10/cm^2. The basket components are fabricated as orthogonally positioned plates with notches to accommodate intersecting plates. All plates are identical with poison on one side of the central aluminum plate, the net effect of which is to poison interior basket positions on all sides and exterior positions on all sides except the radially outward face. Smaller notches opposite the larger ones accommodate the steel rails used to join axial sections when stacked to construct the full length of the fuel basket.

The SRP for Spent Fuel Dry Storage Facilities (NUREG-1567) allows for the credit of borated water that serves as both shielding and absorber in the spent fuel pool. During loading operations, the cask will be located in the pool containing at least 2450 ppm boron. Credit for borated water is not taken to maintain criticality safety during storage in the application.

After loading, the cask seal will prevent the intrusion of fresh water as a third aspect of criticality control.

8.2 Stored Material Specifications

The applicant has limited maximum enrichment, maximum bundle average burnup, and minimum cooling time to 5.0 wt %, 60 GWd/MTU, and 12 years, respectively. An additional thermal limit is placed and the maximum heat generation is 0.80 kW/assembly including heat from any inserts with a maximum cask heat load of 32 kW.

The heat load limits, when applied using the fuel qualification table (FQT) for some fuel assemblies, may result in some scenarios where fuel qualified under the maximum enrichment and burnup limits will not be authorized to load without additional cooling time beyond the stated minimum.

The applicant states that the most reactive assembly type is the Westinghouse Standard 14x14 assembly with 5.0% lattice enrichment. Detailed criticality analyses are presented in the SAR to demonstrate that for all modeling assumptions, the package k_{eff} will remain below 0.95.

Additional authorized contents may include non-fuel hardware limited to Burnable Poison Rod Assemblies (BPRAs) and Thimble Plug Devices (TPDs).

8.3 Analytical Means

8.3.1 Model Configuration

The model geometry explicitly consists of a single axial “egg-crate” section with periodic top and bottom boundaries. This in essence creates an axially infinite stack of 14.49-inch “egg-crate” sections. The actual egg-crate length is 15 inches. Modeling the cask in this way conservatively underestimates the amount of poison per unit length, and eliminates axial neutron leakage.

The applicant replaced the radial neutron shield between casks with un-borated water in the criticality model.

The applicant’s analysis conservatively uses a 90% credit for the poison in Borated-Aluminum alloy and Aluminum/B_4C metal matrix, and 75% credit for Boral. In the case of BPRAs, no credit is taken for absorber in the inserts and is modeled as B_4C with 100% boron-11. This conservatively underestimates the amount of boron-10 poison, both displaced in borated moderator and in the BPRA itself, and includes carbon moderator. Any other potential absorbers in the fuel material, such as Gadolinia or Erbia, are conservatively omitted from the criticality model. All other materials of concern are present in the model.

8.4 Applicant Criticality Analysis

The applicant investigated the reactivity of a full load of the more limiting fuel assemblies. The applicant investigated both the outward and inward shifting of fuel assemblies within the basket, changes in fuel compartment tube internal width, fuel compartment tube thickness, poison plate thickness, stainless steel bar thickness, and basket periphery structure. The assembly average enrichment in this analysis was 4.5 wt% U-235. The intent of this exercise was to determine relative and not absolute reactivity to find the most reactive case. The final criticality analysis was done with fresh fuel enriched to 5.0 wt% U-235. The results of the criticality analyses are summarized in Tables A3.3-23 through A3.3-29.

8.4.1 Computer Program

The applicant used the three-dimensional Monte Carlo CSAS25 module in the SCALE package for criticality analysis to determine the bounding assembly. The calculations used the 44-group ENDF/B-V cross section libraries.

8.4.2 Multiplication Factor

The maximum k_{eff} calculated by the applicant corresponds to a configuration with an initial enrichment of 5.0 wt% with 2450 ppm borated water and inserts. Including uncertainty, the maximum calculated k_{eff} of 0.9373 is below the USL of 0.9419 calculated in accordance to NUREG/CR-6361.

8.4.3 Benchmark Comparisons

A set of 121 experiments was used to determine the USL. A complete list of the benchmarks with the results is presented in Table A.3.3-30. The benchmark problems are representative of UO_2 commercial LWR fuel assemblies with no burnup credit.

8.5 Staff Analysis

Staff utilized MCNP5 to verify the findings of the applicant. The cross section libraries used by the staff are applicable in the 300 K temperature range, as well as a scattering kernel for hydrogen in water at ambient temperature.

MCNP limitations on periodic boundary intersection vectors made their use difficult while using the model submitted by the applicant as a template. Instead, NRC staff modeled the cask in two ways. One method was approximating an infinite stack of 14.49-inch egg-crate sections using white and not periodic or reflective axial boundaries. Reflective boundaries were seen as inappropriate given the axial asymmetry in the egg-crate model; there is a distinct top and bottom to the model. White boundaries utilize isotropic reflection to approximate the presence of additional egg-crates above and below. In this case, the poison plates are modeled as discrete plates clad in aluminum. Poison density and plate thickness were analyzed with the egg-crate stack model.

A second model was made which was infinite in the axial direction. However, it was not possible to discretely model the gaps between the poison plates. As a result, an additional length of poison is added to the model, limiting the utility of this model to relative effects. Staff analyzed the effect of lattice separation and borated water density on the axially infinite model since a relative effect is all that is necessary to draw an appropriate conclusion. The flooded case was determined to be the most reactive.

A final criticality calculation utilized the same bounding parameters as the analyses performed by the applicant. The cask was analyzed with a complete loading of WE 14x14 class assemblies at 5.0 wt% enrichment, 33.6 mg/cm^2 reduced areal boron density in the poison plates, and flooded with 2450 ppm borated water. Staff calculations yielded results in reasonable agreement with the applicant's criticality analysis. Given the additional conservatisms present in the criticality model, the staff finds the applicant's analyses sufficient to demonstrate the package will remain subcritical.

8.6 EVALUATION FINDINGS

F8.1 The design, procedures and materials to be stored in the TN-40HT at Prairie Island provide reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public, in compliance with 10 CFR 72.40(a)(13).

F8.2 The SAR analyses and confirmatory analysis by NRC staff show that acceptable margins of safety will be maintained in the nuclear criticality parameters commensurate with uncertainties in the data and methods used in calculations, and demonstrate safety for the handling, packaging, transfer and storage conditions and in the nature of the immediate environment under accident conditions; in compliance with 10 CFR Part 72.124(a) and 10 CFR 72.124(b).

9 CONFINEMENT EVALUATION

The confinement review ensures that radiological releases to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross rupture. This confinement evaluation focuses on the source term for high burnup fuel, the use of ANSI N14.5-1997 for leakage rate calculations, and the confinement analysis for off-normal and accident conditions.

9.1 Confinement Description

The staff reviewed the applicant's confinement analyses in Section A7A.8 and the drawings in Section A1 of the SAR. The applicant clearly identified the confinement boundary in Figure A1.3-1 of the SAR. The confinement boundary consists of the cask inner shell and bottom plate, shell flange, lid outer plate, lid bolts, vent and drain port cover plates and bolts, the inner portions of the lid seal, and the inner portions of the vent and drain port seals. All confinement boundary components are important to safety Category A components according to NUREG/CR-6407 "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety."

The cask inner shell and bottom plate are made of SA-203, Grade E low alloy steel. The shell flange and lid outer plate are made of SA-350 Grade LF3 or SA-203, Grade E. The vent and drain cover plates are made of SA-240 Type 304. The lid bolts are made of SA-540 Grade B24 or B23 CL1 and the cover plate bolts are made of SA-193 GB7.

The TN-40HT cask confinement boundary is designed, fabricated, and inspected in accordance with the ASME B&PV Code, Section III, Subsection NB, to the maximum extent practicable. The alternatives to the ASME code requirements are documented in Section A3.5 of the SAR. The containment vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200.

The confinement boundary welds consist of longitudinal weld(s) on the rolled plate which close the cylindrical inner shell, and circumferential weld(s) which attach the rolled shells together as well as attach the bottom inner plate and the shell flange to the inner shell. The confinement boundary base material and associated welds will be helium leak tested in accordance with ANSI N14.5-1997 with an acceptance criterion of $1x10^{-7}$ ref cm^3/s.

Helicoflex HND 229 or equivalent double metallic O-ring seals are utilized on the lid and the two lid penetrations. The metallic seals have a stainless steel or nickel alloy liner with an aluminum jacket and contain a Nimonic 90 or equivalent spring material. All seating surfaces which mate with the metal seals are stainless steel clad. The pressure monitoring system of the double seal interspace can be seen in Figure A3.3-1 of the SAR. Once loaded, all lid and cover seals are helium leak tested and the acceptable total cask leakage (both inner and outer seals combined) is $1x10^{-5}$ ref cm^3/s. The double metallic O-ring seals have a maximum temperature limit of 536°F and remain below that limit during normal/off-normal conditions, fire accident, buried cask, and during vacuum drying.

The containment vessel contains an integrally-welded bottom closure and bolted and flanged top closure (lid). The lid is bolted to the cask body with 48 bolts, each with a torque of 1100 to 1150 ft-lbs which maintains confinement under normal and accident conditions. The closure bolt analysis is presented in Section A4A.4. The vent and drain lid penetrations are sealed by flanged covers bolted to the lid with eight bolts each, each with a torque of 60 to 65 ft-lbs which maintains confinement under normal and accident conditions. The penetration bolt analysis is presented in Section A4A.5.

Section A5.1.1 of the SAR states that Section 5.1.1 of the SAR is applicable to the operation of the TN-40HT casks.

9.2 **Radionuclide Confinement Analyses**

Section A7.2.6 of the SAR states that the Westinghouse 14x14 standard is the design basis fuel, with an initial enrichment of 3.4 wt% U-235, burnup of 60GWd/MTU, and a cooling time of 18 years. The applicant determined this combination of fuel parameters was bounding by comparing dose rates for various combinations of burnup, enrichment, and cooling time. The bounding radiological source terms for confinement were generated with SAS2H and Table A7.2-6 of the SAR lists the activity representing the fission gases, volatiles, and fines contributing more than 0.1% of the activity contained in the design basis fuel, actinides that contribute more than 0.01% of the activity contained in the design basis fuel, plus iodine 129. The applicant provided release fractions that were in agreement with those found in NUREG 1567, Table 9-2. The applicant justified using 10% of the fuel fines ejected remaining airborne.

For normal conditions of storage, the cask cavity pressure is always above ambient. The bolted closures have double seals and the seal interspace is pressurized to provide a positive pressure gradient. Leakage of the inner seal would cause helium to flow into the cask cavity, not allowing for release of radioactive material. Leakage of the outer seal would cause helium to leak from the overpressure system to the exterior, and no radioactive material would be released. Because the region between the redundant confinement boundary mechanical seals is maintained at a pressure greater than the cask cavity, the monitoring system boundaries are tested to a leakage rate equal to the confinement boundary, the seal pressure is routinely checked and instrumentation is verified to be operable in accordance with the Technical Specification Surveillance Requirement, the staff has accepted that no discernible leakage is credible.

Under off-normal conditions it is assumed that the overpressure system is not functioning properly and it is assumed that the cask cavity gas will leak out at a rate of 1×10^{-5} ref cm^3/s. The applicant calculated the helium leakage rate for off-normal conditions, $L_{u,he} = 1.518*10^{-5}$ cm^3/s of helium using the methodology of ANSI N14.5-1997. Under hypothetical accident conditions it is also assumed that the overpressure system has stopped functioning, that the cask cavity gas will leak out at a rate of 1×10^{-5} ref-cm^3/s, and fire conditions exist. The applicant calculated the helium leakage rate for hypothetical accident conditions $L_{u,he} = 3.725*10^{-5}$ cm^3/s of helium also using the methodology of ANSI N14.5-1997.

The following two scenarios are considered for confinement analysis. Off-normal conditions are for a 45 day period, seals are leaking at the test rate of $1*10^{-5}$ ref-cm^3/s and 10% of rods have failed. Stability category D, a 5 m/s wind speed, and the assumption that one cask was in an off-normal condition at the ISFSI were used for the off-normal condition analysis. The applicant has justified the use of a 45 day period based on the Technical Specification 3.1.5, where the maximum duration the cask would be in the off-normal condition would be 38 days, which is conservatively bounded. Hypothetical accident conditions are for a 30 day period, seals are leaking at the test rate of $1*10^{-5}$ ref-cm^3/s, 100% of rods have failed, the cask temperature is average cavity gas temperature following the fire = 592 °F. Stability category F, a 1 m/s wind speed, and the assumption that one cask was in hypothetical accident condition at the ISFSI were used for the hypothetical accident conditions.

The methodology of Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants” was used to calculate $(\chi/Q)_{110\ meters} = 1.29*10^{-3}$ s/m^3 during off-normal conditions, where 110 meters is the minimum distance to the site boundary. The same methodology was used to calculate $(\chi/Q)_{700\ meters} = 4.63*10^{-5}$ s/m^3 during off-normal conditions where 724 meters is the distance from the center of the ISFSI to the nearest resident, therefore 700 meters is conservative. The staff confirmed these calculations.

The methodology of Regulatory Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling Storage Facility for Boiling and Pressurized Water Reactors” was used to calculate $(\chi/Q)_{110\ meters} = 6.63*10^{-3}$ s/m^3 during hypothetical accident conditions. The same methodology was used to calculate $(\chi/Q)_{0.45\ miles} = 2.66*10^{-4}$ s/m^3 during hypothetical accident conditions (where 0.45 miles = 724 meters). The staff confirmed these calculations were conservative using Regulatory Guide 1.145.

The methodology of Regulatory Guide 1.109, “Calculation of Annual Doses to Men from Routing Releases of Reactor Effluent for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I” and dose conversion factors (DCF) from EPA Federal Guidance Reports No. 11 “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion” and No. 12 “External Exposure to Radionuclides in Air, Water, and Soil” were used to calculate the dose components for off-normal and hypothetical accident conditions. Because the Sr-90 fission product should not form $SrTiO_3$ within the storage cask, the DCF for $SrTiO_3$ was not used, while the DCF for Sr in all other forms was used.

For off-normal conditions, Tables A7A.8-3 and A7A.8-4 of the SAR present the estimated 45 day airborne doses (internal and external) at 110 meters. The staff confirmed these calculations. A tabular summary of the off-normal conditions doses and limits meeting 10 CFR 72.104(a) is shown in Section A7A.8.6.2.1 of the SAR and the doses shown are within limits.

For hypothetical accident conditions, Tables A7A.8-5 and A7A.8-6 of the SAR present the estimated 30 day internal and external doses at 110 meters. The staff confirmed these calculations. A tabular summary of the accident conditions doses and limits meeting 10 CFR 72.106(b) is shown in Section A7A.8.6.2.1 of the SAR and the doses shown are within limits.

9.3 **Confinement Monitoring Capability**

An overpressure (OP) monitoring system of the double seal interspace which can be seen in Figure A3.3-1 of the SAR is part of the TN-40HT. The pressure of the seal interspace is greater than that of the cask cavity and also greater than ambient. In-leakage of air or out-leakage of the cavity gas is not possible under this configuration. Both seals are collectively leak tested to $1x10^{-5}$ ref cm^3/s. ANSI N14.5-1997 was used to calculate an equivalent maximum hole size based upon equivalent air leaking from 1 atm to 0.01 atm absolute at 77 °F that corresponds to the specified allowable leakage rate. The staff confirmed this calculation.

During operations the overpressure system is initially back filled with 66.2 psig of helium at standard temperature. The temperature of the helium in the overpressure tank at equilibrium is 160 °F (71 °C) and the pressure of the overpressure tank is 79 psig. This pressure is higher than both the cask cavity and the atmosphere. From Table A3.3-16 of the SAR, the cask internal pressure for normal conditions (1% fuel and BPRA failure) is 13.0 psig, for off-normal conditions (10% fuel and BPRA failure) is 17.5 psig, for fire accident conditions (100% fuel and BPRA failure) is 74.3 psig, and for a buried cask accident at 75 hours (100% fuel and BPRA failure) is 94.9 psig. The internal cask cavity pressures are at or below the design limit of 100 psig. The leak rate of the overpressure system to atmosphere was calculated to be $L_{u,He}$ = $4.489*10^{-5}$ cm^3/s of helium. The maximum volume leaked from the overpressure system over the first year and the corresponding pressure reduction was calculated. The pressure of the overpressure tank at the start of the 2^{nd} year was calculated and the calculations were repeated with Figure A7A.8-2 illustrating the overpressure monitoring system pressure drop over the 25 year life of the cask. The overpressure system is set to alarm if the overpressure system drops below 27.8 psig. This set point is based on the maximum off-normal cask cavity pressure (17.5 psig) plus 10.3 psi for margin to ensure pressure decreases in the overpressure monitoring system are identified before any potential out leakage from the cask cavity occurs.

Latent seal failure is addressed in Section A.7A.8.6.3 of the SAR where two tables summarize the following cases: case one, if there is leakage of the overpressure system to the atmosphere; and case two, if there is leakage to the cask cavity. These two tables show the estimated time to alarm and estimated time to lose or equalize overpressure system pressure as a function of leak rate. The tables show that there is an appropriate amount of time to evaluate the leak and the time could be extended by re-pressurizing the overpressure tank.

If a latent seal failure occurs and the overpressure system is removed due to an accident, the results in Section A.7A.8.6.3 of the SAR shows that a failure up to 100 times greater than the test value could occur and still allow for recovery before 10 CFR 72.106(b) limits are exceeded.

9.4 **Protection of Stored Materials from Degradation**

The TN-40HT maintains an inert helium atmosphere inside the cask cavity. The helium assists in heat removal and provides a non-reactive environment to protect the fuel assemblies against fuel cladding degradation. Vacuum drying is discussed in Technical Specification 3.1.1 and the cask helium backfill pressure and associated limits are discussed in Technical Specification 3.1.2. As discussed above, fabrication leakage rate tests will be performed on the entire confinement boundary (including the confinement boundary base material, welds, and seals) in accordance with ANSI N14.5-1997. The confinement boundary seals (lid, vent, and drain) will be helium leakage rate tested during loading per Technical Specification 3.1.3. The thermal analysis of the TN-40HT discussed in Chapter 6 of the SER indicates that the fuel cladding temperatures will not exceed their limits. Finally, as discussed above, the overpressure monitoring system will ensure in-leakage of air or out-leakage of the cavity gas is not possible. The staff found these Technical Specifications, the confinement boundary testing according to ANSI N14.5, the results of the thermal analysis, and the use of the overpressure monitoring system to be appropriate to protect the spent fuel cladding against degradation.

9.5 **EVALUATION FINDINGS**

F9.1 Sections A7A.8.1 through A7A.8.3 of the SAR describes confinement structures, systems, and components (SSCs) important to safety in sufficient detail to permit evaluation of their effectiveness.

F9.2 The design of the TN40-HT provides redundant sealing of the confinement system closure joints by the use of double metal seals utilized on the lid and vent and drain penetrations.

F9.3 The quantity of radioactive nuclides postulated to be released to the environment has been assessed as discussed above. In Section 11 of the SER, the dose from these releases will be added to the direct dose to show that the TN40-HT satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F9.4 The confinement system is monitored with an overpressure monitoring system as discussed above. No instrumentation is required to remain operational under accident conditions.

F9.5 The design and proposed operations of the Prairie Island Independent Spent Fuel Storage Installation provides adequate measures for protecting the spent fuel cladding against degradation that might otherwise lead to gross ruptures of the material to be stored in compliance with 10 CFR 72.122(h)(1).

F9.6 The TN40-HT confinement system has been evaluated by analysis. Based on successful completion of specified leakage tests and examination procedures, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F9.7 The staff concludes that the design of the confinement system of the TN40-HT is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the TN40-HT will allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, the applicant's analyses and the staff's confirmatory analyses, and acceptable engineering practices.

10 CONDUCT OF OPERATIONS EVALUATION

The applicant identified the information throughout this section of the SAR as being independent of cask design. Rather than duplicating the discussions and information already presented in the TN-40 SAR, the discussion for the TN-40HT is presented by reference to the appropriate sections of the TN-40 SAR. The last column of the table shown below lists the appropriate sections.

Table 5 - Conduct of Operations

Section/Subsection		SAR Section
Organizational Structure		9.1
Corporate Organization		9.1.1
	Corporate Functions, Responsibilities and Authorities	9.1.1.1
	ISFSI Project Organization	9.1.1.2
	Relationship with Contractors and Suppliers	9.1.1.3
	Technical Staff	9.1.1.4
Operating Organizations, Management and Administrative Control System		9.1.2
	Onsite Organization	9.1.2.1
	Personnel Functions, Responsibilities and Authorities	9.1.2.2
Personnel Qualification Requirements		9.1.3
	Minimum Qualification Requirements	9.1.3.1
	Qualifications of Personnel	9.1.3.2
Liaison with Outside Organizations		9.1.4
Startup Testing and Operation		9.2
Administrative Procedures for Conducting Test Program		9.2.1
Test Program Description		9.2.2
	Physical Facilities	9.2.2.1
	Operations	9.2.2.2
Test Discussion		9.2.3
Completion of Pre-Operational Test Program		9.2.4

10.1 Normal Operations

This was unaffected by the changes proposed in this amendment request.

10.1.1 Procedures

This was unaffected by the changes proposed in this amendment request.

10.1.2 Records

This was unaffected by the changes proposed in this amendment request.

10.2 Personnel Selection, Training, and Certification

This was unaffected by the changes proposed in this amendment request.

10.3 Emergency Planning

This was unaffected by the changes proposed in this amendment request.

10.4 Physical Security and Safeguards Contingency Plans

This was unaffected by the changes proposed in this amendment request.

10.5 EVALUATION FINDINGS

There are no evaluation findings; conduct of operations were unaffected by the changes proposed in this amendment request.

11 RADIATION PROTECTION EVALUATION

The staff reviewed the radiation protection design features, design criteria, and the operating procedures of the TN-40HT Cask System to ensure that it meets the regulatory dose requirements of 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), and 10 CFR 72.24(e). This amendment was also reviewed to determine whether the TN-40HT Cask System fulfills the acceptance criteria listed in Section 11 of NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities”. Staff conclusions are based on information provided in Appendix A of the Prairie Island Independent Spent Fuel Storage Installation Safety Analysis Report.

11.1 Radiation Protection Design Criteria and Design Features

11.1.1 Design Criteria and Features

Sections A7 and A7A of the SAR define the shielding and radiological protection design features which provide radiation protection to operational personnel and members of the public. The radiation protection design features and operational procedures include the following:

- The casks are loaded, sealed, and decontaminated prior to transfer to the ISFSI.
- The fuel will not be unloaded nor will the casks be opened while at the ISFSI.
- The fuel will be stored dry inside the casks, so that no radioactive liquid is available for leakage.
- The casks will be sealed with a helium atmosphere to preclude oxidation of the fuel.
- Shielding is provided by a thick-walled cask body (e.g., a neutron shield surrounding the cask body and cask lid, and a steel shell surrounding the neutron shield).
- The containment vessel prevents leakage of radioactive material from the cask cavity.
- The confinement system consists of multiple welded barriers to prevent atmospheric release of radionuclides, and is designed to maintain confinement of fuel during accident conditions,
- ALARA principles are implemented into the cask design and operating procedures to reduce occupational exposures.

The staff evaluated the radiation protection design features and design criteria for the TN-40HT cask system as detailed in the SAR and found them acceptable. The SAR analysis provides reasonable assurance that use of the TN-40HT cask in a manner described in the SAR can meet the regulatory requirements in 10 CFR Part 20, 10 CFR 72.24(e), 10 CFR 72.104(a), and 10 CFR 72.106(b). Section 7 of the SER discusses staff’s evaluation of the TN-40HT shielding features.

11.1.2 Occupational Exposures

Section A7 of the SAR discusses the estimated exposures involved in maintenance and surveillance activities for the storage of the TN-40HT cask. Table A7.4-1 shows the estimated design basis occupational exposures to ISFSI personnel during the loading, transport, and placement of the storage casks. Table A7.4-2 shows the estimated design basis annual exposure for surveillance and maintenance activities. Both Tables A7.4-1 and A7.4-2 list, for each task, the estimated time required for the task, the number of personnel required, the design basis dose rates, and the exposure.

11.1.3 Exposures to the Public during Normal and Off-Normal Conditions

The SAR defines the site boundary and identifies the exclusion area as the controlled area for the ISFSI. In calculating the offsite collective dose, the entire permanent population within 2 miles surrounding the site, are assumed to be located at the residence subject to the highest exposure. In addition, the analysis also took into consideration a large transient population as a result of those visiting a local hotel and casino. Dose rates resulting from cask storage at the ISFSI as a function of distance are shown in Table A7A.7-2. Table A7.5-2 summarizes the calculated total dose to off-site population within a 2-mile radius during normal and off-normal operations.

The staff evaluated the public dose estimates and found them acceptable. The primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions is from direct radiation (including skyshine). The cask confinement function is not affected by normal or off-normal conditions; therefore, no discernable leakage is credible during normal and off-normal conditions. A discussion of the staff's evaluation and confirmatory analysis of the shielding calculations are presented in Section 7 of the SER.

The applicant must also have an established radiation protection program as required by 10 CFR Part 20 and must demonstrate compliance with dose limits to individual members of the public, as required in 10 CFR Part 20 by calculations or measurements.

11.1.4 Public Exposures from Accidents and Events

Section A8 of the SAR shows various accident conditions such as extreme winds, explosion, and cask drop and asserts that these conditions are bounded by the scenarios involving a complete loss of the neutron shield combined with the effects of a loss of one confinement barrier and a complete cladding failure. The concluding dose rates for these accident conditions to individuals beyond the controlled area were calculated to be below the 5 rem TEDE limit as specified in 10 CFR Part 72.106 (B).

Staff evaluated the public dose estimates from direct radiation from the accident conditions and found them acceptable. A discussion of the staff's evaluation and any confirmatory analysis of the shielding analysis is presented in Section 7 of this SER. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents will be below the regulatory limits in 10 CFR 72.106(b).

11.2 ALARA

As part of the review, staff was unable to identify areas within the shielding evaluation and radiation protection sections of Appendix A of the SAR where ALARA policies were addressed with the exception of Section A7.3. However, Section 7.1 of the SAR provides a discussion of the ALARA program established in accordance with the requirements listed in 10 CFR 72.126. Section 7.1 provides some detail of the ALARA policies in place that govern design considerations and operational practices. Section A7 states that ALARA policies listed in Section 7.1 of the SAR are independent of cask design, and that design considerations listed in Section 7.1.2 are applicable to the TN-40HT cask system.

11.3 EVALUATION FINDINGS

F11.1 The design and operating procedures of the ISFSI provide acceptable means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR 20 and for meeting the objective of maintaining exposures ALARA, in compliance with 10 CFR 72.24(e).

F11.2 The SAR and other documentation submitted in support of the application provide acceptable and reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public, in compliance with 10 CFR 72.40(a)(13).

F11.3 The design of the ISFSI provides suitable shielding for radiation protection under normal and accident conditions, in compliance with 10 CFR 72.128(a)(2).

12 QUALITY ASSURANCE EVALUATION

12.1 Review Objective

The purpose of the review is to determine whether the applicant for a license to store spent fuel or high-level waste has a quality assurance (QA) program that complies with the requirements of 10 CFR Part 72, Subpart G. The basis for that determination is a review and evaluation of the applicant's QA program submitted as a part of the application in accordance with 10 CFR 72.24(n). The results of the review and evaluation are documented in this section of the SER.

12.2 Areas of Review

The Quality Assurance information in Section 11.1 of the Prairie Island Independent Spent Fuel Storage Installation Safety Analysis Report is independent of cask design. The unchanged subsections of Section A11 are listed below.

A.11.1.1	Organization
A.11.1.2	Quality Assurance Program
A.11.1.3	Design Control
A.11.1.4	Procurement Document Control
A.11.1.5	Instructions, Procedures and Drawings
A.11.1.6	Document Control
A.11.1.7	Control of Purchased Materials, Equipment and Services
A.11.1.8	Identification and Control of Materials, Parts and Components
A.11.1.9	Control of Special Processes
A.11.1.10	Inspection
A.11.1.11	Test Control
A.11.1.12	Control of Measuring and Test Equipment
A.11.1.13	Handling, Storage and Shipping
A.11.1.14	Inspection, Test and Operating Status
A.11.1.15	Non-Conforming Materials, Parts or Components
A.11.1.16	Corrective Action
A.11.1.17	Quality Assurance Records
A.11.1.18	Audits
A.11.2	Quality Assurance Program – Contractors
A.11.2.1	Architect – Engineer
A.11.2.2	Cask Supplier
A.11.2.3	Concrete Storage Pad Contractor

The change in section A11.1, the Quality Assurance Program Description, is the exception. The location of the TN-40HT safety related components were changed and those changes are listed in Table A4.5-1.

12.3 EVALUATION FINDINGS

The staff concludes from the information provided in the application that the regulatory requirements continue to satisfy and the QA program continues to meet the acceptance requirements.

F12.1 The QA program describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of 10 CFR Part 72, Subpart G.

F12.2 The QA program covers activities affecting SSCs important to safety as identified in the

Safety Analysis Report.

F12.3 The organizations and persons performing QA functions have the independence and authority to perform their functions without undue influence form those directly responsible for costs and schedules.

F12.4 The licensee's description of the QA program is in compliance with applicable NRC regulations and industry standards, and the QA program can be implemented for the design, fabrication and construction, and operation phases of the installation's life cycle.

13 DECOMMISSIONING EVALUATION

The primary objective of the decommissioning evaluation is to ensure that the applicant's provisions for eventual decontamination and decommissioning of the independent spent fuel storage installation give reasonable assurance of adequate protection of public health and safety. In the review, an examination of the design and operational features intended to facilitate eventual decommissioning, and the proposed decommissioning plan and associated financial assurance and recordkeeping requirements would ordinarily be reviewed. As part of this amendment request the applicant did not include the associated financial assurance and recordkeeping requirements. The applicant does, however, specify in Table 1, "Criteria Sections Affected by the TN-40 Design Modifications" of Enclosure 3 (L-PI-08-020), that the regulatory requirement for satisfying the criteria for decommissioning (10 CFR 72.130) was unaffected by the changes proposed in this amendment request.

13.1 Decommissioning Plan

The applicant states in Section A4.6 of the SAR: "The information outlined in Section 6.1 is applicable to the TN-40HT casks except for the location of the decommissioning plan for the TN-40HT casks which is located in Section A4.6."

13.1.1 General Provisions

By application dated April 16, 2008 (ML081090353) Nuclear Management Company, LLC (NMC), licensed operator for PINGP and the Prairie Island ISFSI, and Northern States Power Company, a Minnesota Corporation (NSPM), licensed power of the aforementioned facilities, requested an order consenting to the transfer of operating authority for those facilities from NMC to NSPM.

The following "Financial Qualifications and Decommissioning Funding Assurance", is as stated in the Safety Evaluation dated September 15, 2008 (ML082240750).

Financial Qualifications

Per ML082240750, "With respect to the Prairie Island ISFSI license, 10 CFR 72.22(e) requires that NSPM show that it possesses the necessary funds or that it has reasonable assurance of obtaining the necessary funds to cover estimated operating costs over the planned life of the ISFSI. In connection with an order dated May 12, 2000, the Commission, in approving the transfer of the operating license for the ISFSI to NSPM (then referred to as "New NSP"), stated in the associated safety evaluation that "the application [for the transfer of the ISFSI operating license to NSPM] states that the ratemaking process to which New NSP will be subject as an electric utility provides reasonable assurance that New NSP will be financially qualified to operate and decommission the Prairie Island ISFSI." Since NSPM remains an electric utility subject to total cost of service ratemaking, NSPM is financially qualified to hold the operating authority under the subject license."

Decommissioning Funding Assurance

Per ML082240750, "The NRC has determined that the requirements to provide assurance of decommissioning funding and provision of an adequate amount of decommissioning funding are necessary to ensure the adequate protection of public health and safety. The regulation in 10 CFR 50.33(k) requires that an applicant for an operating license for a utilization facility contain information to demonstrate how reasonable assurance will be provided that funds will be available to decommission the facility. A similar requirement is imposed on ISFSIs under 10 CFR 72.22(e)."

NSPM, as the licensed owner of the facilities, is already responsible for decommissioning the NSPM facilities. The transfer of operating authority under the respective licenses has no impact on decommissioning funding, and no decommissioning funding assurance analysis, therefore, is necessary.

13.2 EVALUATION FINDINGS

There are no evaluation findings. The regulatory requirement for satisfying the criteria for decommissioning (10 CFR 72.130) was unaffected by the changes proposed in this amendment request.

14 WASTE CONFINEMENT AND MANAGEMENT EVALUATION

Section A6 – Waste Management – Design (Section A6.1) references the information outlined in Section 6.1 as being applicable to the TN-40HT casks except for the location of the decommissioning plan for the TN-40HT casks which is located in Section A4.6.

Section 6.1 of the TN-40 SAR states the following: "No radioactive wastes will be generated during storage of the casks at the ISFSI or during cask transport outside of the Auxiliary Building. Radioactive wastes generated during loading operations in the Auxiliary Building will be treated using existing PINGP radioactive waste control systems as described in Section 9 of the PINGP USAR (Reference 1).

Contaminated pool water removed from loaded storage casks will normally be drained back into the spent fuel pool with no additional processing. A small amount of liquid waste will result from storage cask decontamination. The decontamination procedure will result in a small amount of a detergent/demineralized water mixture being collected in the cask decontamination area. Liquid wastes collected in the cask decontamination area are directed to the aerated waste sump tank, where it will be mixed with other plant liquid wastes, treated or held up for decay, and released.

Potentially contaminated air and helium purged from the storage casks following spent fuel loading will be handled by the spent fuel pool ventilation systems, as described in the PINGP USAR, Section 10.3.7, or the gaseous radwaste system, as described in the USAR, Section 9.3. Air in the spent fuel pool area is normally exhausted through roughing and HEPA filters. In the event of a high radiation signal, ventilation is performed by the spent fuel pool special ventilation system, which has roughing, HEPA and activated charcoal filters.

A small quantity of low level solid waste will be generated as a result of storage cask loading operations and transfer cask decontamination. The solid waste generated will be processed as described in the PINGP USAR, Section 9.4. This low level waste will consist of disposable anticontamination garments, tape, blotter paper, rags, etc."

14.1 EVALUATION FINDINGS

There are no evaluation findings; waste confinement and management was unaffected by the changes proposed in this amendment request.

15 ACCIDENT ANALYSIS

The objective the review is to perform a systematic evaluation of the applicant's identification and analysis of hazards for both off-normal and accident or design basis events involving structures, systems, and components (SSCs) important to safety.

Section A8 of the applicant's SAR presents the off-normal operations and accidents anaylzed.

15.1 Off-Normal Operations

Off-normal operations are design events of the 2nd type (as defined in ANSI/ANS 57.9). The loss of electrical power was design event analyzed. The postulated cause of the event, detection of events, analysis of effects and consequences, and corrective action were presented. No radiological impact from off-normal operations was postulated.

15.2 Accidents

The accidents, design events of the 3rd and 4th type (as defined in ANSI/ANS 57.9), analyzed by the applicant are listed below. The cause of the accident, accident analysis, and accident dose calculations are presented for each event in the SAR. The staff review of public exposure from the following accidents and events is discussed in section 11.1.4 of the SER.

- Earthquake
- Extreme wind
- Flood
- Explosion
- Fire
- Inadvertent loading of a newly discharged fuel assembly
- Loss of confinement barrier.

The applicant also analyzed the following accidents:

- Cask seal leakage – discussed in section 9.2 of the SER.
- Hypothetical cask drop accident – discussed in section 5.3 of the SER. (cause of accident, accident analysis, and accident dose calculations)
    - Dynamic impact loads
    - Cask body analysis
    - Lid bolt analysis
    - Basket analysis

15.3 EVALUATION FINDINGS

F15.1 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the ISFSI will acceptably meet the requirements without endangering the public health and safety, in compliance with the overall requirements of 10 CFR 72.122.

F15.2 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the ISFSI will acceptably meet the requirements of 10 CFR 72.124 regarding the maintenance of the spent fuel in a subcritical condition.

F15.3 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the ISFSI will acceptably meet the requirements of 10 CFR 72.126 regarding criteria for radiological protection.

F15.4 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design of the ISFSI will acceptably meet the requirements of 10 CFR 72.128 regarding handling, storage, and retrievability of the spent fuel and other radioactive material.

16 TECHNICAL SPECIFICATIONS

The review of the technical specifications (TS) is based on information presented in the various technical design chapters and the technical specifications chapter presented in the applicant's SAR.

16.1 Revised TS – Cross Reference Matrix

The applicant completely revised the technical specifications for the Prairie Island ISFSI. The revised TS (RTS) are formatted in accordance with the format adopted for the Prairie Island Nuclear Generating Plant Technical Specifications. Technical Specification Bases are relocated in a separate, independent document that will be controlled as specified in the RTS. For each current TS (CTS) requirement, action, and surveillance there is a corresponding RTS requirement, action, or surveillance.

The applicant provided a Revised Technical Specification Matrix, to serve as a cross reference **from** the Current Technical Specifications **to** the Revised Technical Specifications. For each change of the Current Technical Specifications a characterization was made as to whether the change was: No Change to a technical requirement (NC); a More Restrictive technical requirement (MR); or a Less Restrictive technical requirement (LR). For NC characterizations, additional explanatory notes are provided where the applicant determined appropriate. For MR characterizations, additional explanation notes are provided. For LR characterizations, additional explanation notes are provided as well as a summary safety determination.

Tables 6 – 8 at the end of this section list the NC, LR, and MR technical requirements respectively. The acronyms used are as described below:

LCO – Limiting Condition for Operation
FOL – Functional and Operating Limits
Def – Definition
Spec – Specification
SR – Surveillance Requirement

16.2 EVALUATION FINDINGS

F16.1 The staff concludes that the conditions for use at the Prairie Island ISFSI identify necessary technical specifications to satisfy 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The proposed technical specifications provide reasonable assurance that the ISFSI will allow safe storage of spent fuel. This finding is based on the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

F16.2 In addition to the applicant's proposed technical specifications, the staff finds that the following additional technical specifications are required:

4.3 <u>Neutron Poison Loading in the TN-4HT Casks</u>

The minimum areal boron-10 density of the neutron poison plates shall meet that specified in Table 4.3-1. This will ensure that the poison loading is consistent with that assumed in the criticality analysis. **Sections A9.7.3, A9.7.4, and**

A9.7.5 of the TN-40HT SAR are incorporated into the technical specifications by reference.

Dose rates on the external surface of the cask may not be bound by localized dose rates due to streaming during loading, maintenance, or surveillance activities. Therefore, appropriate measures should be implemented to ensure exposures are consistent with good ALARA practices. (This is included as a footnote on page 3.2.2-2 of the Tech Specs.)

Table 6 - No Change to a Technical Requirement (NC)

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
SR 4.6.2	Administrative Controls 5.2	NC	
FOL 2.3	Design Feature 4.2	NC	
LCO 3.1.1(5)	FOL 2.1.a, Def of UNDAMAGED FUEL ASSEMBLIES, & LCO 3.4.1	NC	
LCO 3.1.1(6)	FOL 2.1.a, Def of UNDAMAGED FUEL ASSEMBLIES, & LCO 3.4.1	NC	
LCO 3.1.1(1)	FOL 2.1.b, FOL 2.1.c, & LCO 3.4.1	NC	Since there are no VANTAGE+ fuel assemblies that meet the other requirements for storage in a TN-40 cask, e.g. enrichment and burnup limits, the inclusion of VANTAGE+ fuel as a part of the OFA fuel type is not a change to the fuel that may be stored in a TN-40 cask.
LCO 3.1.1(8)	FOL 2.1.d.i, & LCO 3.4.1	NC	
LCO 3.1.1(7)	FOL 2.1.d.ii, & LCO 3.4.1	NC	Corrected technical error of applying term “burnup” to BPRAs.
LCO 3.1.1(9)	FOL 2.1.e.ii, & LCO 3.4.1	NC	Corrected technical error of applying term “burnup” to TPDs.
LCO 3.1.1(11)	FOL 2.1.f, & LCO 3.4.1	NC	
LCO 3.1.1(12)	FOL 2.1.g, & LCO 3.4.1	NC	
LCO 3.1.1(2)	FOL 2.2.a, & LCO 3.4.1	NC	
LCO 3.1.1(3)	FOL 2.2.b, & LCO 3.4.1	NC	

Table 6 - No Change to a Technical Requirement (NC) continued

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
LCO 3.1.1(4)	FOL 2.2.c, & LCO 3.4.1	NC	
LCO 3.1.1(13)	FOL 2.2.d, & LCO 3.4.1	NC	
LCO 3.1.1, Action	FOL 2.4.1 & LCO 3.4.1 Required Action A.1	NC	
FOL 2.2, Action 2	LCO 3.1.2, Required Action A.2	NC	If the inability to meet the backfill pressure limit is due to exceeding the criterion, then helium will have to be released in order to accomplish the new Required Action A.2. Thus there is no change to the technical requirements.
LCO 3.3.1	LCO 3.1.3, & SR 3.1.3.1	NC	
LCO 3.8.1	LCO 3.1.4	NC	
LCO 3.5.1	LCO 3.1.6, & SR 3.1.6.1	NC	
LCO 3.4.1 Action	LCO 3.2.1 Required Action A.1	NC	
LCO 3.4.1	LCO 3.2.1, & SR 3.2.1.1	NC	
LCO 3.2.1 Action 1	LCO 3.3.1, SR 3.3.1.1, SR 3.3.1.2, & SR 3.0.4	NC	Surveillance SR 3.3.1.1 requires verifying that the dissolved boron concentration limit is satisfied within 4 hours prior to commencing LOADING OPERATIONS. SR 3.0.4 will not allow entry into LOADING OPERATIONS unless SR 3.3.1.1 is satisfied. Surveillance SR 3.3.1.2 requires verifying that the dissolved boron concentration limit is satisfied within 4 hours prior to flooding the cask for UNLOADING OPERATIONS. Hence there would be no activities involving cask loading and unloading unless the boron concentration is above the limit, i.e., they would be suspended.
Section 5.0	Section 4.0	NC	No change other than to include TN-40HT casks.
Section 6.1	Section 5.1	NC	
Section 6.2	Section 5.2	NC	

Table 6 - No Change to a Technical Requirement (NC) continued

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
Section 6.3	Section 5.3	NC	
SR 4.8.1	SR 3.1.4.1 and SR 3.1.4.2	NC	
SR 4.4.1	SR 3.2.1.1	NC	While the proposed SR requires that removable contamination on the exterior surfaces meet the limits rather than just the accessible surfaces, there is no change in technical requirements since only accessible exterior surfaces can be surveyed.
SR 4.6.1	SR 3.2.2.1	NC	
SR 4.1.1	SR 3.4.1.1	NC	
SR 4.1.2	SR 3.4.1.2	NC	

Table 7 - Less Restrictive Technical Requirement (LR)

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
Def 1.0.f	1.1 – LOADING OPERATIONS	LR	The CTS definition of loading operations includes "all cask preparation steps". This definition would include steps taken prior to placing fuel into the cask. The revised definition applies when the first fuel assemblies is being placed in the cask and ends when cask is supported by the transporter. Hence the revised definition does not cover activities that would be covered under the CTS definition and is therefore less restrictive. However, prior to placing fuel into the cask, there are no capabilities or performance levels of the cask needed to protect the workers or the public, therefore the proposed change is safe.
Def 1.0.a	Deleted	LR	The term "ADMINISTRATIVE CONTROLS" is a standard industry term and hence the term does not need to be defined in Section 1.1 of the Technical Specifications. Therefore deleting the definition of "ADMINISTRATIVE CONTROLS" does not affect safety.
Def 1.0.b	Deleted	LR	Other than for the title of Section 4.0, the term "DESIGN FEATURES" is not used in these Technical Specifications. Therefore deleting the definition of "DESIGN FEATURES" does not affect safety.
Def 1.0.c	Deleted	LR	The term "FUEL ASSEMBLY" is a commonly used term and therefore does not require a specific definition for the purpose of the ISFSI Technical Specification. Therefore deleting the definition of "FUEL ASSEMBLY" does not affect safety.
Def 1.0.d	Deleted	LR	The term "FUNCTIONAL AND OPERATING LIMITS" is the title of Section 2.0 and is a standard industry term. Hence the term does not need to be defined in Section 1.1 of the Technical Specifications. Therefore deleting the definition of "FUNCTIONAL AND OPERATING LIMITS" does not affect safety.
Def 1.0.e	Deleted	LR	The term "LIMITING CONDITIONS" is part of the title of Section 3.0 and is a standard industry term. Hence the term does not need to be defined in Section 1.1 of the Technical Specifications. Therefore deleting the definition of "LIMITING CONDITIONS" does not affect safety.
Def 1.0.g	Deleted	LR	The term "SURVEILLANCE INTERVAL" is not used in the proposed Technical Specifications. Therefore deleting the definition of "SURVEILLANCE INTERVAL" does not affect safety.

Table 7 Less Restrictive Technical Requirement (LR) – continued

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
Def 1.0.h	Deleted	LR	The term “SURVEILLANCE REQUIREMENTS” is a standard industry term and hence the term does not need to be defined in Section 1.1 of the Technical Specifications. Therefore deleting the definition of “SURVEILLANCE REQUIREMENTS” does not affect safety.
FOL 2.1, Action 1	Deleted	LR	If the inability to meet the cavity pressure limit is due to the vacuum drying system, then it will have to be checked and repaired in order to satisfactorily complete Surveillance SR 3.1.1.1. If the system is not repaired and thus SR 3.1.1.1 cannot be satisfied, LCO 3.1.1 Condition A would be entered and Required Action A.1 will require the cask to be returned to the pool and reflooded. Once the cask is reflooded, the fuel is provided with adequate heat removal and the cask is in a safe condition. Therefore deleting Action 1 does not result in a reduction of safety. However since the specific actions are being deleted, this change was classified as a less restrictive change.
FOL 2.1, Action 2	Deleted	LR	If the inability to meet the cavity pressure limit is due to the cask seals, then they will have to be checked and repaired in order to satisfactorily complete Surveillance SR 3.1.1.1. If the seals are not repaired and thus SR 3.1.1.1 cannot be satisfied, LCO 3.1.1 Condition A would be entered and Required Action A.1 will require the cask to be returned to the pool and reflooded. Once the cask is reflooded, the fuel is provided with adequate heat removal and the cask is in a safe condition. Therefore deleting Action 2 does not result in a reduction of safety. However since the specific actions are being deleted, this change was classified as a less restrictive change.
FOL 2.2, Action 1	Deleted	LR	If the inability to meet the back pressure limit is due to the cask seals, then they will have to be checked and repaired in order to satisfactorily complete Surveillance SR 3.1.2.2. If the seals are not repaired and thus SR 3.1.2.2 cannot be satisfied, LCO 3.1.2 Condition A would be entered and Required Action A.1 will require a helium environment be established in the cask. Once a helium environment is established, the heat transfer will be improved and the cask is in a safe state. If the helium environment cannot be established, then Condition B is entered and cask returned to the pool and reflooded. Therefore deleting Action 1 does not result in a reduction of safety. However since the specific actions are being deleted, this change was classified as a less restrictive change.

Table 7 Less Restrictive Technical Requirement (LR) – continued

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
FOL 2.3, Action	Deleted	LR	The current Action statement only applies “in the event of a cask drop from a height greater than 18 inches” and includes allowances for subsequently transferring fuel that has been inspected to the ISFSI provided it meets the storage requirements. This allowance is redundant with other specifications that contain the requirements for fuel stored in a cask. Since the allowance is redundant, deleting it does not result in a reduction in safety.
Introduction	Deleted	LR	The Technical Specifications are an Appendix to the ISFSI operating license which by definition governs the safety of Prairie Island ISFSI. The current introduction contains no requirements or instructions for control of activities associated with operation of the casks or ISFSI. Therefore deleting the introduction does not affect safety.
Table 3/4-1	Deleted	LR	The operating limits contained in CTS Table 3/4-1 are contained within the specific FOLs and LCOs. Thus the information presented in the table is redundant and therefore deleted. Since the limits are contained within the specific FOLs and LCOs, there is no reduction in safety.
Table 3/4-2	Deleted	LR	The SR frequency requirements contained in CTS Table 3/4-2 are contained within the specific LCOs. Thus the information presented in the table is redundant and therefore deleted. Since the SR frequency requirements are contained within the specific LCOs, there is no reduction in safety.
FOL 2.1, Action if still unable to meet the FOL	LCO 3.1.1, Required Action A.1	LR	If the cask cavity vacuum drying pressure limit cannot be established in the cask, Required Action A.1 would require the cask be placed back in the pool and reflooded. Although this Action does not call for unloading the cask, and hence is considered less restrictive, it will ensure that the fuel cladding is cooled. Therefore replacing the Action in FOL 2.1 with one that calls for reflooding the cask but not unloading it does not result in a reduction of safety.
FOL 2.2, Action if still unable to meet the FOL	LCO 3.1.2, Required Action B.1	LR	If a helium environment cannot be established in the cask, Required Action B.1 would require the cask be placed back in the pool and reflooded. Although this Action does not call for unloading the cask, and hence is considered less restrictive, it will ensure that the fuel cladding is cooled. Therefore replacing the Action in FOL 2.2 does not result in a reduction of safety.

Table 7 Less Restrictive Technical Requirement (LR) – continued

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
LCO 3.6.1 Action	LCO 3.2.2 Required Action A.1	LR	The CTS Action contains 4 separate actions. 1st – Verify correct fuel loading. In the RTS the correct fuel loading has already been verified via SR 3.4.1.1 and SR 3.4.1.2. Hence this action was not included in the RTS 3.2.2 Actions. 2nd – Demonstrate compliance with 10 CFR Part 20 and 10 CFR Part 72. This is retained as Required Action A.1. 3rd – Take appropriate action to comply with acceptable limits. Taking action to satisfy a LCO limit is always an option and does not need to be listed as an option. Hence this Action was not included in the RTS Actions. 4th – If acceptable limits cannot be achieved, the cask shall not be placed in service at the ISFSI. This is equivalent to completing Required Action A.1 prior to TRANSPORT OPERATIONS. Since Required Action A.1 does not explicitly call for the verification of the correct fuel loading, it is considered less restrictive. However, the correct fuel loading has already been verified in RTS Surveillances SR 3.4.1.1 and SR 3.4.1.2 and therefore there is no reduction in safety.
LCO 3.6.1	LCO 3.2.2, & SR 3.2.2.1	LR	The new dose rate limits are based on those associated with the TN-40HT cask. Since the dose analyses for the TN-40HT cask show that the offsite limits are meet with these surface dose rates, verifying that the surface dose rates for a TN-40 cask are less than these limits will also ensure that the offsite doses limits are met. Therefore the proposed surface dose limits do not result in a reduction of safety.
SR 4.3.1	SR 3.1.3.1	LR	While the requirement to perform the leak test in accordance with ANSI N14.5 has been relocated to the bases, and hence is considered less restrictive, there is no intended change in how the test is conducted. Therefore the there is no reduction in safety.

Table 8 - More Restrictive Technical Requirement (MR)

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
LCO 3.1.1(10)	FOL 2.1.e.i, & LCO 3.4.1	MR	To bound the assumptions used in the dose calculations of the TN-40HT casks (and thus have a single specification applicable to both the TN-40 casks and the TN-40HT casks.) the minimum cooling time for a TPD has been increased to 16 years.
FOL 2.1	LCO 3.1.1, & SR 3.1.1.1	MR	Corrected the specification so that the 10 mbar value is applied to the amount of pressure in the cask cavity rather than the amount of vacuum. The new LCO requires that the cask cavity be isolated from the vacuum drying system. Hence this is a more restrictive change.
FOL 2.2	LCO 3.1.2, & SR 3.1.2.2	MR	So that the specification is applicable to both the TN-40 casks and the TN-40HT casks, the allowable low side value has been increased from 19 psia (20 psia minus 1psi) to 19.5 psia. This tightens the allowable range of values and is thus more restrictive. The limit was also converted from psia to mbar using the 68.9 mbar/psi conversion factor.
LCO 3.3.1 Action	LCO 3.1.3 Required Action A.1	MR	While the actions to be taken in the proposed RTS Action are essentially equivalent to those in the CTS, i.e., do what is necessary to establish the leak rate within the limit, the RTS include a required completion time and hence is more restrictive.
LCO 3.8.1 Action	LCO 3.1.4 Required Action A.1	MR	The CTS Action only requires corrective action if a safety function of the cask is impaired. The RTS Required Action requires action regardless of the impact on a safety function and includes a required completion time. Hence the RTS Required Action is more restrictive.
LCO 3.7.1	LCO 3.1.5	MR	RTS SR 3.1.5.1 contains a numerical limit that must be maintained. Hence the LCO it is a more restrictive.
LCO 3.7.1 Action	LCO 3.1.5 Required Action B.1	MR	The CTS Action calls for returning the cask to the Auxiliary building and repairing or replacing seals as necessary. This is the same end state as requiring the cask to be placed in the pool and reflooded, i.e., seals would have to be replaced after reflooding the cask. Hence RTS Required Action B.1 is equivalent to the CTS Action. However, since Required Action B.1 includes a specified completion time it is considered more restrictive.

Table 8 More Restrictive Technical Requirement (MR) – continued

CTS	RTS	Comparison	Additional Notes , Explanations, and Safety Determination Summary
LCO 3.5.1 Action	LCO 3.1.6, Required Actions A.1 & A.2	MR	While the actions to be taken in the CTS Action are essentially equivalent to those in the proposed RTS, i.e., unload the cask and send a report to the NRC, the RTS include a required completion time for removing the fuel and hence is more restrictive. Note that the verification that the fuel assemblies meet the requirements of Specification 2.0 is performed in SR 3.4.1.1 and SR 3.4.1.2
LCO 3.2.1 Action 2	LCO 3.3.1 Required Action A.1 & A.2	MR	Per SR 3.0.1, failure to meet a Surveillance between performances of the Surveillance, shall be a failure to meet the LCO. Hence if it is discovered that the boron concentration has fallen below the limit then Condition A would be entered and Required Actions A.1 and A.2 taken. Required Action A.1 differs from CTS Action 2 in that it only suspends loading of fuel assemblies but would allow unloading to continue. In addition Required Action A.2 would require all fuel to be removed from the cask within 24 hours. The proposed changes result in a safer end state in that the fuel would be removed from the cask and placed back into the racks in the Spent Fuel Pool thus eliminating any criticality concerns within the cask. Since the proposed Actions require removal of fuel from the cask within 24 hours, the new Actions are more restrictive.
LCO 3.2.1	LCO 3.3.1, SR 3.3.1.1, & SR 3.3.1.2	MR	Increased required boron concentration from 1800 ppm, to 2450 ppm. This increase is necessary to ensure that the boron concentration in the pool is equivalent to or greater than that assumed in criticality analysis for the TN-40HT cask. Increasing the required concentration is conservative for the TN-40 casks.
SR 4.7.1	SR 3.1.5.1	MR	SR 3.1.5.1 contains a numerical limit that must be maintained. Hence it is a more restrictive.
SR 4.7.2	SR 3.1.5.2	MR	In addition to requiring an annual test, RTS SR 3.1.5.2 also requires a COT within 7 days of commencing STORAGE OPERATIONS. Hence it is considered more restrictive.
SR 4.5.1	SR 3.1.6.1	MR	SR 4.5.1 requires that the temperature measurement not be taken until 24 hours after completing cask loading, while SR 3.1.6.1 requires 24 hours after commencing of cask draining. Since the commencement of cask drain does not occur for several hours after the fuel has been loaded into the cask, SR 3.1.6.1 provides for time for the cask to heat up and is thus more restrictive than SR 4.5.1.

Table 8 More Restrictive Technical Requirement (MR) – continued			
CTS	**RTS**	**Comparison**	**Additional Notes , Explanations, and Safety Determination Summary**
SR 4.2.1.1	SR 3.3.1.1	MR	Increased required boron concentration from 1800 ppm, to 2450 ppm. This increase is necessary to ensure that the boron concentration in the pool is equivalent to or greater than that assumed in criticality analysis for the TN-40HT cask. Increasing the required concentration is conservative for the TN-40 casks.
SR 4.2.1.2	SR 3.3.1.2	MR	Increased required boron concentration from 1800 ppm, to 2450 ppm. This increase is necessary to ensure that the boron concentration in the pool is equivalent to or greater than that assumed in criticality analysis for the TN-40HT cask. Increasing the required concentration is conservative for the TN-40 casks.

17 MATERIALS EVALUATION

This is a site-specific amendment to change the specifications of the TN40 cask so it can accommodate the higher heat load of high burnup (HBU) fuel from the Prairie Island reactor. The amendment is for storage only. The only change in the cask system will be the structure of the basket. No changes affecting the ISFSI structure, i.e., pad etc., were reviewed. The materials review concentrated on items that might be affected by the higher heat load, higher temperatures, and the inclusion of depleted uranium dioxide replacement rods if necessary.

Other than contents, there are no direct regulatory requirements on the materials. The ability of the package to meet the criticality, structural, thermal, shielding, and retrievability requirements are based on the ability of the materials to function properly within the range of conditions that the package will experience under normal, off-normal, and hypothetical accident conditions. The regulatory requirements, cited below, are the ones that require the proper functioning of the materials.

17.1 CONTENTS

Only four types of unconsolidated, but possibly reconstituted (uranium dioxide, inert, natural uranium dioxide, or stainless steel rods replacing fuel rods), or not damaged Pressurized Water Reactor (PWR) fuel assemblies, subject to restrictions in Technical Specification (TS) 2.1, are allowable contents for the TN-40HT cask:

- Westinghouse 14 x 14 standard
- Exxon 14 x 14 standard (including HBU standard)
- Exxon 14 x 14 TOPROD
- Westinghouse 14 x 14 OFA (including Vantage +)

Burnable poison rod assemblies (BPRA) and thimble plug devices (TPD), constructed of Type 304 stainless steel and Inconel 718 are also permitted. The maximum burnup is 60 GWd/MTU bundle average.

17.1.1 Damaged Fuel

A comprehensive definition of damaged fuel is given in the TS 1.1. Any rods with cladding breaches are considered damaged. Damaged rods are not permitted in the TN-40HT cask. The maximum fuel cladding temperature will not exceed 400 °C (752 °F) for normal operations and 570 °C (1058 °F) for accident conditions. During and after the draining process an inert cover gas will be used at all times. Under these conditions, as delineated in Interim Staff Guidance (ISG) -11, Revision 3, (Cladding Considerations for the Transportation and Storage of Spent Fuel) fuel with zirconium base cladding and a burnup below 62.5 Wd/MTU is not expected to degrade. The staff finds the materials are suitable for meeting 10 CFR 72.122(h)(1).

17.1.2 Characteristics and Properties

All the assembly design characteristics (Table A7.2-1, Table A4.2-26), and rod (and tube) characteristics (Tables A3.1-1, and A3.3-19) have been checked and are within the variability of the various data bases. The initial bow on the assemblies as they come out of the reactor has been confirmed by the staff using proprietary data. For rod hoop stress calculations, the cladding wall thickness in Table A4.2-26 has been reduced by 69 μm (0.0027 in), which is equivalent to the upper limits on the oxide thickness of 120 μm (0.0047 in.).

The modulus of elasticity and yield strength of the Zircaloy-4 cladding shown in SAR, Section A4.2.3.8.3, Table A4.2-25 and A4.2-18, were calculated using the Geelhold-Beyer formula. Although there is limited high burnup data incorporated into the basis for the Geelhood-Beyer formulations, it is accepted by the NRC staff for use in calculating mechanical properties of the cladding as applied for storage. Since there is no consideration of the possibility of hydride reorientation, these properties are not acceptable for transportation calculation if a 10 CFR Part 71 license for this cask is contemplated. The thermal conductivity for the Zircaloy-4 cladding given in SAR, Section A3.3.2.2.3.6.2.2, agrees with the materials properties database (MATPRO) values [4]. The values for Zircaloy-2 are used for all zirconium based alloys. Based on the law of mixture, this is reasonable. The emissivity of the oxidized Zircaloy rods is given in the SAR, Section A3.3.2.2.4.2, as 0.8. This is typical of the emissivity of a thin layer of zirconium oxide [MATPRO] and shouldn't be affected by any further oxidation while in the cask. The thermal expansion, used in various parts of the SAR agrees with the values in MATPRO. The thermal conductivity and specific heat of unirradiated UO_2 (SAR Sec A.3.3.2.2.3.6.2.1) is used in the analysis. The value drops by about 50% for a fuel burnup of 60 GWd/MTU in the temperature range of interest [6]. This drop results in a change of fuel temperature of approximately 0.6 °C (1 °F). The staff finds the materials are suitable for meeting 10 CFR 72.122(h)(1) and
10 CFR 72.124(a).

17.1.3 Source Term

The source terms available for release from a fuel rod are given in SAR, Section A7A.8.6.1. They are consistent with the values recommended in ISG-5 (Confinement Evaluation). The staff finds the materials are suitable for meeting 10 CFR 72.104(1).

17.1.4 Reflood Analysis

A reflood analysis was conducted (SAR Section A4.2.3.9) to evaluate the pressure build-up in the cask and the thermal stress on the cladding. As the water flashes to steam, the pressure will be monitored and vented through a vent port to keep the internal cask pressure below an acceptable 0.5 MPa (75 psig) (SAR Section A3.3.3.3.5.2 and A4.2.3.9). An ANSYS finite element analysis was conducted. Values for the rod dimensions and cladding oxidation wastage were used that maximized the rod stress. A maximum thermal stress of 0.16 MPa (24 ksi) was calculated by the applicant. This is sufficiently below the yield stress range [3] of ~ 0.49 – 0.65 MPa (71 – 92 ksi) at 400 °C (750 °F) acceptable to the staff for storage applications. The staff finds the materials are suitable for meeting 10 CFR 72.122(l).

17.1.5 Drying

Vacuum drying is specified in the TS (SR 3.1.1.1 and B 3.1.1). After draining, the cask is to be evacuated to a pressure ≤ 1 x 10^3 Pa (10 mbar, ~6 torr) absolute and held for ≥ 1.8 x 10^3 s (30 min), after the pump is isolated from the cask. This hold-pressure is slightly higher than that recommended in NUREG-1536, "Standard Review Plan for Dry Storage Cask Systems," but is still considered acceptable. The TS B3.1.2 allows air to be introduced into the cask after draining and prior to vacuum drying. This does not violate the guidance in ISG-22 (Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere during Short-Term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel) since no rods with cladding breaches can be loaded (TS 2.1). The atmosphere during storage will be helium. The staff finds the materials are suitable for meeting 10 CFR 72.122(h)(1).

17.2 CASK

17.2.1 Cask Materials

The cask wall (shell flange) is made of SA-350 Grade LF3 or SA-203, Grade E carbon steel. The inner shell and the bottom plate are made of SA-203, Grade E. The lid outer closure plate is constructed with SA-350 Grade LF3 or SA-203 Grade E. The gamma shield shell and the bottom shield are SA-266, CL2, or SA-516, Grade 70. The lid shield plate is SA-105 or SA-516, Grade 70. The lid closure bolts are of SA-540 Grade B24 steel. The trunnions are of ASME SA-105 or SA-266 Class 2 or 4 carbon steel. All materials used in this system can be subjected to a minimum environmental temperature under normal storage conditions of -40 °C (-40 °F) without adverse affects. All materials of construction are listed on drawings No TN-40HT-72-1. All weld information is listed on the appropriate drawings.

The ultimate strength, yield strength, Young's modulus, and thermal expansion coefficient for the steels used in the cask body, lid and bolts, as a function of temperature, as stated in the SAR Table A4.2-18, were checked against American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Part D, and found to be accurate. The thermal (SAR Table A3.3-8) and mechanical properties (SAR Table A4.2-18) yield and ultimate strengths, Young's modulus, and thermal expansion coefficient conductivity) as a function of temperature for the steels and aluminums used in construction of the basket were found to be correct. These properties were also all checked against ASME B&PV Code, Part D, or MATPRO. All these properties were transferred accurately to other pages and Tables in the SAR where they were used. A hemispheric emissivity of 0.3 for 304 stainless steel was appropriate for high burnup UO_2 fuel. The staff finds the materials are suitable for meeting 10 CFR 72.24(c)(3), 10 CFR 72.122(2)(i), and 10 CFR 72.122(h)(1).

17.2.2 Welds and Codes

The containment vessel is designed, fabricated, examined and tested to the maximum extent possible in accordance with the rules of the ASME B&PV Code, Section III, subsection NB. Material properties from Section II, Part D, are used. The containment boundary welds consist of the circumferential welds attaching the bottom inner plate and the shell flange to the inner shell, and longitudinal weld(s) on the rolled plate, closing the cylindrical inner shell. Weld material conform to NB-2400 and the materials specification requirements of Section III, Part C of ASME B&PV. The containment vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200.

The neutron shield outer shell is designed, fabricated, and inspected, in accordance with the ASME Code, Subsection NF, to the maximum extent possible. The basket is designed, fabricated and inspected in accordance with Subsection NG of the ASME Code to the maximum extent practicable. Structural and structural attachment welds are examined by the liquid penetrant or the magnetic particle method, in accordance with ASME Code, Subsection NB, requirements, and acceptance standards in accordance with Section III, Subsection NF, Paragraphs NF-5340 and NF-5350. The welders and welding procedures are qualified in accordance with Section IX of the ASME Code.

Any exceptions to the ASME codes and alternatives codes are listed by component along with the reference ASME code and section, code requirement and alternatives cited in SAR Sec A3.5). These alternatives are acceptable. The staff finds the materials are suitable for meeting 10 CFR 72.24(c)(4), 10 CFR 72.104(1), 10 CFR 72.150, 10 CFR 72.192, and 10 CFR 72.170.

17.2.3 Fracture Toughness of Ferritic Steel

The cask body and closure lid are ferritic steel and are subject to fracture toughness requirements in order to assure ductility at the lowest service temperature of -29 °C (-20°F). The analysis considers a weld defect of 1.26 cm (0.5 in.) in depth and 12 cm (4.7 in.) in length at 10 critical locations as depicted in SAR Figure A4A.9-1. The calculations show that under both normal and accident conditions the applied stress intensity factors for those weld cracks are below the fracture toughness of the base material, SA-266, Class 2. Thus, they are stable, and won't pose any safety issues.

The fracture toughness of the base metal is not the limiting factor since the cracks are located inside the welds (either at the heat affected zone (HAZ) or at the filler metal). SAR section A9.7.1 indicates that during cask fabrication Charpy Impact testing will be done on the base metals for the TN-40HT shield shell and shield bottom, weld material, and filler material at -29 °C (-20 °F) instead of -40 °C (-40 °F). This is allowable for the use of the cask for storage since, at the time of the movement of the cask, the decay heat will be high enough to keep the material above -29 °C (-20 °F). It should be noted though, that in the future (40 to 60 years), if transport of this cask is contemplated, the decay heat may not be sufficient to support the -29 °C (-20 °F) limit, and further testing of aged material at -40 °C (-40 °F) may be required. The evaluation in the SAR shows that the TN-40HT cask materials meet the fracture toughness criteria of NUREG/CR-3826 (Recommendations for Protecting against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater than 4 inches Thick) and NUREG/CR-1815 (Recommendations for Protecting against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to 4 inches Thick). The staff finds the materials are suitable for meeting 10 CFR 72.122(2)(i).

17.2.4 Gamma and Neutron Shield

Gamma shielding is provided around the inner shell and bottom plate of the containment vessel by an independent shell and bottom plate of carbon steel (SA-516, Grade 70 or SA-266 Class 4). Additional shielding is provided by the stainless steel in the basket. SA-105 is used for the shield plate attached to the lid.

The neutron shielding is provided by a proprietary borated polyester resin compound that surrounds the gamma shield shell and it is subject to thermal and radiation fields during service. These fields have the potential for degrading the properties of the material, including its thermal conductivity. The resin has four components: polyester resin, styrene, aluminum hydrate, and zinc borate. A 10.2 cm (4-inch) thick disk of polypropylene enclosed in steel plates is on the top lid to provide additional neutron shielding during storage. An adequate acceptance plan (SAR section A9.7.7) has been incorporated in the SAR for the neutron shield material.

There should be no radiation stability issues with the polypropylene. This is supported by measurements of a gamma irradiated (50 kGy) tensile bar that was examined after 15 years of ambient storage. A layer of embrittled surface degradation product covered the entire surface, while the core of the sample remained largely intact [12]. In response to an RAI, the applicant suggested that there would be no degradation of the polypropylene since it has been used in other casks for up to 14 years with no reported increase in dose during surveillance monitoring. The validity of this argument is dependent on the total dose seen in those other casks compared to the 40 year dose expected in the TN40HT cask. Since these values are not given, this agreement is not accepted as proof of the radiation stability of the shield material. It does give good support that the acceptance plan need not be in the technical specifications since any problem with the shield material can be readily detected and the cask replaced if necessary.

Additional arguments for radiation stability were made based on the total dose that the shield would see would be less than an Mrad where deterioration might be expected to occur.

The applicant argued that since the shielding is completely enclosed, even if it melted there would be no loss of function. Nevertheless the temperature at the shielding is not expected to be above 149 °C (300 °F), which is 15 °C (27 °F) below the melting point. There is no issue with thermal stability. The neutron shield resin can withstand, without degradation, the maximum temperature of 149 °C (300 °F) it expects to see under normal operation. The staff finds the materials are suitable for meeting 10 CFR 72.24(c)(3), and 10 CFR 72.104(2).

17.2.5 Coatings

The cask cavity surfaces and outer shell have a thermally sprayed metallic coating of Zn/Al for corrosion protection. The low-alloy carbon steel cavity surface is grit blasted prior to coating. During the lifetime of the cask the sprayed coating is exposed to air for a short time, borated water for a short time during cask loading and to a He atmosphere at storage temperatures for an extended period of time. As indicated in the galvanic interaction and gas generation section, no deterioration of this coating is expected due to the short times and low temperatures.

The external cask carbon steel surfaces are painted with epoxy, acrylic urethane, or equivalent enamel coating, for ease of decontamination. These have excellent resistance to salt spray and weathering, and are chip resistant. The staff expects the specified coatings to protect the cask system as specified. The staff finds the materials are suitable for meeting 10 CFR 72.120(d).

17.2.6 Lubricants

Loctite N-5000 Nuclear Grade or Neolube is used on the bolt threads. Loctite N-5000 is a nickel based lubricant made for use with 304 stainless steel. According to the technical data sheet it has very low halide content and an operating range of 129 °C to 1315 °C (264 to 2399 °F). Neolube is a low halide graphite based lubricant made for use on stainless steel and other materials. According to the technical data sheet it has been used in fuel rods and is compatible with UO_2 pellets. It has an operating range of -57 °C to 204 °C (-70 to 400 °F) and can withstand fields up to 1×10^9 rads. This material has an applicable temperature range and compatibility for the designated purpose.

Never-seez will be used to lubricate the trunnions to prevent impregnation of contaminates. The Never-seez compound and Loctite N-5000 compounds have been chosen to be compatible with the trunnions and bolts. The lubricant will be removed from the trunnions prior to insertion of the cask into the pool. The staff finds the materials are suitable for meeting 10 CFR 72.120(d).

17.2.7 Seals

Double metallic O-ring seals of the Helicoflex HND type are used on the lid and the two lid penetrations. The metallic seals have a stainless steel or nickel alloy liner with an aluminum jacket and contain a Nimonic 90 or equivalent material spring. All seating surfaces are stainless steel clad. The lid seal has a long-term operating temperature to 350 °C (662 °F) and can operate up to 550 °C (1022 °F) for short terms before annealing occurs. Viton O-rings are used for the seals in the vent and drainport valves. The minimum radiation dose before radiation effects on Viton occur is 2×10^7 rads [4]. The radiation level at the location of these O-rings is at least two orders of magnitude below this limit so that deterioration and release of fluorine is not expected over the 20 year period. No adverse chemical or galvanic interaction of the seal materials is expected. The staff finds the materials are suitable for meeting 10 CFR 72.104(1).

17.3 FUEL BASKET

17.3.1 Materials and Properties

The basket is constructed of SA 240 Type 304 stainless steel plates, boxes, and rails and 6061-T6 aluminum, according to the ASME B&PV Code, subsection NG, to the maximum extent possible, or approved alternatives. Rectangular stainless steel tubes are joined by a proprietary fusion welding process to stainless steel bars. Above and below the bars are neutron poison plates for criticality control and heat conduction. These boxes are separated by panels consisting of two aluminum plates sandwiching a poison plate. The basket is assembled by passing steel bars through the bounding poison plates and fusion welding to the adjacent box section. Fusion welds between compartments shall be qualified by testing to a margin of safety of 1.43 corrected for temperature differences between testing and operating conditions, and the maximum weld load at any location (SAR section A4.2.3.3.3). The aluminum plate, outer plates, and basket periphery plates are made of SB-209 6061-T651 aluminum alloy. Creep of the aluminum components is not expected to be a problem. According to Section A4B.1.5.6 of the SAR the applicant states "The long term storage load compressive stresses in the limiting aluminum components were compared to allowable stress values that have been reduced to limit the effects due to materials creep."

The thermal conductivity, thermal expansion, and thermal diffusivity of the AL 6061 and type 304 stainless steel used in the basket (SAR pages 3-4) were checked against ASME B&PV Code part D and found to be accurate. The staff finds the materials are suitable for meeting 10 CFR 72.24(c)(3), and 10 CFR 72.124(b).

17.3.2 Neutron Poison

Boral, borated aluminum or boron carbide/aluminum metal matrix composite plates are used for the neutron poison. The applicant takes 90% credit for the B-10 in the B-Al poison plates and 75% credit for the Boral (SAR section A3.3.4.1.4.1.3). The metal matrix composite and the borated aluminum alloy are designated as B-Al. The minimum areal density of the B-10, specified in the TS, Table 4.3-1, is 45 mg/cm^2 for the Boral and 37.5 mg/cm^2 for the B-Al plates.

The qualified neutron poisons are the systems or processes that meet an American Society for Testing of Materials (ASTM) standard that specifies how a product will be made, and that have successfully completed a set of qualification tests for durability and homogeneity. Some of these qualified systems are borated aluminum and boron carbide/aluminum metal matrix composite plates. All three neutron poisons have been previously qualified as neutron poisons for storage casks. An acceptance plan for the neutron poison of choice is given in the Technical Specifications.

The thermal conductivity and specific heat of the neutron absorber plates and solid neutron shield resin given in SAR, Table A3.3-8, are the same values that were used in the analysis of the TN-40, TN-32, and TN-68 casks. No additional confirmation of these values is necessary. The thermal properties of the Boral were used to bound the properties of the metal matrix composite. The staff finds the materials are suitable for meeting 10 CFR 72.124(b).

17.4 GALVANIC INTERACTIONS/GAS GENERATION

The aluminum/zinc coating may react with the borated pool water but does not present any safety issue. The cask is bolted shut and the interior of the cask is vacuum-dried, which would remove any generated hydrogen prior to backfilling with helium for storage. Since it is a bolted closure, analysis [2] has shown that galvanic action and hydrogen generation are insignificant in the TN-40HT cask.

The interior of the cask will be vacuum dried and is backfilled with a helium atmosphere. Due to the lack of moisture and oxygen, the helium atmosphere will not support chemical or galvanic reactions between the steel or aluminum components of the basket or cask, or the Zircaloy components of the fuel assembly [1]. The resin in the radial neutron shielding is fully enclosed in aluminum boxes. The resin itself is inert after curing and does not interact with the aluminum (SAR section A4.2.3.6).

Under storage conditions, the vapor pressure of Zn is low but not negligible. Minimal amounts of Zn would be expected to be deposited on the fuel rods since they are hotter than the cask surface. The potential for stress corrosion cracking exists if the Zn that does deposit on the rods penetrates the cladding grain boundaries. This is highly unlikely since most of the Zn that does deposit would be on the ZrO_2 or CRUD on the rod surface, and the grain boundary diffusion constant of Zn into Zr is low enough that penetration should be limited to only 2.5% at typical storage temperatures [1].

No interaction of exposed fuel with residual moisture is expected since only unbreached fuel rods are approved as allowable contents. The staff finds the materials are suitable for meeting 10 CFR 72.120(d).

17.5 EVALUATION FINDINGS

F17.1 The SAR describes the materials that are used for structures, systems, and components important to safety (SSCs) and the suitability of those materials for their intended functions in sufficient detail to facilitate evacuation of their effectiveness.

F17.2 The selection of materials adequately protects the spent fuel cladding against degradation that might otherwise lead to gross rupture.

F17.3 The storage system employs only noncombustible materials which will help maintain safety control functions.

F17.4 The materials that comprise the storage system will maintain their mechanical properties during all conditions of operation.

F17.5 The storage system employs materials that are compatible with wet and dry spent fuel loading and unloading operations and facilities. These materials are not expected to degrade over time, or react with one another, during any conditions of storage.

18 REFERENCES

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