TN-40HT LAR RAI MEETING

MAY 7TH, 2009

HANDOUT OF

NON-PROPRIETARY RAI RESPONSES AS OF 05/07/09

Revised Technical Specifications

RAIs
ATT 1.1, ED-3, & Other Changes

Revised Technical Specifications

General Technical Specification Format and Content Discussion

The Technical Specifications (TS) proposed in the TN-40HT license amendment request (LAR) are intended to be consistent with the format and content (level of detail) of the Prairie Island Nuclear Generating Plant (PINGP) TS. Plant TS are primarily intended to direct plant operator activities to safely operate the plant. Similarly, the Independent Spent Fuel Storage Installation (ISFSI) TS should direct operator activities to safely store spent fuel within casks and the ISFSI.

The NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors in the Federal Register at 58 FR 39132, issued July 22, 1993, stated:

> "… since 1969 there has been a trend towards including in technical specifications not only those requirements derived from the analyses and evaluation included in the plant's safety analysis report but also essentially all other NRC requirements governing the operation of nuclear power plants. … In the Commission's view, this has diverted both NRC staff and licensee attention from the more important requirements in these documents to the extent that it has resulted in an adverse but unquantifiable impact on safety."

This NRC policy statement endorsed improved Standard TS for each reactor vendor (NUREGs 1430 through 1434) and promulgated 10CFR 50.36 criteria for items which are required to be included in TS. The overall philosophy of the improved Standard TS is that plant safety is improved when operators are directed by TS to focus on matters of safety consequences. The improved Standard TS NUREGs, which have been approved by the NRC, provide TS content guidance and do not explicitly include all aspects of operability in the TS.

ISFSI TS are governed by the requirements of 10CFR 72.44, not 10CFR 50.36. However, limited guidance is provided in 10CFR 72.44 for TS format and content so the proposed ISFSI TS were crafted within the framework of 10CFR 72.44 requirements using other considerations such as, the philosophy for format and content of the plant TS, 10CFR 50.36, and NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance". In general, the proposed TS include activities and variables that direct operators to safely use and handle casks, and store them in the ISFSI. Cask design and fabrication activities and variables can be adequately controlled in the ISFSI SAR and thus were not included in the proposed TS (except for items included in Section 4.0 Design Features consistent with recent precedent).

Revised Technical Specifications

RAI: ATT 1.1

Perform a transient impact structural integrity evaluation, similar to that of Section A4.2.3.8 of the SAR, of the fuel rod cladding for the 18-inch cask handling end-drop accidents, considering the "undamaged fuel assembly" characterized with: (1) uniform rod bowing and (2) missing, displaced, or damaged structural components that can still be handled with normal means.

The applicant defines undamaged fuel assemblies as those with uniform rod bowing and that can be handled by normal means, even if there exist missing, displaced, or damaged structural components. However, since fuel rod buckling performance has not been analyzed for the undamaged configurations described above, a structural evaluation must be included in the SAR to substantiate the subject definition.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: ATT 1.1

Interim Staff Guidance (ISG) -1 Revision 2 provides guidance on the classification of spent nuclear fuel as either: (1) damaged; (2) undamaged; or (3) intact. The guidance provides the flexibility to base the definition on the ability of the fuel assembly to perform fuel-specific and system-related functions rather than specific characteristics of the fuel. The licensee is to perform assessments/analyses of specific characteristics of the fuel, e.g. missing individual fuel rods, to demonstrate that the fuel will still perform the fuel-specific and system-related functions. For licensees that do not wish to perform these assessments and thus not take advantage of the flexibility of the performance-based definition of damaged fuel, the ISG Appendix contains a default definition of damaged spent nuclear fuel.

Since the Prairie Island Nuclear Generating Plant does not currently need the flexibility provided by the performance based definition allowed by the ISG, use of the default definition of Damaged Fuel is proposed. This will be accomplished by revising the proposed SAR Section A3.3.7.1 and the Technical Specifications to not allow the storage of a DAMAGED FUEL ASSEMBLY. The definition of UNDAMAGED FUEL ASSEMBLIES will be changed to DAMAGED FUEL ASSEMBLY consistent with the default definition in ISG-1 Revision 2. Note that while the default definition would allow cladding breaches provided they are not gross breaches, the proposed definition is more restrictive in that any spent fuel assembly that contains cladding breaches of any size would be classified as a DAMAGED FUEL ASSEMBLY and thus not eligible for storage in a TN-40HT cask. This change in the default definition is proposed to address the concern in ISG-22.

Since the Technical Specifications are applicable to both the TN-40 and the TN-

40HT casks, the above changes necessitated changing the definition of UNDAMAGED FUEL ASSMEBLIES in a TN-40 cask to a DAMAGED FUEL ASSEMBLY. To avoid unintentional consequences with changing what is allowed to be stored in a TN-40 cask, the new definition is based on the wording from the current Technical Specification 3.1.1.(5) and 3.1.1.(6).

The changes to the proposed SAR and Technical Specifications address RAI question ATT 1.1. However, with respect to performing a transient impact structural integrity evaluation for fuel assemblies with (1) uniform rod bowing or (2) missing, displaced, or damaged structural components that can still be handled with normal means, the following additional explanation is provided:

- Uniform rod bowing is considered in the proposed SAR Section A4.2.3.8 “Analysis of Fuel Cladding Under Accident Condition Impact Loading”.

- A fuel assembly with missing, displaced, or damaged structural component that adversely affects the analysis in Section A4.2.3.8 would either a) be assumed to adversely affect the radiological and/or criticality safety and thus be classified as a DAMAGED FUEL ASSEMBLY, or b) an evaluation/analysis would be performed to determine if the radiological and/or criticality safety would be adversely affected and the fuel assembly classified accordingly.

Therefore with the SAR and Technical Specification changes described above, additional transient impact structural integrity evaluations are not needed and were not performed.

The following is the proposed definition of a DAMAGED FUEL ASSEMBLY:

In TN-40 casks, a DAMAGED FUEL ASSEMBLY is a spent nuclear fuel assembly that:

a. is a partial fuel assembly, that is, a fuel assembly from which fuel pins are missing unless dummy fuel pins are used to displace an amount of water equal to that displaced by the original pins; or

b. has known or suspected to have structural defects or gross cladding failures (other than pinhole leaks) sufficiently severe to adversely affect fuel handling and transfer capability.

In 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).

RAI: ED-3

In “PI ISFSI Technical Specifications Bases” ANSI 14.5 references are from 1977 and should be updated to be from 1997.

Response: ED-3

The Technical Specification bases for SR 3.1.3.1 will be changed to reference the 1997 version of ANSI N14.5 for the determination of the leak rate.

Revised Technical Specifications

Other TS Related Changes

TS 5.1 General Administrative Controls

Changed “Nuclear Management Company, LLC” to “Northern States Power Company, a Minnesota corporation (NSPM)”.

TS Bases B 2.0 Functional and Operating Limits

Added the following sentence to the APPLICABLE SAFETY ANALYSIS section:

“Reactor coolant radiochemistry data from the fuel assembly’s final cycle of operation, fuel sipping, eddy current exams, or ultrasonic testing may be used to determine that a particular fuel assembly has no cladding breaches.”

TS Bases SR 3.1.2.1, establishing a helium environment in the cask within 34 hours.

Clarified that a “fraction of a mbar” of helium satisfies the helium properties used in the thermal analyses.

TS Bases SR 3.3.1.1, Verifying that boron concentration is ≥ 2450 ppm.

Added a reference to License Condition 15G as the source of the requirement that the chemical analysis is to be performed by two different individuals on two separate samples.

TS Base SR 3.4.1.1, Verifying that the Fuel and inserts meet the loading requirements.

Added a reference to License Condition 15F as the source of the requirement that verification that the fuel meets the loading requirement is to be performed by two independent individuals.

TS Base SR 3.4.1.2, Verifying identity of the Fuel and inserts.

Added a reference to License Condition 15F as the source of the requirement that identity of the fuel assemblies and inserts is to be performed by two independent individuals.

MATERIALS

RAIs
M1 through M19

MATERIALS

RAI: M1

Specify the type of Never-seez to be used for lubrication of the trunnions. Provide justification for the compatibility with borated water and stainless steel. Specify the applicable temperature range of use.

Never-seez comes in a number of different varieties with different preferred applications and recommended environments for use.

This information is needed to determine compliance with 10 CFR 72.122(b)(2), and 72.122(1).

Response: M1

The current cask receipt procedure used at the Prairie Island Nuclear Generating Plant calls for the application of Loctite N-5000 anti-seize lubricant (KMP1FJ and KMP1FK) to the outer shoulder of the upper trunnions, the engagement surface of the lift beam lifting arms and the bearing surface of the lower trunnions prior to rotating and lifting the cask off the rail car. After the cask as been lowered onto the floor and the lifting beam disengaged, the cask receipt procedure calls for removal of the lubricant from the trunnions and the lift beam. Since the lubricant is removed prior to immersing the cask into the spent fuel pool there is no compatibility issue with borated water.

If in the future a lubricant becomes available that is compatible with the spent fuel pool water and the trunnion material, it may only be used after it is approved for that application via the site's formal Chemical Control Program.

There are no SAR or TS changes proposed as part of the response to this RAI question.

RAI: M2

Analyze the potential of a pyrophoric event during the loading, transporting, or unloading of the uranium replacement rods.

SAR Section A3.1.1 indicates "uranium" as a suitable replacement for fuel rods in reconstituted assemblies. The use of uranium requires the analysis of potential interactions and pyrophoric events.

This information is needed to determine compliance with 10 CFR 72.120(d), 72.166, and 72.122(h)(1).

MATERIALS

Response: M2

Section A3.1.1 lists uranium rods as rods that may replace fuel rods in reconstituted assemblies. The uranium rods are not simply rods made from solid uranium. The uranium replacement rods are identical to the other fuel rods except that they are made with natural uranium pellets rather than with enriched uranium fuel pellets. Since the uranium in the replacement rods is contained within the same cladding material and end plugs as any other fuel rod, there is no change in the potential for a pyrophoric event.

To clarify want is meant by uranium rods, the sentence in Section A3.1.1 will be changed to read as follows.

> "Also reconstituted assemblies, (natural uranium replacement rods, Zirconium inert rods, or stainless steel rods replacing fuel rods), may also be stored in the cask."

RAI: M3

Provide copies of the references, or the NRC Agency Document and Management System (ADAMS) accession numbers if relevant, that substantiate the guide tube and instrument wall thickness (Table A7.2-1). Correct guide and instrument tube diameters (Table A3.3-19) to reflect the correct wall thicknesses, if necessary.

Assembly and rod specifications in the tables were reviewed by the staff. While in most cases there was agreement, in some cases the staff identified discrepancies with the staff's reference values (multiple sources). For example, the reviewers' sources indicate a substantially thicker tube wall (0.034 in).

This information is needed to determine compliance with 10 CFR 72.124(a) and 72.11.

Response: M3

The dimensions of the guide tubes and instrument tube do reflect the fuel used at the Prairie Island Nuclear Generating Plant. Appropriate references will be provided.

RAI: M4

Correct or give references for the existing maximum MTU/assembly for the Westinghouse Electric Company (WEC) standard 14 x 14 fuel assembly in Table A7.2-1 and other tables.

This information is needed to determine compliance with 10 CFR 72.124(a) and 72.11.

MATERIALS

Response: M4

The correct value for the loading of a Westinghouse Standard Fuel type should be 0.410 MTU. Table A7.2-1 and Table A3.1-1 will be revised to reflect the correct value.

RAI: M5

State the assumptions with respect to time out of reactor, uniformity of layer thickness, etc. used to determine the quantity of CRUD available to spall.

While the spallation fraction for the CRUD is stated in SAR Section A7A.8.5.1, no values and assumptions are given for CRUD quantities.

This information is needed to determine compliance with 10 CFR 72.126(d).

Response: M5

The radioactive inventory of the CRUD for confinement calculations is obtained from Table 7.1 of ISG-5 Revision 1. Note "#" of this Table provides the value of 140 $\mu Ci/cm^2$ for the CRUD activity per rod for PWR fuel assemblies at the time of discharge. This value is directly employed in the source term calculations. Therefore, no other assumptions were employed in these calculations.

There are no SAR or TS changes proposed as part of the response to this RAI.

RAI: M6

Specify the radiation dose over 20 years at the location of the drain port valve and evaluate its affect on the Viton o-ring.

At significant dose, deterioration of the Viton o-ring may release fluorine into the cask resulting in loss of containment of the Zircaloy cladding. Such degradation would also affect the effectiveness of the seal.

This information is needed to determine compliance with 10 CFR 72.126(d).

Response: M6

Two Viton o-ring seals are employed at the lower end of the adapter fitting in the drain port in the TN-40HT cask. The purpose of these seals is to ensure that the drain port fitting / drain tube junction is able to maintain a seal during the cask draining operation. These seals are not part of the confinement boundary and thus do not perform any confinement function.

The radiation dose rate at the Viton o-rings is not explicitly calculated but may be estimated based on measured dose rate experience at the drain port. Dose rate measurement experience from several loadings of Transnuclear casks indicates that the dose rates at the drain port are less than 1 Rem/hour. Assuming a constant dose rate of 10 Rem/hr for 25 years (the minimum design life of the TN-40HT per SAR Table A3.4-1, 1 Rem/hour = 1 Rad/hour) at the o-ring location, the total exposure is 2.2×10^6 Rads. This estimated exposure is conservative since it does not take into account the exponential decay of the source.

Degradation of Viton is not expected at exposures below 2×10^7 rads. At the total exposure in the range of 10^7 to 10^8 Rads, the Viton o-rings are expected to experience loss of hardness and ductility. It is not until after the o-ring has experienced a loss of hardness and ductility that it will break down to the point that it will release fluorine to the cask cavity. Since the exposure of the Viton o-rings in the TN40HT cask is less than 10^7 rads, no degradation / deterioration that may release fluorine into the cask cavity is expected to occur.

There are no SAR or TS changes proposed as part of the response to this RAI.

RAI: M7

Specify in the Technical Specifications the % credit for the boron-10 for both the Boral and the B-Al alloy.

This information is needed to determine compliance with 10 CFR 72.124(a).

Response: M7

SAR Section A3.3.4.1 states that 90% credit is taken for the neutron poison in the Borated-Aluminum alloy and Aluminum/B4C metal matrix composite materials, and 75 % credit is taken for the presence of neutron poison for Boral® plates. This information may be used by the staff to determine compliance with 10 CFR 72.124(a).

Regulation 10 CRR 72.44(c) requires that Technical Specifications include requirements in the following categories:

- Functional and operating Limits and monitoring instruments and limiting control setting
- Limiting conditions
- Surveillance Requirements

- Design Features
- Administrative controls

A review of these categories, as described in 10 CRR 72.44(c), concluded that the regulation does not require that assumptions used in analyses, e.g. the % credit for the boron-10 in the neutron poison plates, be included in the Technical Specifications. Note that the minimum areal Boron-10 density design feature requirement is already specified in proposed Technical Specification 4.3.

Although NUREG-1745, "Standard Format and content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance", is not directly applicable to site specific Technical Specifications, it was reviewed to determine if it called for the inclusion of the Boron-10 % credit assumption in the Technical Specifications. The review concluded that NUREG-1745 did not call for % credit of Boron-10 to be included in Technical Specifications.

Finally, it is NSPM's understanding that Technical Specifications (both the Prairie Island Nuclear Generating Plant's and the ISFSI's) are to be written focusing on the operational controls, limits and design needed to ensure safe operation (see Technical Specification Content Discussion above). This would not include non-operational assumptions in the safety analyses.

Since the information needed to demonstrate compliance with 10 CRR 72.124(a) is already in proposed SAR Section A3.3.4.1, and for the reasons above, NSPM does not propose to include the % credit for the Boron-10 for both the Boral and the B-Al alloy into the proposed Technical Specifications.

RAI: M8

Include an acceptance plan for the neutron poison plates in the SAR and include it by reference into the proposed Technical Specifications. Correlate the acceptance testing of the neutron absorber with expected performance. Indicate how the acceptance tests indicate an adequate percentage of H and B in the absorber material. Describe the significance of the density measurement, and the sensitivity of measurements to the percentage of critical components (H & B).

This information is needed to determine compliance with 10 CFR 72.124(a).

Response: M8

An acceptance plan will be developed and added to the SAR

MATERIALS

RAI: M9

Provide a reflood analysis.

This information is needed to determine compliance with 10 CFR 72.122(h)(1).

Response: M9

Initially, as pool water is added to the cask cavity containing hot fuel and basket components, some of the water will flash to steam causing internal cavity pressure to rise. The pressure of the cask cavity is monitored to ensure that it does not exceed the design pressure of the cask. The cask pressure is controlled by controlling the reflood rate.

The second paragraph in SAR Section A3.3.2.2.5.2 will be replaced with the following:

> "As pool water is added to the cask cavity containing hot fuel and basket components, some of the water will flash to steam causing the internal cavity pressure to rise. This steam pressure is released through the vent port. The reflooding procedures will require that the pressure be monitored and the reflood flow controlled such that the pressure does not exceed the analyzed internal pressure of 100 psig. To provide margin to the analyzed limit and to account for any pressure drop between the monitoring location and the cask internal pressure, the procedure shall limit the monitored pressure to less than 75 psig."

RAI: M10

Provide an acceptance plan for the neutron shield material. Provide data or analyses to show that the neutron shield material (both resin and polypropylene) will retain adequate properties for the application during the storage period. Include the testing procedure, and data that were collected to determine the maximum temperature that the resin can withstand without degradation. This plan should be included by reference to the SAR in the proposed CoC.

The neutron shield material is a borated polyester resin compound that surrounds the gamma shield shell. It is subject to thermal and radiation fields during service, which have the potential for degrading properties of the material including its thermal conductivity.

This information is needed to determine compliance with 10 CFR 72.126(6).

Response: M10

An acceptance plan will be developed and added to the SAR

MATERIALS

RAI: M11

Provide temperature-dependent fracture property data for the filler metal and the heat affected zone (HAZ) in the temperature range of Hypothetical Accident Condition (HAC) to support the claim that the weld cracks in the base metal of carbon steel (SA-266, Class 2) are stable (SAR Sec A4A.9).

This response should provide justification that any testing, using a limited combination of potential base metals, filler materials, and weld techniques, bounds the worst case fracture toughness expected from all potential combinations of these three parameters. Explain how the TransNuclear (TN) fabricators choose the combinations of weld processes, electrodes and base material to demonstrate the toughness of the weld and HAZ. Defend why any data provided are representative of all other possible combinations which can be used, or are these data the best case scenario?

Various weld techniques, parameters and/or procedural steps can be used to maintain or improve base metal, HAZ, and weld metal mechanical properties. For example, control heat input, bead placement, weld bead type, etc., are such parameters. For any test that results in abnormally high fracture toughness, the response should state the weld parameters utilized in the weld procedure.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

Response: M11

To ensure that the fracture toughness evaluation in Section A4A.9 is applicable to the fabricated casks, a requirement to perform Charpy impact testing on the base metal, weld filler material, and HAZ will be added to the SAR via a new Section A9.7.1.

The following will be added to the SAR:

A9.7.1 Charpy Impact Testing

The base metals for the TN-40HT shield shell and bottom shield shall be subject to Charpy impact testing in accordance with ASME Code (Reference 4) NF-2320 at -20°F during cask fabrication. The acceptance standard shall be a minimum energy absorption of 18 ft-lb.

The weld filler material and Heat Affected Zone (HAZ) shall be subject to Charpy impact testing per ASME Code NF-2431.1(a) through (d), except that:

a) In lieu of the base materials specified for weld test assemblies in the governing weld material specification (SFA), the weld test assemblies for Charpy impact testing shall be prepared using the same base metals that are used for the shield shell and bottom shield.

b) Charpy impact testing shall be performed for both the weld filler material and the heat affected zone of each base metal.

c) The acceptance standard shall be a minimum energy absorption of 18 ft-lb.

References:

4. American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Sections II, III, V, and IX, 2004 edition including 2006 addenda.

 SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing,".

RAI: M12

Justify, as stated in the SAR, that the filler metal is as tough as the base metal (Sec A4A.9.5). Specify the code requirements that the weld filler materials satisfy.

The application provides fracture toughness data of the base metal (SA-266, Class 2) and presumably uses it to show that potential weld cracks in the 10 critical locations remain stable during storage since no fracture toughness for the welds is provided. It is known that mechanical properties of filler material as well as HAZ, in general, can be dramatically different from that of the base metal. It is clear that data of the weld material should be used as the cracks are located within the welds, not in the base metal.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

Response: M12

See Response to RAI M11

RAI: M13

Specify the weld inspection requirements for the fuel basket, and include these requirements in the proposed Technical Specifications.

The staff position is the basket must be inspected per the requirements of American Society of Mechanical Engineers (ASME) Code, Subsection NF, due to the prevalent use of fillet welds, not full penetration welds, as would generally be the case for Subsection NB construction.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

Response: M13

As stated in SAR Sections A3.4 and A4.2.3.3.3, the TN-40HT basket is designed, fabricated and inspected in accordance with the ASME Code Subsection NG to the maximum practical extent. Alternatives to the Code relative to the basket, design, construction, and testing are discussed in SAR Section A3.5.

Note Number 14 on SAR Drawing TN40HT-72-21, Sheet 1 of 7, calls for the seam welds of the fuel compartments to be 100% penetration welds and to meet the requirements of NG-3352.

Note Number 13 on SAR Drawing TN40HT-72-21, Sheet 1 of 7, calls for the capacity of the fusion welds to be demonstrated by qualification and production testing. This alternative to the requirements of NG-3352 is discussed in SAR Section A3.5.

Drawing TN-40HT-72-21 Sheet 2 of 2 shows that the rail assembly welds are groove welds. Notes 6 and 12 on SAR Drawing TN40HT-72-22 Sheet 1 of 2, calls for the welds of the rails to be inspected in accordance for the requirements of Subsection NG.

As shown on the drawings referred to above, welds used to construct the basket are full penetration welds, fusion welds, or groove welds. Thus fillet welds are not used as stated in RAI question M13. Therefore it is appropriate to inspect the welds per the requirements of ASME Code Subsection NG and not per Subsection NF.

The above information may be used by the staff to determine compliance with 10 CRR 72.122(b)(2).

Proposed Technical Specification 4.4 states the following:

> "The TN-40HT basket is design, fabricated and inspected in accordance with Subsection NG of the ASME Code to the maximum practical extent. Exceptions to the Code are listed in Table 4.4-1."

Therefore the proposed Technical Specifications include the weld inspection requirements for the fuel basket.

Regulation 10 CRR 72.44(c) requires that Technical Specifications include requirements in the following categories:

- Functional and operating Limits and monitoring instruments and limiting control setting
- Limiting conditions
- Surveillance Requirements
- Design Features
- Administrative controls

A review of these categories, as described in 10 CRR 72.44(c), concluded that the regulation does not require additional detail on the weld inspection requirements beyond that already provided in the proposed Technical Specification.

Although NUREG-1745 “Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance” is not directly applicable to site specific Technical Specifications, it was reviewed. The review concluded that NUREG-1745 does not call for more detail on the weld inspection requirements than already provided.

Finally, it is NSPM’s understanding that Technical Specifications (both the Prairie Island Nuclear Generating Plant’s and the ISFSI’s) are to be written focusing on the operational controls, limits and design needed to ensure safe operation (see Technical Specification Content Discussion above). This would not include details of the fabrication weld inspection requirements.

For these reasons and since the information needed to demonstrate compliance with 10 CRR 72.122(b)(2) is already in SAR Section A4.2.3.3.3 and the Drawings in SAR Section A1, NSPM does not propose to include additional detail on the weld inspection requirements for the basket welds in the Technical Specifications.

However: NSPM does propose to add the following statement to the SAR in a new Section A9.7.2:

> “Basket welds shall be inspected to the NDE acceptance criteria of ASME Code Subsection NG as described on the drawings in Section A1. Alternatives to the ASME Code are specified in SAR Section A3.5.”

RAI: M14

Specify the acceptance standards or codes for the structural and containment welds. Include these standards or codes in the proposed Technical Specifications.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

MATERIALS

Response: M14

SAR Section A4.2.3.1.1, lists the inspections and codes for inspecting the structural and containment boundary welds. In particular Section A4.2.3.1.1 calls out ASME Code Section III, Subsection NB for the design, fabrication, examination and testing of the containment vessel. It also calls out ASME Code Section III Subsection NF and ASME Code Section V for examination and standards for the other structural and attachment welds.

The above information may be used by the staff to determine compliance with 10 CRR 72.122(b)(2).

Proposed Technical Specification 4.4 states the following:

> "The TN-40HT cask containment boundary is designed, fabricated and inspected in accordance with Subsection NB of the ASME Code to the maximum practical extent. Exceptions to the Code are listed in Table 4.4-1."

Therefore the proposed Technical Specifications include the weld inspection requirements for the structural and containment welds.

Regulation 10 CRR 72.44(c) requires that Technical Specifications include requirements in the following categories:

- Functional and operating Limits and monitoring instruments and limiting control setting
- Limiting conditions
- Surveillance Requirements
- Design Features
- Administrative controls

A review of these categories, as described in 10 CRR 72.44(c), concluded that the regulation does not require additional detail on the weld inspection requirements beyond that already provided in the proposed Technical Specification.

Although NUREG-1745 "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance" is not directly applicable to site specific Technical Specifications, it was reviewed. The review concluded that NUREG-1745 does not call for more detail on the weld inspection requirements than already provided.

Finally, it is NSPM's understanding that Technical Specifications (both the Prairie Island Nuclear Generating Plant's and the ISFSI's) are to be written focusing on the operational controls, limits and design needed to ensure safe operation (see Technical Specification Content Discussion above). This would not include details of the fabrication weld inspection requirements.

For these reasons and since the information needed to demonstrate compliance with 10 CRR 72.122(b)(2) is already in SAR Section A4.2.3.1.1, NSPM does not propose to include additional detail on the weld inspection requirements for the structural and containment welds in the Technical Specifications.

However: NSPM does propose to add the following statements to the SAR in a new Section A9.7.2:

> "The ASME Code qualified materials (i.e. containment boundary) used in the construction of the TN-40HT shall be examined following the requirements of ASME Code Section II. Section V of the ASME Code shall be is used in producing Non-destructive examination (NDE) specifications and procedures. NDE requirements for welds are specified on the drawings provided in Chapter A1. Acceptance criteria are as specified by the governing code. NDE personnel shall be qualified in accordance with SNT-TC-1A, Reference 5.
>
> The confinement welds on the TN40HT shall be inspected in accordance with ASME Code Subsection NB including alternatives to ASME Code specified in SAR Section A3.5.
>
> Nonconfinement welds shall be inspected in accordance with ASME Code Subsection NF including alternatives to the Code as specified in SAR Section A3.5"

RAI: M15

Specify the codes used for welders and weld procedures qualifications. These codes should be placed in the Technical Specifications.

This information is needed to determine compliance with 10 CFR 72.122(b)(2).

Response: M15

SAR Section A4.2.3.1.1, lists the following

> "The welding procedures, welders and weld operators are qualified in accordance with Section IX (and NB-4300 where required) of the ASME Code".

The above information may be used by the staff to determine compliance with 10 CRR 72.122(b)(2).

Proposed Technical Specification 4.4 states the following:

> "The TN-40HT cask containment boundary is designed, fabricated and inspected in accordance with Subsection NB of the ASME Code to the maximum practical extent. Exceptions to the Code are listed in Table 4.4-1."

Since Subsection NB invokes the weld qualifications requirements in Section IX, the proposed Technical Specifications include the welder and weld procedure requirements.

Regulation 10 CRR 72.44(c) requires that Technical Specifications include requirements in the following categories:

- Functional and operating Limits and monitoring instruments and limiting control setting
- Limiting conditions
- Surveillance Requirements
- Design Features
- Administrative controls

A review of these categories, as described in 10 CRR 72.44(c), concluded that the regulation does not require additional detail on the weld procedures and welder qualifications beyond that already provided in the proposed Technical Specification.

Although NUREG-1745 "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance" is not directly applicable to site specific Technical Specifications, it was reviewed. The review concluded that NUREG-1745 does not call for more detail on the weld procedures and welder qualifications requirements than already provided.

Finally, it is our understanding that Technical Specifications (both the Prairie Island Nuclear Generating Plant's and the ISFSI's) are to be written focusing on the operational controls, limits and design needed to ensure safe operation (see Technical Specification Content Discussion above). This would not include details of the fabrication weld procedures or welder qualifications.

For these reasons and since the information needed to demonstrate compliance with 10 CRR 72.122(b)(2) is already in SAR Section A4.2.3.1.1, NSPM does not propose to include additional detail on the weld procedures and welder qualifications requirements in the Technical Specifications.

However: NSPM does propose to add the following statement to the SAR in a new Section A9.7.2:

> "Qualification of welding procedures and welders shall be determined using Section IX of the ASME Code".

RAI: M16

a) Provide a discussion or calculation that shows that the various aluminum alloy canister components will meet their life-time design requirements when operating at temperatures where the material is subjected to creep-induced deformation.

b) Provide or cite references for the long-term creep properties of any aluminum alloy canister component(s) which exceed the stress or temperature limits of the ASME Code, Section II, Part D. Show that these properties are adequate for meeting the component's design-life performance requirements during the specified operating condition(s).

In 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."

This information is required for compliance with 10 CFR 72.122(b)(2).

Response: M16

SAR Section A4B.1.5.6 contains the evaluations/calculations that show that the aluminum components meet their life-time design requirements.

As stated in SAR Section A4B.1.5.6, the allowable stress values are provided in TN Technical Report No. E-25768, "Evaluation of Creep of NUHOMS® Basket Aluminum Components under Long Term Storage Conditions". A copy of this report has already been provided to the NRC via:

> Enclosure 2 to Transnuclear Letter E-25506, "Revision 1 to Transnuclear, Inc. (TN) Application for Amendment 10 to the Standardized NUHOMS® System (DOCKET No. 72-1004; TAC NO. L24052)", dated November 7, 2007

There are no SAR or TS changes proposed as part of the response to this RAI question.

RAI: M17

Justify the use of the hemispherical emissivity of 0.46 for 304 stainless steel, in SAR Section A3.3.2.2.3.6.2.3.

Staff's reference gives a value 0.35 to 0.3 in the temperature range of 200-400 º C.

This information is needed to determine compliance with 10 CFR 72.128(4).

Response: M17

The justification for the hemispherical emissivity of 0.46 for stainless steel has been addressed in the TN response to RAI 3-17 for TN-40 transportation license application in Enclosure 2 of TN E-26726, Reference 1 below.

A copy of SAR Reference 34 of Section A3.6 "*Emissivity Measurements of 304 Stainless Steel*" is also provided in Enclosure 10 of TN E-26726.

Although Reference 34 of SAR Section A3.6 reports the hemispherical emissivity of 0.46 for the temperature range from room temperature to 1100 ºF, an emissivity of 0.3 was conservatively used for the fuel compartments to generate radiation super-element files in the calculation of the transverse effective fuel conductivity described in the SAR Section A.3.3.2.2.3.6.3.1. The use of the 0.3 value is discussed in SAR Section A3.3.2.2.3.6.2.3.

There are no SAR or TS changes proposed as part of the response to this RAI.

References to RAI-M17:

- Transnuclear, "TN Response to RAIs *for TN-40 Transportation SAR* (Docket No.71-9313, TAC No.L24106)", TN E-26726, August 29, 2008.

RAI: M18

Provide references for the thermal characteristics of the neutron shield resins given on page 6 of SAR, Table A3.3-8.

This information is needed to determine compliance with 10 CFR 72.126(6).

Response: M18

The resin used for the radial neutron shield is a proprietary formulation that has been utilized for the TN-40, TN-32 and TN-68 casks which have been licensed for storage. Information on the resin has been provided to the NRC in support of their license applications.

Thermal properties for the neutron shield resin provided in Table A3.3-8 of the SAR are identical to those given in the TN-68 storage UFSAR Section 4.2, Item 5 (Reference 1 below), and the TN-40 transportation SAR Section 3.2, Item 9 (Reference 2 below). The original data can be found in the TN-24 Dry Storage Cask Topical Report (Reference 3 below).

References to RAI-M18:

1) Transnuclear, Inc., TN-68 Storage Cask UFSAR, Revision 4, May 2008, NRC Docket No.72-1027.

2) TN E-23861, "TN-40 Transportation Packaging Safety Analysis Report," Rev. 2, August, 2008, NRC Docket No.71-9313.

 Transnuclear, Inc., TN-24 Dry Storage Cask Topical Report, Revision 2A, 1989, NRC Docket No.72-1005.

Note that while preparing this response, a typographical error was identified in SAR Table 3.3-8 for the thermal conductivity of the Solid Neutron Shield Resin. The value of 0.0833 Btu/hr-in-ºF should be 0.0083 Btu/hr-in-ºF. The SAR Table will be revised to correct this error.

RAI: M19

Provide thermal conductivities for fuel with a burnup of 60 GWd/MTU.

Values of thermal conductivity in SAR section A.3.3.2.2.3.6.2.2.1 are for unirradiated UO2.

This information is needed to determine compliance with 10 CFR 72.128(4) and 72.122(c).

Response: M19

Effects of irradiation on the thermal conductivity of UO_2 are studied by Amaya et al. [1] and Ronchi et al. [2]. Based on the study by Ronchi et al. [2], the thermal conductivity of irradiated UO_2 with ~62 GWd/t and irradiation temperature T_{irr} ≥1300K (average T_{irr} for fuel pellet during irradiation according to Amaya et al. [1]) can drop significantly (more that 50%) compared to un-irradiated UO_2.

Using irradiated UO_2 conductivity decreases the effective fuel conductivity in the transverse direction. Note that as discussed in SAR Section A3.3.2.2.3.6.3.2, the axial effective fuel conductivity is calculated based on the fuel cladding material only and does not include the UO_2 fuel pellet thermal conductivity. Therefore, the axial effective conductivity of the fuel assembly is not impacted.

The thermal conductivity values of UO_2 in SAR Section A3.3.2.2.3.6.2.1 for un-irradiated pellets are compared to the values obtained from the study by Ronchi et al. [2] and are shown in the Figure M19-1. For the temperature range of interest, the comparison shows that the SAR conductivity values are higher by approximately a factor of two compared to the values obtained from Ronchi et al. [2].

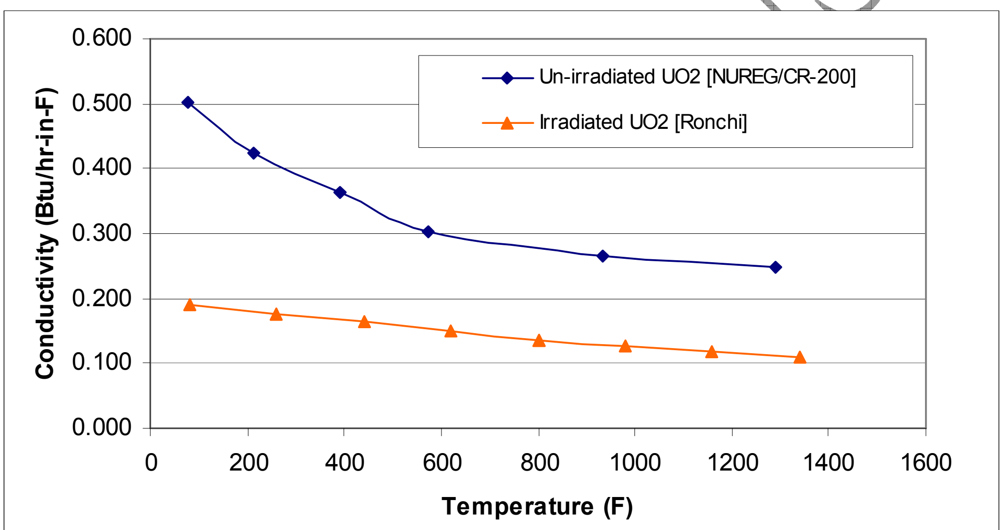

Figure M19–1 UO$_2$ Thermal Conductivity

The transverse effective conductivities for the bounding fuel assembly calculated using unirradiated UO_2 are presented in SAR Table A3.3-9.

A study performed by Transnuclear (TN) and provided to the NRC via an RAI response to NUHOMS® HD System, Amendment 1 [3] shows that the transverse effective fuel conductivity with irradiated UO_2 conductivity is approximately 3% lower than the one with un-irradiated UO_2 conductivity at the operating temperature of 700°F.

The sensitivity runs in the TN study show that the fuel cladding temperature changes by approximately 1°F due to use of irradiated UO_2 conductivity. A 1ºF temperature change is considered to be negligible. These results show that the fuel cladding temperatures are not sensitive to the conductivity of UO_2.

Therefore, use of un-irradiated UO_2 fuel pellet conductivity from NUREG/CR-0200 (SAR Reference 14) is reasonable for irradiated UO_2.

However, it should be noted that the transverse effective fuel conductivities used in the SAR ANSYS thermal models and presented in SAR Table A3.3-8 are at least 20% lower than the calculated transverse effective conductivities presented in SAR Table A3.3-9. This conservatism exceeds any reduction of the transverse effective fuel conductivity due to the effect of fuel pellet irradiation. A comparison between the transverse effective fuel conductivities from SAR Table A3.3-8 and Table A3.3-9 is depicted in SAR Figure A3.3-19. This comparison is discussed in SAR Section A3.3.2.2.3.6.5.

Use of the lower transverse effective fuel conductivity values in the ANSYS model results in higher calculated fuel cladding and basket component temperatures. Therefore, the calculated maximum component temperatures are conservative and the differences in irradiated and un-irradiated UO_2 fuel pellet thermal conductivity values do not affect the thermal analysis results reported in the SAR.

There are no SAR or TS changes proposed as part of the response to this RAI.

References to RAI-M19:

1) Masaki Amaya et al. "Thermal Conductivities of Irradiated UO_2 and $(U,Gd)O_2$ Pellets," Journal of Nuclear Materials, 300 (2002) 57–64.
2) C. Ronchi et al. "Effect of Burn-up on the Thermal Conductivity of Uranium Dioxide up to 100.000 MWd t^{-1}" Journal of Nuclear Materials, 327 (2004) 58-76.
3) TN Letter to NRC, "Revision 1 to Transnuclear, Inc. (TN) Application for Amendment 1 to the NUHOMS® HD System, Response to Request for Additional Information (Docket No. 72-1030; TAC No. L24153)", Enclosure 2, Response to RAI 4.1, TN Document No. E-27377, December 15, 2008.

PRINCIPAL CASK CRITERIA

RAIs
A3.1 through A3.4, & ED-1

PRINCIPAL CASK CRITERIA

RAI: A3.1

NRC staff was unable to locate some general information on the proposed contents. Table A3.1-1 lists some general parameters for each of the approved fuel assembly classes however physical specifications of the assembly are missing, notably maximum assembly weight. Table A3.2-1 lists the presumed weight for all 40 assemblies, but isn't clear if this should be considered an upper bound for any particular class of fuel assembly.

Please present this information in chapter A3 or if located elsewhere in the SAR, indicate within chapter A3 where it may be found.

This information is necessary to verify compliance with 10 CFR 72.11.

Response: A3.1

The maximum weight of a fuel assembly plus an insert is listed as 1,330 lbs in Section A4B.1.3 and proposed Technical Specification 2.1.f. This weight was used to bound all fuel types and thus listing the maximum weight for an individual fuel assembly in Table A3.1-1 is not needed. However; the following paragraph will be added to SAR Section A3.1.1:

> "The maximum combined weight of any fuel assembly and insert is limited to 1,330 lbs and the total weight of all fuel assemblies and inserts is limited to 52,000 lbs."

RAI: A3.2

Table A3.1-1 lists maximum MTU/assembly for the Westinghouse standard assembly as 410 MTU. It is believed the applicant intends this number to be 0.410 MTU and all confirmatory analyses have used this assumption.

Please correct the error in table A3.1-1.

This information is necessary to verify compliance with 10 CFR 72.124.

Response: A3.2

The correct value for the loading of a Westinghouse Standard Fuel type should be 0.410 MTU. Table A3.1-1 will be revised to reflect the correct value.

RAI: A3.3

Provide the time-to-boil calculation for the liquid in the cask during wet fuel transfer operations.

PRINCIPAL CASK CRITERIA

In the Standard Review Plan for Dry Cask Storage Facilities (NUREG 1567) Section 6.5.1.2 states that the applicant should provide a time-to-boil calculation for the loaded cask during transfer operations. This calculation is important to determine if any conditions could exist that might impact the performance of the fuel cladding. The staff did not find the calculation for the time-to-boil in the application. If the time-to-boil calculation was mentioned within the application, provide the appropriate location of this information.

This information is needed to confirm compliance with 10 CFR 72.24(c)(3).

Response: A3.3

The operational sequence for the TN-40HT cask is identical to those described for the TN-40 cask in Section 5.1.1 and 5.1.2. As shown in Table 5.1-1, Steps B.6 through B.11, the cask lid is installed after the fuel assembly loading is completed. The cask is then lifted to the pool surface after completion of the fuel assembly loading and installation of the cask lid. The water in the cask cavity is drained or blown out while the cask body remains partially in the pool. The outer surface of the cask is cooled by pool water during the drainage/blow out operation.

During the short period of drainage/blow out operation, the water in contact with the fuel assemblies within the cask cavity might boil or evaporate. The rising steam condenses at the cask inner walls, or will be vented through the vent port. The hypothetical evaporation/condensation process maintains the temperatures of the components within cask cavity approximately at the saturation temperature of water/vapor. Since the cask is open to the atmosphere through the vent port, the saturation temperature is close to boiling temperature of water at 212 °F.

The criticality evaluations described in Section A3.3.4.1.4.2 are carried out for various moderator densities ranging from 1% to 100% of full density. Most importantly, the calculated effective multiplication factor (k_{eff}) is based on "optimum" moderator density (moderator density where k_{eff} is maximized) thereby inherently including the effects of boiling in the criticality calculations. This implies that the criticality calculations do not require any time limits for boiling and that calculations performed demonstrate sub-criticality when boiling is considered.

The vacuum drying analysis described in Section A3.3.2.2.5.1 considers a conservative initial temperature of 215 °F at the start of water being drained from the cask cavity. This temperature is higher than the expected temperature for the cask content for hypothetical water boiling. Furthermore, the vacuum drying analysis shows that 34 hours after the start of water drainage, the fuel cladding temperature is 725 °F and remains below the allowable limit of 752 °F. As seen, the effects of hypothetical water boiling are considered in the vacuum drying analysis and it is shown that boiling water has no adverse effect on the fuel cladding temperature.

Therefore, the time-to-boil calculations are not necessary.

There are no SAR or TS changes proposed as part of the response to this RAI.

RAI: A3.4

Provide a description of the reconstituted assemblies authorized to be stored in the TN-40HT cask. Include in the description, the enrichment, dimensions, and material of the stainless steel, inert, and uranium replacement rods. In addition, describe how reconstituted assemblies, having uranium rods, were addressed in the shielding evaluation.

Section A3.1.1 states, in part, that reconstituted assemblies (uranium, inert, or stainless steel rods replacing fuel rods) may also be stored in the cask. However, no information (i.e., dimensions, enrichment, etc.) was identified with regards to the uranium rods which may be used. Furthermore, use of the uranium rods was not analyzed in the shielding evaluation.

This information is needed to determine compliance with 10 CFR 72.24.

Response: A3.4

The Prairie Island Nuclear Generating Plant (PINGP) has used natural uranium fuel rods, solid Zirconium inert rods, and solid stainless steel rods to reconstitute fuel assemblies (i.e., replace fuel rods that are damaged). These replacement rods would have the same dimensions as the damaged rods they are replacing such that there is no change in the fuel rod pitch of the assembly. PINGP has always had a very strong fuel integrity program resulting in only a few failed pins within a given fuel assembly, i.e. less than four per fuel assembly.

The natural uranium replacement rods are identical to the fuel pins they are replacing except that natural uranium pellets are used instead of enriched uranium pellets. Since the fuel assemblies will have already seen at least one cycle of operation prior to being reconstituted, the replacement rods will see at least one cycle of exposure less than the damage rods would have seen. Thus, the burnup of the natural uranium replacement pins will be at most 2/3 of the burnup that the damaged fuel pin would have seen. This difference is enough to ensure that the source term of the design basis fuel calculated in Section A7.2 bounds reconstituted fuel assemblies with natural uranium replacement pin(s).

The Zirconium inert rods are solid rods with the same dimensions as the fuel pins they replace. Since the source term due to activation of the Zirconium is much less than the source term of the fuel pin being replaced, the source strength of a reconstituted fuel assembly with the Zirconium inert rod(s) would be bounded by the source term calculated in Section A7.2. Thus the shielding evaluation bounds reconstituted fuel assemblies with Zirconium inert rods.

The stainless steel replacement rods are solid rods made from 304 SS with the same dimensions of the fuel pin they are replacing. The source term (primarily Cobalt-60) for the activated steel pin is greater than what the replaced fuel pin would have been at time of discharge. The decay of the steel rod source term is much greater than the replaced rod. After the specified minimum cooling time of 12 years, the Cobalt-60 activity in the steel rod has decayed to less than ¼ of its original value. This decay is sufficient to ensure that the source strength of the stainless steel replacement pin is bounded by the source term calculated in Section A7.2. Thus the shielding evaluation bounds reconstituted fuel assemblies with stainless steel pins.

The following section will be added to the USAR:

A7.2.7 RECONSTITUTED FUEL ASSEMBLIES

Reconstituted fuel assemblies are fuel assemblies that have replaced damaged fuel pins with either natural uranium replacement rods, Zirconium inert rods, or stainless steel rods. These replacement rods have the same dimensions as the damaged fuel pin being replaced. While lower enriched fuel rods will have a higher source term than higher enriched rods with the same burnup, and activated stainless steel rods will initially have a higher source than a fuel pin due to Cobalt-60, the source term of the design basis fuel described in Section A7.2.1 will bound reconstituted fuel assemblies for the following reasons:

Since the replacement rods will see at least one cycle of exposure less than the damage rods would have, the burnup of the natural uranium replacement pins will be at most 2/3 of the burnup that the damaged fuel pin would have seen. This difference is enough to ensure that the source term of the design basis fuel bounds reconstituted fuel assemblies with natural uranium replacement pin(s).

The source term due to activation of a Zirconium inert rod is much less than the source term would be for the fuel pin being replaced, thus source term of a reconstituted fuel assembly with Zirconium inert rod(s) is bounded by the source term of the design basis fuel.

The source term (primarily Cobalt-60) for the activated steel pin is greater than what the replaced fuel pin would have been at time of discharge. The decay of the steel rod source term is much greater than the replaced rod. After the specified minimum cooling time of 12 years, the Cobalt-60 activity in the steel rod has decayed to less than ¼ of its original value. This decay is sufficient to ensure that the source term of the design basis fuel bounds reconstituted fuel assemblies with natural uranium replacement pin(s).

RAI: ED-1

Clarify the apparent misspelling of the word properties found within the following sections of the SAR.

Within Sections A3.3.2.2.3.2, A3.3.2.2.3.3, and A3.3.3.2.2.3.4 of the SAR, properties was spelled "propoerties".

Response: ED-1

The misspelling of the word properties in the titles for SAR Sections A3.3.2.2.3.2, A3.3.2.2.3.3, and A3.3.3.2.2.3.4 will be corrected.

STORAGE SYSTEM

RAIs
A4.1 through A4.3

STORAGE SYSTEM

RAI: A4.1

Section A4.2.3.3.3, Basket.

Revise the underscored description in the statement, "[T]he required minimum tested capacity of the weld connection shall be based on a margin of safety (test to design) of 1.43 (see Appendix F, Section F-132 (c) of Reference 1), corrected for temperature difference between testing and basket operating conditions and the maximum weld load at any weld location in the basket."

Section F-132 (c) and related margin of safety requirement cannot be found in Appendix F of the ASME code.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: A4.1

The third paragraph of Section A4.2.3.3.3 has been revised to correct a typographical error and to make an editorial change. The ASME reference should have been "Appendix F, Section F-1342(c)".

RAI: A4.2

Table A4.2-2, Containment Vessel Stress Limits.

Revise the table to include also the stress allowable criteria for lid closure bolts as part of confinement boundary of the cask system.

Tables A4.2-2, -3, and -4 presents stress limits for the containment vessel, non-containment structures, and basket, respectively. To meet the 10 CFR 72.122(a) quality standard requirements, the lid closure bolt stress limits, which must be ASME Subsection NB compatible, should also be described in the SAR and tabulated accordingly to facilitate staff safety evaluation.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: A4.2

Table A4.2-2 has been revised to include the following containment bolt stress allowables for both normal and accident conditions. These allowables are from

NUREG/CR-6007.

Containment Bolt Normal (Level A) Conditions[3]	
Tensile Stress, F_{tb}	2/3 S_y
Shear Stress, F_{vb}	0.4 S_y
Combined Stress Intensity, S.I.	0.9 S_y
Interaction limit	$\frac{\sigma_{tb}^2}{F_{tb}^2}+\frac{\tau_{yb}^2}{F_{yb}^2}\le 1.0$
Containment Bolt Hypothetical Accident (Level D)[3]	
Tensile Stress, F_{tb}	Minimum (0.7 S_u, S_y)
Shear Stress, F_{vb}	Minimum (0.42 S_u, 0.6 S_y)
Combined Stress Intensity, S.I.	Not Required
Interaction Limit	$\frac{\sigma_{tb}^2}{F_{tb}^2}+\frac{\tau_{yb}^2}{F_{yb}^2}\le 1.0$

RAI: A4.3

Table A4.2-10, Linearized Stress Evaluation for Normal Condition Load Combinations.

With respect to load combination Case N5, use nodal stress intensities at Nodes 938 and 1218 and any intervening nodes, as appropriate, in an explicit calculation to verify that the stress linearization post-processing is properly implemented for calculating the primary membrane, Pm, and primary membrane-plus-bending, Pl + Pb, stress intensities.

The listed Pm and Pl + Pb stress intensities of 1.98 ksi and 5.67 ksi, respectively, are much smaller than the referenced peak nodal stress intensity of 14.52 ksi. This raises a general concern on whether the ANSYS stress linearization post-processing is properly implemented for the cask body stress evaluation.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: A4.3

With respect to load combination Case N5, the process of nodal stress components combination at Nodes 938 and 1218 and intervening Nodes 936, 1223, 1222, 1221 & 1220 (see Figure A4.3-2, cross section 1) and stress intensity computations are properly implemented by ANSYS postprocessor by following the procedure given in ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Para NB-3215. According to ANSYS postprocessor procedure, stress components Sx, Sy and Sz for normal stresses and Sxy, Syz and Szx for shear stresses (in global or prescribed coordinate system) of individual loads are algebraically combined for the load combination case at all the nodes defining the cross section. Each combined stress component is then linearized to get membrane, bending, membrane plus bending and peak stress categories at the beginning, mid-length and the end of the cross section. Principal stresses S1, S2 and S3 are computed using the membrane, bending, membrane plus bending and peak component stresses. Stress differences S12, S23 and S32 are calculated using these principal stresses and stress intensity S is the largest absolute value of S12, S23 and S31 for membrane, membrane plus bending and peak stresses.

The listed primary stress intensities Pm and Pl + Pb of 1.98 ksi and 5.67 ksi, respectively, at the cross section defined by Nodes 938 and 1218, are much smaller than the maximum nodal stress intensity of 14.52 ksi at Node 938. The reason being that a high stress intensity of 14.52 ksi occurs locally at the corner of the Bottom Shield Plate (see Figure A4.3-1) where the gamma cylinder is contacting. In the remaining large portion of the cross section, stresses are small. The stress linearizing details at this cross section (Figure A4.3-2, cross section 1), using the ANSYS processor, are given in Table A4.3-1. It can be seen that although membrane and membrane plus bending stress intensities are small at top (Node 938) and bottom (Node 1218) of the cross section, a high peak stress intensity of 9.90 ksi occurs at the cross section top and the sum of the membrane plus bending and peak stress intensities is quite close to the maximum nodal stress intensity. This shows that the ANSYS stress linearization post-processing is quite proper.

For further verification, an adjoining cross section defined by Nodes 2393 and 2433 (and intervening nodes) is selected for linearizing (see Figure A4.3-2, cross section 2). The maximum nodal stress intensity at this section is 2.42 ksi. Local stresses at this cross section are expected to be negligible. The linearized stress intensities at this cross section are listed in Table A4.3-2. It is seen that the maximum membrane plus bending and peak stress intensities are calculated as 2.16 ksi and 0.68 ksi respectively and the sum of membrane plus bending and peak stress intensities is quite close to the maximum nodal stress intensity.

It is seen from the above that the ANSYS stress linearization postprocessor has properly implemented the load combination cases for the cask body stress evaluation.

There are no SAR or TS changes proposed as part of the response to this RAI

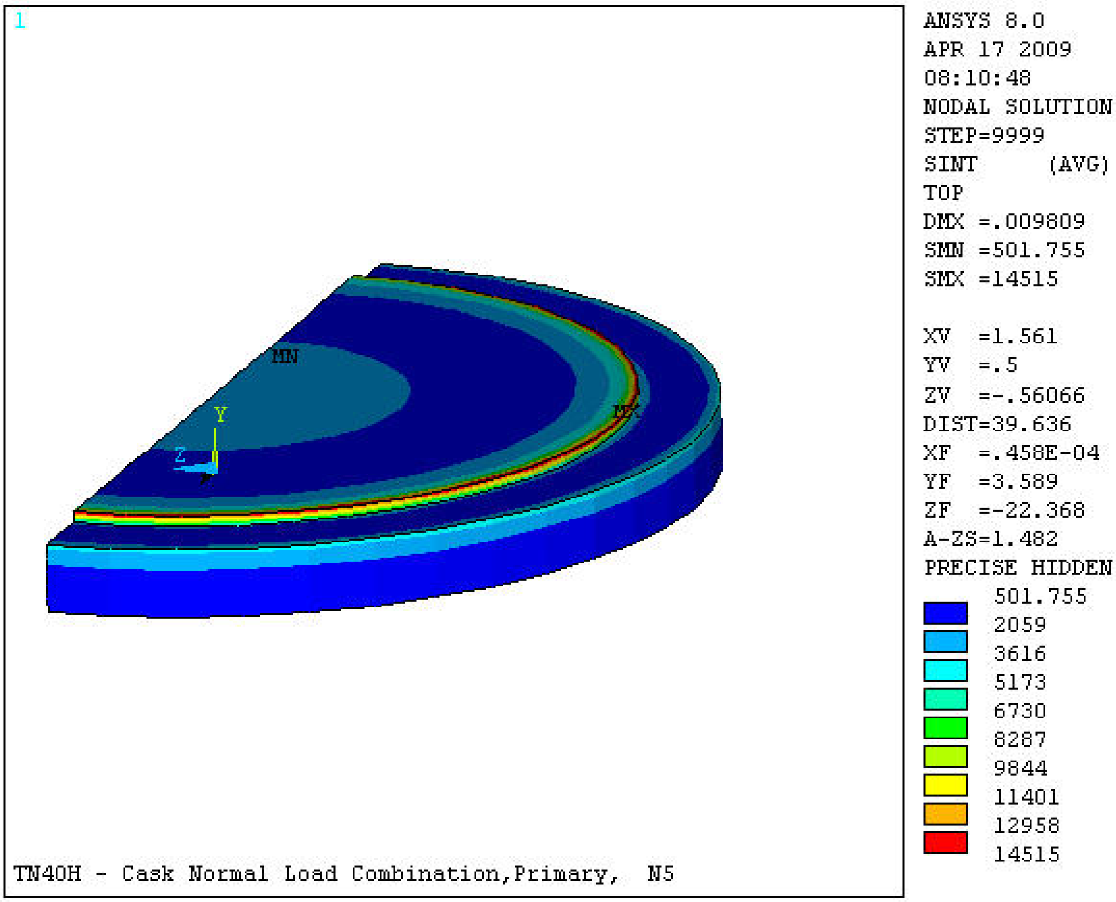

Figure A4.3-1 Bottom Shield Plate, Nodal Stress Intensity Distribution

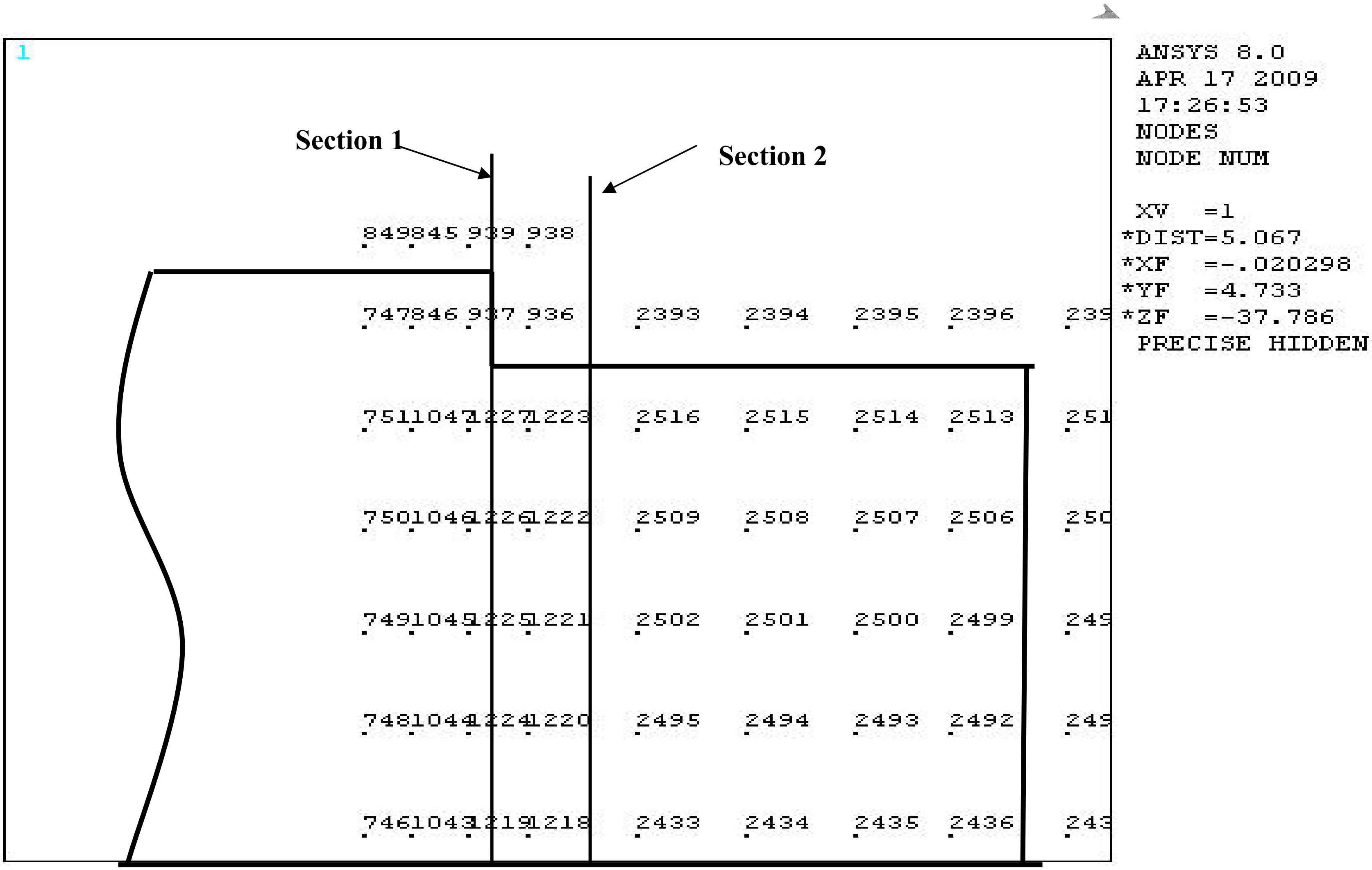

Figure A4.3-2 Bottom Shield Plate, Cross Section Locations

Table A4.3-1 Stress Linearization at Section 1 (Nodes 938-1218)

```
***** POST1 LINEARIZED STRESS LISTING *****
         INSIDE NODE =    938     OUTSIDE NODE =   1218

 THE FOLLOWING X,Y,Z STRESSES ARE IN GLOBAL COORDINATES.


             ** MEMBRANE **
     SX          SY          SZ          SXY         SYZ          SXZ
   688.3       242.5       575.6      0.3821E-02  -974.8     -0.6019E-03
     S1          S2          S3          SINT        SEQV
   1398.       688.3      -579.9       1978.       1735.

             ** BENDING **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY          SZ          SXY         SYZ          SXZ
 I  -90.80      -367.7       2865.     -0.1322E-02  -1230.     -0.5210E-02
 C   0.000       0.000       0.000       0.000       0.000        0.000
 O   90.80       367.7      -2865.      0.1322E-02   1230.      0.5210E-02
     S1          S2          S3          SINT        SEQV
 I   3279.      -90.80      -782.2       4061.       3764.
 C   0.000       0.000       0.000       0.000       0.000
 O   782.2       90.80      -3279.       4061.       3764.

             ** MEMBRANE PLUS BENDING **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY          SZ          SXY         SYZ          SXZ
 I   597.5      -125.2       3440.      0.2499E-02  -2204.     -0.5812E-02
 C   688.3       242.5       575.6      0.3821E-02  -974.8     -0.6019E-03
 O   779.1       610.2      -2289.      0.5143E-02   254.8      0.4609E-02
     S1          S2          S3          SINT        SEQV
 I   4493.       597.5      -1177.       5670.       5023.
 C   1398.       688.3      -579.9       1978.       1735.
 O   779.1       632.4      -2311.       3090.       3020.

             ** PEAK **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY          SZ          SXY         SYZ          SXZ
 I   1308.      -2953.       6871.     -0.3516E-02  -599.2      0.6618E-02
 C  -42.69       414.7      -485.1      0.2367E-02   216.2     -0.6996E-03
 O   87.11      -562.5       704.9     -0.3028E-02  -379.2      0.2002E-02
     S1          S2          S3          SINT        SEQV
 I   6907.       1308.      -2989.       9896.       8595.
 C   463.9      -42.69      -534.3       998.2       864.5
 O   809.7       87.11      -667.3       1477.       1279.

             ** TOTAL **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY          SZ          SXY         SYZ          SXZ
 I   1906.      -3078.     0.1031E+05 -0.1017E-02  -2804.      0.8057E-03
 C   645.6       657.2       90.53      0.6188E-02  -758.6     -0.1302E-02
 O   866.2       47.66      -1584.      0.2115E-02  -124.5      0.6610E-02
     S1          S2          S3          SINT        SEQV         TEMP
 I  0.1087E+05   1906.      -3641.     0.1452E+05  0.1269E+05    0.000
 C   1184.       645.6      -436.0       1620.       1429.
 O   866.2       57.10      -1594.       2460.       2171.        0.000
```

Table A4.3-2 Stress Linearization at Section 2 (Nodes 2393-2433)

***** POST1 LINEARIZED STRESS LISTING *****

INSIDE NODE = 2393 OUTSIDE NODE = 2433

THE FOLLOWING X, Y, Z STRESSES ARE IN GLOBAL COORDINATES.

** MEMBRANE **

SX	SY	SZ	SXY	SYZ	SXZ
816.3	1172.	-172.3	0.3276E-02	-262.6	-0.9086E-03

S1	S2	S3	SINT	SEQV
1221.	816.3	-221.8	**1443.**	1289.

** BENDING ** I=INSIDE C=CENTER O=OUTSIDE

	SX	SY	SZ	SXY	SYZ	SXZ
I	193.7	1672.	1364.	-0.4452E-04	-228.6	-0.7638E-02
C	0.000	0.000	0.000	0.000	0.000	0.000
O	-193.7	-1672.	-1364.	0.4452E-04	228.6	0.7638E-02

	S1	S2	S3	SINT	SEQV
I	1794.	1242.	193.7	1600.	1408.
C	0.000	0.000	0.000	0.000	0.000
O	-193.7	-1242.	-1794.	1600.	1408.

** MEMBRANE PLUS BENDING ** I=INSIDE C=CENTER O=OUTSIDE

	SX	SY	SZ	SXY	SYZ	SXZ
I	1010.	2844.	1192.	0.3232E-02	-491.2	-0.8547E-02
C	816.3	1172.	-172.3	0.3276E-02	-262.6	-0.9086E-03
O	622.6	-499.9	-1536.	0.3321E-02	-33.94	0.6730E-02

	S1	S2	S3	SINT	SEQV
I	2979.	1057.	1010.	1969.	1946.
C	1221.	816.3	-221.8	1443.	1289.
O	622.6	-498.8	-1538.	**2160.**	1871.

** PEAK ** I=INSIDE C=CENTER O=OUTSIDE

	SX	SY	SZ	SXY	SYZ	SXZ
I	261.5	357.2	511.1	-0.1917E-02	91.40	-0.1921E-02
C	-113.7	-346.6	-17.26	0.1252E-02	-25.00	0.1431E-03
O	151.8	575.6	-104.8	-0.1653E-02	-11.42	0.4046E-03

	S1	S2	S3	SINT	SEQV
I	553.6	314.6	261.5	292.1	269.5
C	-15.38	-113.7	-348.5	333.1	296.4
O	575.8	151.8	-105.0	**680.8**	595.5

** TOTAL ** I=INSIDE C=CENTER O=OUTSIDE

	SX	SY	SZ	SXY	SYZ	SXZ
I	1271.	3201.	1703.	0.1315E-02	-399.8	-0.1047E-01
C	702.6	825.3	-189.6	0.4529E-02	-287.6	-0.7655E-03
O	774.5	75.66	-1641.	0.1668E-02	-45.36	0.7134E-02

	S1	S2	S3	SINT	SEQV	TEMP
I	3301.	1603.	1271.	2030.	1886.	0.000
C	901.2	702.6	-265.4	1167.	1081.	
O	774.5	76.86	-1642.	**2417.**	2155.	0.000

STRUCTURAL ANALYSIS OF BASKET

RAIs
A4B.1 through A4B.3

STRUCTURAL ANALYSIS OF BASKET

RAI: A4B.1

Table A4B.1-1, Summary of Individual Loads for Storage Conditions – Basket.

Revise the table, as appropriate, to include also the load case associated with the 18-inch cask handling end-drop accident.

For clarity and completeness, the cask end-drop accident condition, as a licensing basis, should be included in the table to facilitate staff safety evaluation.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: A4B.1

Table A4B.1-1 will be modified to reflect that individual load IL-1 corresponds to the 50g bottom end drop. This change will provide consistency between Table A4B.1-1 and Table A4.2-7.

RAI: A4B.2

Section A4B.1.5.2.1, Finite Element Model Description.

Revise Figures A4B.1-2, A4B.1-3 and add additional sketches to provide sufficiently legible details to depict element types, discretization schemes, and interface as well as boundary conditions, as appropriate, for the structural analysis of the basket subject to lateral loads.

The SAR text and figures are short of necessary details for the adequacy of the basket finite element model.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: A4B.2

The fuel compartment tubes, support plates, and transition rails are modeled with shell elements. The fusion welds that connect the fuel compartments and plates are modeled utilizing pipe elements connected at each end to adjacent fuel compartment boxes. All other interfaces (i.e., between fuel compartments, between fuel compartments and support plates, between fuel compartments and transition rails, and between transition rails and the cask) are modeled by gap elements. For all interfaces through aluminum and poison plates, the plates are assumed to be in contact to simulate support provided by the aluminum and poison plates.

The title of Figure A4B.1-2 will be modified to clarify that it shows the loading orientations only. Figure A4B.1-3 will be modified to the new figure shown below. Figure A4B.1-21 below will be added to clearly depict the element types, discretization schemes, interface, and boundary conditions. In addition to modifying the figures, the text in Section A4B1.5 will be modified to reflect the appropriate figure numbers and the last sentence in the third paragraph in Section A4B1.5.2.1 will be replaced with the following:

> "All other interfaces (i.e., between fuel compartments, between fuel compartments and support plates, between fuel compartments and transition rails, and between transition rails and the cask) are modeled by gap elements. For all interfaces through aluminum and poison plates, the plates are assumed to be in contact to simulate support provided by the aluminum and poison plates."

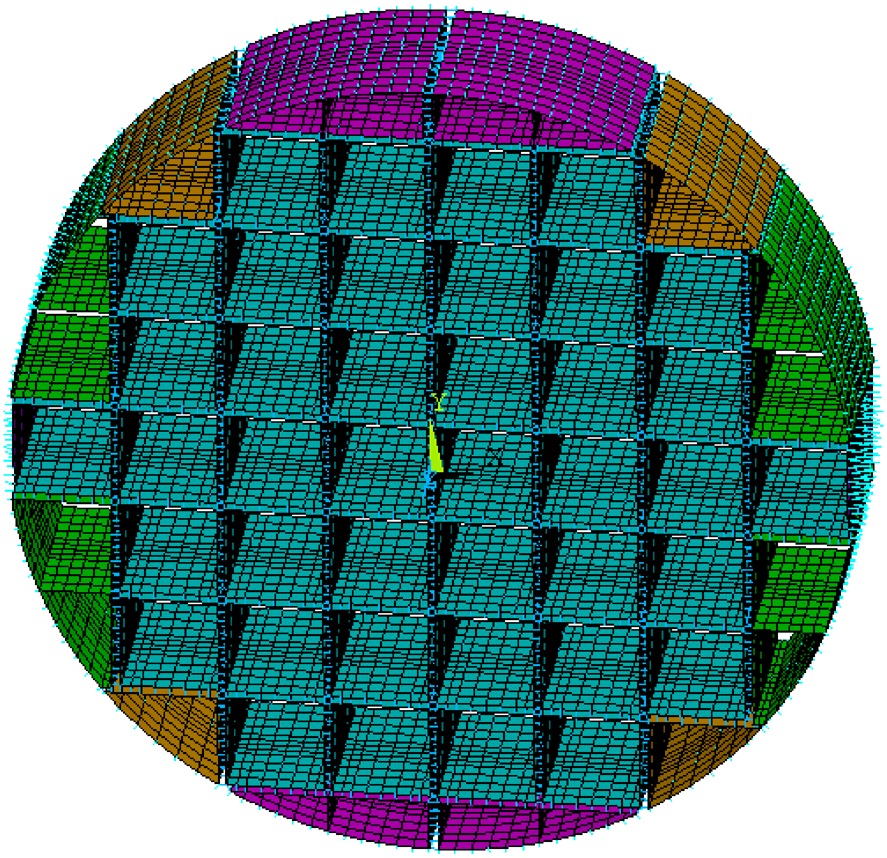

FIGURE A4B.1-3
BASKET FINITE ELEMENT MODEL

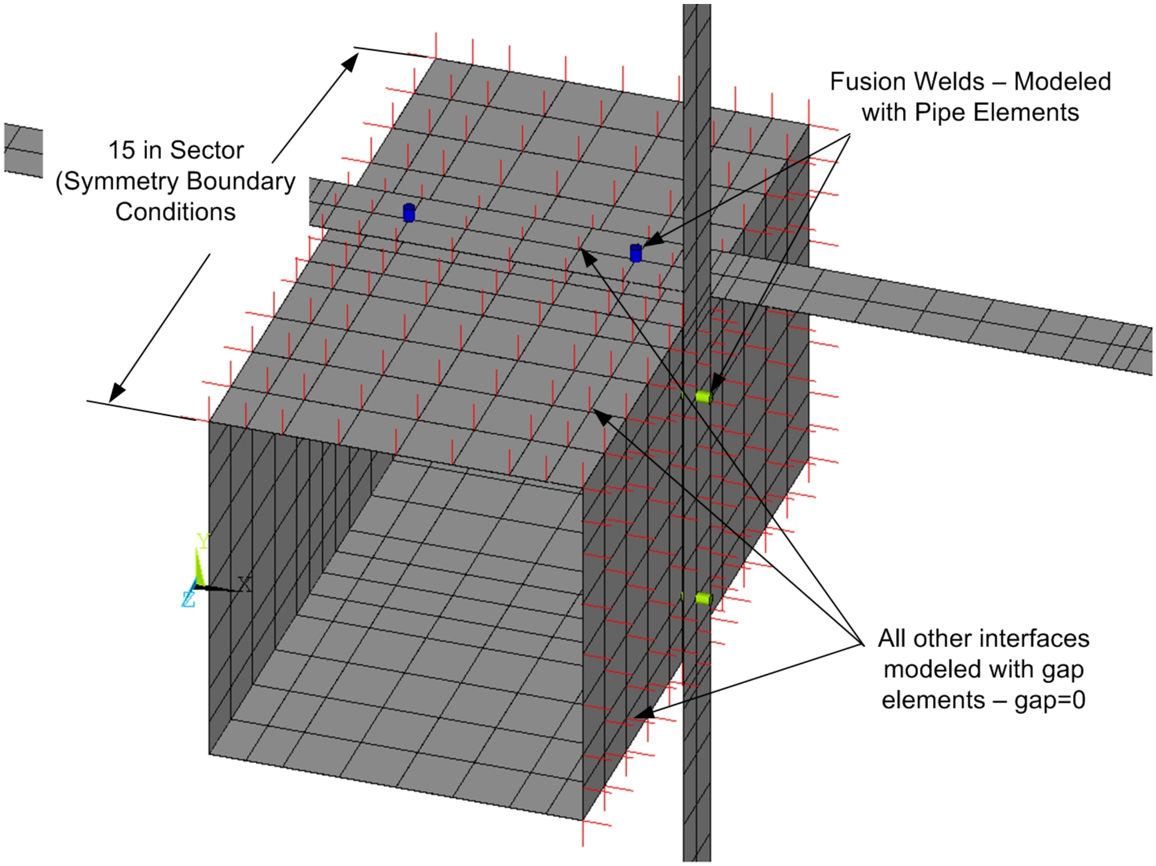

FIGURE A4B.1-21
BOUNDARY CONDITIONS AND INTERFACES FOR TYPICAL FUEL COMPARTMENT TUBE

STRUCTURAL ANALYSIS OF BASKET

RAI: A4B.3

Figure A4B.1-4.

Considering the connectivity between the 1.75-inch wide spacer bar and the fuel compartment walls, provide sketches to illustrate the interface conditions for which the load paths at the nodes other than the fusion weld locations must be properly accounted in the basket structural analysis.

Section A4B.1.5.2.1 of the SAR states: "[t]he strengths of aluminum plates and poison plates in the basket are neglected by excluding them from the finite element model." Properly annotate modeling details are needed to facilitate staff review of the model assumptions made on interface conditions.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: A4B.3

See Response to RAI A4B.2.

PLANS FOR A8.1

RAI
A8.1

PLANS FOR A8.1

RAI: A8.1

Section A8.2.8.2.1, Dynamic Impact Loads.

Considering the approach similar to that for the NUHOMS-HD storage system (Docket 72-1030), perform a transient dynamic impact dynamic analysis of the cask for the 18-inch handling end-drop accident to define applicable loading conditions for cask component evaluations.

A comprehensive review of the EPRI NP-7551 target hardness method and its benchmarking for TN-40HT application may involve long lead-time without certitude for closure. The staff will review other justifiable methods, including the NUHOMS-HD approach, for determining loading conditions for cask components.

The information requested is needed for evaluating the cask for complying with the 10 CFR 72.122(b) requirements for protection against environmental conditions and natural phenomena.

Response: A8.1

NSPM Plans to have TN perform cask drop analysis using the same approach as used for the NUHOMS-HD storage system.

STORAGE SYSTEM OPERATIONS

RAI
A5.1

STORAGE SYSTEM OPERATION

RAI: A5.1

Specify the sensitivity of the cask helium leakage rate test. Also, clarify that monitoring system boundaries are tested to a leakage rate equal to the confinement boundary.

The Technical Specifications should include a minimum test sensitivity of 5 x 10^{-6} atm-cm3/sec for the cask helium leakage rate, consistent with ANSI N14.5-1997. ISG-5, "Confinement Evaluation," states that monitoring system boundaries should be tested to a leakage rate equal to the confinement boundary. The staff could not find where this was described in the application. This information should be provided in the storage system operations or the Technical Specifications.

This information is required to determine compliance with 10 CFR 72.122(h)(4) and 128(a)(1).

Response: A5.1

The content of Technical Specification Surveillance Requirements is to prescribe what must be tested and the appropriate limit. This approach is consistent with that used in the development of the Prairie Island Nuclear Generating Plant Technical Specification. How to perform a Surveillance is to be located in the bases and or the Safety Analysis Report.

Therefore the minimum test sensitivity will be added to SAR Section A7A.8.2 and Technical Specification Base B3.1.3 rather than to the Technical Specifications.

SAR Section A5.1.2 states that the information in SAR Section 5.1.2 is applicable to the TN-40HT casks. SAR Section 5.1.2 refers to Table 5.1-1 for the sequence of operations performed in loading a cask. Step C.15 of Table 5.1-1 already requires that a leak test be performed on the overpressure system (i.e. Drawing TN40HT-72-8 for the TN-40HT cask). Therefore, the SAR already addresses the system boundary for the leak tests.

The following information will be added to the end of last paragraph in proposed SAR Section A7A.8.2. (Note that there were two subsections in A7A.8 numberedA7A.8.1. This has been corrected).

> "… with a minimum test sensitivity of 5 x 10^{-6} atm-cc/sec."

The following information will be added to the end of the second paragraph in the Technical Specification Basses for Surveillance Requirement SR 3.1.3.1.

> "The minimum sensitivity of the leak rate test is 5 x 10^{-6} atm-cc/sec and the test includes the overpressure system up to the isolation valve."

RADIATION PROTECTION

RAIs

A7.2, A7.5, & A7.6

RADIATION PROTECTION

RAI: A7.2

Provide justification for the use of the particular burnup and cooling time values used in the calculation of the inserts. In addition, include information as to whether any downtimes existed between cycles during the overall burnup.

Section A7.2.1 describes the methodology for inclusion of the Fuel Insert Thimble Plug Device (TPD) and the Burnable Poison Rod Assembly (BPRA). The results of the SAS2H/ORIGEN calculations for the TPD and BPRA were included in the results for the design basis fuel gamma source. Staff has some degree of confidence that the TPD burnup was based on the total number of cycles. However, the basis for other relative assumptions (e.g., burnup of the BPRA for 30 GWd/MTU) used in the analysis concerning the TPDs and BPRAs are not discussed.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

Response: A7.2

Two types of inserts (BPRAs and TPDs) will be authorized to be stored along with the spent fuel assemblies within the TN40HT cask. Section A7.2.1 provides the irradiation history and decay time employed to generate the BPRA and TPD source terms to be used in the shielding calculations.

The TPD irradiation history is based on a host assembly burnup of 45,000 MWD/MTU spread equally over three cycles with a 30-day down time between cycles. The resultant source term was increased by a factor of (125,000/45,000) to achieve the equivalent host assembly exposure of 125,000 MWd/MTU. The BPRA irradiation history is based on a host assembly burnup of 30,000 MWD/MTU spread equally over two cycles with a 30-day down time between cycles.

The most important parameters for calculating the insert source term are the material composition and irradiation history. The material composition is specified in Table A7.2-5 and the irradiation history is described on Page A7.2-2. No other assumptions are utilized.

The justification for the burnup and cooling time values used in the insert source term calculations is that these values will represent limiting values for loading of the inserts. These limits have been incorporated into proposed Technical Specification Section 2.1.

It is to be noted that the shielding calculations were carried out assuming that all fuel assemblies would contain inserts. Furthermore, the insert source term was based conservatively on the BPRA source term for the in-core region and the TPA source term for the plenum and top nozzle regions. This is discussed in Section A7.2.4

The description of the TPD and BPRA source terms in SAR section A7.2.1 will be modified to read as follows:

<u>Fuel Insert Thimble Plug Device (TPD)</u>

The TPD materials and masses for each irradiation zone are listed in Table A7.2-5. The TPD is irradiated to an equivalent host assembly life burnup of 125 GWd/MTU. The model assumes that the TPD is irradiated in an assembly with an initial enrichment of 3.85 weight % U-235. The fuel assembly containing the TPD is burned for three cycles with a burnup of 15 GWd/MTU per cycle and a down time of 30 days between cycles. This is equivalent to an assembly life burnup of 45 GWd/MTU over the three cycles. The results are increased by a factor of 2.7778 to achieve the equivalent 125 GWD/MTU source. The source term for the TPD is taken at 16 years cooling time.

<u>Fuel Insert Burnable Poison Rod Assembly (BPRA)</u>

The BPRA materials and masses for each irradiation zone are also listed in Table A7.2-5. These materials are irradiated in the appropriate zone for two cycles of operation. The model assumes that the BPRA is irradiated in an assembly with an initial enrichment of 3.85 weight % U-235. The fuel assembly containing the BPRA is burned for two cycles with a burnup of 15 GWd/MTU per cycle and a down time of 30 days between cycles. This is equivalent to an assembly life burnup of 30 GWd/MTU over the three cycles. The source term for the BPRA is taken at 18 years cooling time.

RAI: A7.5

Identify the localized regions of elevated dose rates due to streaming. Please provide dose rates for vent and drain ports and what methods will be used to ensure doses are maintained ALARA.

In Section A7.4 of the SAR, it states that localized regions of elevated dose rates should be anticipated and minimized with good ALARA practices. Such regions exist due primarily to radiation streaming, including for example, streaming through the vent and drain ports.

Section A1.3.2 states, in part, that penetrations exist for leak detection and venting. There are also vent and drain covers in the steel lid. Staff finding is that no dose rate estimates were identified for those regions where radiation streaming could occur, and no discussion was included detailing what the estimated radiological impacts were as a result.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

Response: A7.5

Prior to the cask draining, the Hansen coupling fitting in the vent port (Item 35 on Section E-E on Drawing TN40HT-72-6) is removed to provide a vent path to the interior of the cask cavity. At the end of the cask draining, the Hansen coupling fitting in the drain port (Item 35 on Section D-D on Drawing TN40HT-72-6) is removed to allow a lance to be inserted into the cask cavity. The lance is used to ensure that all the water has been drained out of the cask. It is with these fittings removed that the highest streaming dose rates will occur directly above the ports. However the streaming has little affect on the general area dose rates around the cask lid and flange area. During subsequent loading steps, workers will reinstall the Hansen fitting into the vent port, install/remove the vacuum drying fitting, install/remove the helium backfilling fitting and install the port covers. While these evolutions do require workers to “reach” over the ports, they do not require the workers to place their whole bodies over the ports. Thus, the dose is limited to the hand and arm extremities. Prior to these evolutions, the Radiation Protection department will perform a pre-job brief with the workers. This brief will include a discussion on these higher dose rate areas and will remind workers to minimize the time needed to perform the evolutions above the ports. During periods where work is not being performed on the ports and until the ports are covered, the Radiation Protection Department will place temporary shielding over the ports with instructions that it is not to be removed without Radiation Protection permission. These ALARA practices minimize any worker dose resulting from the streaming from the vent and drain ports. Once the port covers and top neutron shield have been installed (note that the top neutron shield will cover the ports), the dose rate above these locations is reduced and streaming is no longer a concern.

In the final configuration of the cask, the Hansen fitting in the vent port, the adapter fitting in the drain port, the port covers, and the top neutron shield all provide shielding. Thus any radiation streaming from the ports is reduced to the point where it would have a negligible effect on the offsite dose. Even if a calculation of the effect that any streaming would have on the offsite dose were to be attempted, there is reasonable assurance that the results would not increase the current calculated dose to the nearest real individual (2.20 mrem per SAR Section A7.5) to a point that would challenge the 25 mrem limit cited in 10 CFR 72.104.

Because workers are protected from the impact of radiation streaming out the ports during cask loading by ALARA practices and there is reasonable assurance that the offsite doses will remain below the regulatory limits, NSPM does not see the need to attempt to quantify dose rates due to streaming from the vent and drain ports. Therefore no dose rates are provided and no SAR changes were identified.

RAI: A7.6

Provide confirmation as to whether fuel assemblies authorized for storage in the TN-40HT cask include natural uranium blankets.

Section A7.2.1 of the SAR provides information used in determining the neutron and gamma source terms. It states, in part, that the fuel assemblies acceptable for storage in the TN-40HT cask are listed in Table A3.1-1. Table A3.1-1 provides some detail about the authorized assemblies but gives no indication that natural uranium blankets were used with these assemblies.

From information found in a separate SAR, natural uranium blankets were used for fuel authorized for the TN-40 transportation package.

Confirmation is needed for fuel assemblies authorized to be stored in the TN-40HT cask.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

Response: A7.6

The fuel assemblies authorized for storage in the TN-40HT cask include natural uranium blankets. The presence of blankets (regions of lower enrichment) at the axial ends of the fuel assembly could result in small changes to the axial shape of the fuel assembly neutron and gamma source distribution. Depending on the enrichment of the blanket regions, the source distribution is likely to be slightly depressed at the axial ends and slightly more peaked at the central regions of the fuel assembly. However, this is likely to be conservative since, the maximum dose rates on and around the TN-40HT casks are shown (Table A7A.2-1 and Table A7A.5-1) to be in the vicinity of the top and bottom ends of the fuel assembly and not confined to the central region. Therefore, the presence of axial blankets may result in a slight reduction in the maximum dose rates on and around the TN-40HT cask.

Regardless of the presence or absence of axial blankets, the proposed Technical Specification 3.2.2 “Cask Dose Rates” will provide the necessary radiological protection and assurance that the SAR calculated dose rates bound the loaded cask.

To ensure that the appropriate assembly enrichment (with/without the presences of blankets) is utilized when determining the assembly decay heat, and for determining the allowed burnup values, Technical Specification 2.3 will be modified to clarify that the initial assembly average enrichment is to be used. This will ensure that a conservative value of enrichment will be employed for fuel assemblies containing blankets.

The following shows the proposed TS changes, note the changed wording is in bold:

2.3 Additional Fuel Characteristics for Fuel Stored in a TN-40HT Cask

a. The initial enrichment shall be ≤ 5.0 weight percent U-235;

b. The assembly average burnup shall be:

Initial percent U-235 (%)	Assembly Average Burnup (MWd/MTU)
Average Enrichment < 3.4	≤ 44,000
3.4 ≤ **Average** Enrichment ≤ 5.0	≤ 60,000

c. The cooling time prior to loading shall ≥ 12 years;

d. The combined heat load of an assembly and any associated BPRA or TPD shall be ≤ 800 Watts. The following formula shall be used to determine the heat load of an assembly:

$$\text{Heat load} = F * e^{\left(-0.309*\left(1-\frac{12}{C}\right)*\left(\frac{C}{B}\right)^{0.431}*\left(\frac{E}{B}\right)^{-0.374}\right)}$$

Where :

$$F = 18.76+(11.27*B)+(6.506*E)+(0.163*B^2)+(-1.826*B*E)+(6.617*E^2)$$

B is the assembly average burnup in GWd/MTU

E is initial **average** enrichment in wt. % U-235

C is cooling time in years

PLANS FOR

RAIs

A7.1, A7.3, & A7.4

PLANS FOR A7.1, A7.3, & A7.4

RAI: A7.1

Provide information regarding burnup, enrichment, cooling time combinations for other candidate fuel assemblies authorized for storage in the TN-40HT cask.

Section A7.2.1 states that the 14x14 Westinghouse standard is the design basis fuel for shielding purposes because it has the highest initial metal loading and therefore results in the highest radioactive source terms for a given irradiation history. This includes a burnup, bundle average enrichment, and cooling time of 60 GWd/MTU, 3.4 wt% U-235, and 18-year cooling time, respectively. It is also noted that CRUD is maximized at the minimum cooling time of 12 years.

Section A3.1.1 states, in part, that fuel with various combinations of burnup, enrichment, and cooling time can be stored in the TN-40HT cask as long as the combination results in decay heat, surface dose rates, and radioactive sources for confinement that are bounded by the design basis fuel.

The SAS2H evaluation yielding the bounding source terms used for shielding and confinement were taken for what was identified as the design basis fuel. Additional information is required to justify the licensee's selection of the initial enrichment wt.% U-235, burnup, and cooling time combination as having the bounding parameters for the shielding and confinement analyses.

This information is needed to determine compliance with 10 CFR 72.24 and 10 CFR 72.104(a).

Response: A7.1

The fuel qualification for the shielding evaluation of the TN-40 HT cask is described in Section A7.2.6. As described in Section A7.2.1, the Westinghouse standard 14x14 fuel design is selected as the design basis fuel because it contains the highest initial heavy metal loading.

For the purpose of fuel qualification, it is sufficient to demonstrate that the burnup, enrichment and cooling time combinations for the design basis fuel assembly are selected such that the resulting source terms are bounding for shielding and containment calculations. For a given burnup and cooling time, the fuel assembly with the lowest enrichment will result in a more limiting radiation source terms. For the TN-40HT cask, a minimum enrichment 3.4 wt. % U-235 was selected when burnup could be as high as 60,000 MWd/MTU. Due to the limitation in the minimum cooling time of 12 years and a maximum decay heat of 800 watts, it is sufficient to evaluate only a few burnup and enrichment combinations for the purpose of fuel qualification.

The fuel qualification methodology is described in detail in Section A7.2.6 of the

SAR. A response function based on a simplified representation of the TN-40 HT cask is employed for this purpose. The response function is utilized to determine the dose rate at 2m from the surface of the TN-40 HT cask for the candidate burnup and cooling time combinations as described above. The design basis fuel assembly parameters are then selected based on the combination that resulted in the highest calculated dose rate. Based on the results of this evaluation, the design basis source terms for shielding are obtained from the Westinghouse 14x14 standard fuel assembly with an enrichment of 3.40 wt. % U-235, a burnup of 60,000 MWD/MTU and a cooling time of 18 years.

The cooling time for calculating the CRUD source term for confinement is independent of the spent fuel parameters as discussed in Response to RAI M5.

The SAR section A7.2.6 will be modified to provide additional details about the response function and the dose rate ranking calculations. The following text will be added after the fourth paragraph.

> *The response function is shown in Table A7.2-10. As described above, the response function for neutrons and secondary gamma is a total source to dose factor while that for the primary gamma is a function of the energy spectrum. Table A7.2-10 also provides the additional dose rate contribution from the active fuel portion of the BPRA. A comparison of the neutron, gamma and total dose rate results for the design basis fuel based on the response function and the calculational MCNP results (mid-plane average from Table A7A.5-2) indicates that the response function results are adequate (ratio of neutron to gamma) for the purpose of fuel qualification (relative comparison of source terms).*
>
> *The response function is employed to determine the design basis spent fuel parameters from among seven limiting combinations of burnup and cooling time (BECT). These combinations are selected such that the resulting decay heat is greater than the maximum allowable decay heat of 800 watts per fuel assembly.*
>
> *Four sets of calculations (A, B, C and D) are performed to determine the design basis spent fuel parameters by a comparison of the resulting response function dose rates for the combinations of spent fuel parameters.*
>
> *The results of these calculations are shown in Table A7.2-11. Cases A1 through A7 show the results of the response function dose rate calculations for the seven limiting BECT combinations. These calculations show that Case A7 results in the highest dose rate. Cases B1 through B8 show the results of the response function dose rate calculations for eight*

BECT combinations with a decay heat of approximately 800 watts per fuel assembly. Cases B1 through B8 represent the actual BECT combinations (fuel that would more closely qualify for loading) while A1 through A7 represent conservative combinations. As expected the dose rates for the cases B1 through B7 are lower than those for cases A1 through A7. Based on the results of this evaluation, the design basis source terms for shielding are obtained conservatively from the Westinghouse 14x14 standard fuel assembly with an enrichment of 3.40 wt. % U-235, a burnup of 60,000 MWD/MTU and a cooling time of 18 years.

Cases C1 through C8 are sensitivity calculations where the soluble boron concentration is increased from 600 ppm to 1000 ppm. A boron concentration of 1000 ppm averaged over the entire depletion is a conservative representation of the boron concentration during actual depletion. The results of these evaluations show that the increase in boron concentration results in an increase in the dose rate by approximately 1.5% and an increase in the decay heat by approximately 1%.

Cases D1 through D8 are sensitivity calculations where the moderator temperature is increased from 558 K (545ºF) to 590 K (602ºF, representative of an average hot leg moderator temperature) and the moderator density is correspondingly reduced from 0.733 g/cm^3 to 0.690 g/cm^3. The soluble boron concentration is maintained at 1000 ppm, similar to that of the previous sensitivity evaluation. The results of these evaluations show that the increase in moderator temperature and soluble boron concentration results in an increase in the dose rate by approximately 4% and an increase in the decay heat by approximately 2%.

However, a comparison of the results from the A, B, C and D cases demonstrate that the highest calculated dose rate is obtained from Case A7. Therefore Case A7 represents the design basis case from a fuel qualification standpoint.

PLANS FOR A7.1, A7.3, & A7.4

Table A7.2-10 Response Function for TN-40 HT Cask

	Response Function ((mrem/hour) per particle)
Neutron	5.38E-09
Secondary Gamma	2.32E-08

Primary Gamma Energy Range (MeV)	Response Function ((mrem/hour) per particle)
0.40 to 0.60	8.11E-18
0.60 to 0.80	7.86E-16
0.80 to 1.00	5.39E-15
1.00 to 1.33	4.03E-14
1.33 to 1.66	1.84E-13
1.66 to 2.00	5.73E-13
2.00 to 2.50	1.57E-12
2.50 to 3.00	3.43E-12
3.00 to 4.00	7.32E-12

Dose Rate from BPRA	0.29 mrem/hour

Calculational Model	Neutron (mrem/hour)	Gamma (mrem/hour)	Total Dose (mrem/hour)
Response Function	13.52	11.07	24.59
Response Function (BPRA)	13.52	11.36	24.88
TN-40 HT Shielding	10.10	9.10	19.20
Ratio	0.75	0.80	0.77

PLANS FOR A7.1, A7.3, & A7.4

Table A7.2-11 Fuel Qualification Calculations for TN-40 HT Cask

Case	Burnup (GWD/MTU)	Enrichment (wt.% U-235)	Cooling Time (years)	Decay Heat (watts)	Dose Rate (mrem/hour)		
					Neutron	Gamma	Total
	Design Basis Cases for Fuel Qualification						
A1	52	3.4	12.2	813	9.82	14.55	24.36
A2	53	3.4	12.8	817	10.31	14.09	24.40
A3	56	3.4	14.9	829	11.77	12.65	24.42
A4	57	3.4	15.6	835	12.25	12.27	24.52
A5	58	3.4	16.4	838	12.68	11.82	24.50
A6	59	3.4	17.2	841	13.10	11.43	24.53
A7	60	3.4	18.0	844	13.52	11.07	**24.59**
	Fuel Qualification for Decay Heat of 800 Watts/Assembly						
B1	52	3.4	12.7	798	9.63	13.80	23.43
B2	53	3.4	13.5	799	10.05	13.12	23.17
B3	56	3.4	16.1	803	11.25	11.36	22.62
B4	57	3.4	17.0	805	11.63	10.88	22.50
B5	58	3.4	18.1	801	11.90	10.28	22.18
B6	59	3.4	19.1	802	12.21	9.84	22.05
B7	60	3.4	20.2	800	12.45	9.37	21.83
B8	60	4.9	18.0	798	7.46	10.05	17.52
	Sensitivity - Soluble Boron Concentration of 1000 ppm						
C1	52	3.4	12.7	806	9.86	13.85	23.72
C2	53	3.4	13.5	806	10.27	13.18	23.45
C3	56	3.4	16.1	811	11.48	11.42	22.90
C4	57	3.4	17.0	812	11.85	10.94	22.79
C5	58	3.4	18.1	811	12.12	10.34	22.46
C6	59	3.4	19.1	811	12.42	9.90	22.32
C7	60	3.4	20.2	809	12.66	9.44	22.10
C8	60	4.9	18.0	806	7.65	10.16	17.81
	Sensitivity - Moderator Temperature of 590 K						
D1	52	3.4	12.7	818	10.26	14.01	24.27
D2	53	3.4	13.5	818	10.67	13.33	24.01
D3	56	3.4	16.1	824	11.87	11.57	23.45
D4	57	3.4	17.0	825	12.24	11.09	23.33
D5	58	3.4	18.1	823	12.50	10.49	22.99
D6	59	3.4	19.1	823	12.81	10.04	22.85
D7	60	3.4	20.2	821	13.04	9.58	22.62
D8	60	4.9	18.0	818	8.01	10.36	18.37

PLANS FOR A7.1, A7.3, & A7.4

RAI: A7.3

Provide justification supporting your use of a lower boron concentration. Has a comparative analysis been performed on the change in boron concentration between 900 ppm and 600 ppm?

Section A7.2.1 states, in part, that typical cycle average boron concentration is on the order of 900 ppm. It also states that for modeling purposes in the current analysis, 600 ppm was chosen to be the average boron concentration for the first irradiation cycle, with the second having 95% of this value. The SAR makes reference in the paragraph that there is essentially no effect on dose rates and cooling times based on certain studies which were not discussed in adequate detail or referenced in the SAR. It is also stated in the discussion that "studies were performed showing that the use of a lower boron concentration leads to a tiny underproduction of decay heat, neutron and gamma source strength in the energy groups that contribute the most to casks dose rates." The "studies" discussed in the paragraph provided no direction to sources supporting the use of a lower boron concentration.

This information is needed to determine compliance with 10 CFR 72.24.

Response: A7.3

The SAR section A7.2.6 will be expanded to include details of the fuel qualification calculations to determine the design basis fuel assembly parameters for shielding as part of response to RAI A7.1.

A sensitivity analysis is included in Section A7.2.6 that determines the effect of the soluble boron concentration on the decay heat and source strength of the fuel assembly. These results demonstrate that the increase in boron concentration (from 600 ppm to 1000 ppm) results in an increase in the dose rate by approximately 1.5% and an increase in the decay heat by approximately 1%.

The fuel qualification, however, ensures that the design basis fuel assembly utilized in the shielding calculations results in bounding dose rates even though the boron concentration utilized is lower than that of a typical cycle average value (see Response to A7.1).

Section A7.2.1 of the SAR (Page A7.2.3) that discusses the "Reactor Coolant System Boron Concentration" will be modified to include the results of the sensitivity evaluation discussed above. The following text will be added after the second paragraph.

The results of the fuel qualification sensitivity calculations with soluble boron concentration are shown in Table A7.2-11. Cases C1 through C8 *are sensitivity calculations where the soluble boron concentration is increased from 600 ppm to 1000 ppm. The results of these evaluations show that this increase in boron concentration results in an increase in the dose rate by approximately 1.5% and an increase in the decay heat by approximately 1%. The fuel qualification, however, ensures that the design basis fuel assembly utilized in the shielding calculations results in bounding dose rates even though the boron concentration utilized is lower than that of a typical cycle average value.*

RAI: A7.4

Provide your technical justification for the use of 566 ° F as the moderator temperature.

The SAR states that moderator temperatures can vary between 500 – 600 ° F. The SAR states that a higher average moderator temperature results in increased epithermal absorption in U-238, which results in an increase in the actinide inventory in the fuel for a given total fuel burnup. The SAR states that a moderator density corresponding to a temperature of 566 ° F was used in the SAS2H calculation.

This information is needed to determine compliance with 10 CFR 72.24.

Response: A7.4

The SAR section A7.2.6 will be expanded to include details of the fuel qualification calculations to determine the design basis fuel assembly parameters for shielding as part of response to RAI A7.1.

A sensitivity analysis is included in Section A7.2.6 that determines the effect of an increase in the moderator temperature from 558 K (representative of a core average moderator temperature) to 590 K (representative of an average hot leg moderator temperature) and the moderator density is correspondingly reduced from 0.733 g/cm^3 to 0.690 g/cm^3. The results of these evaluations show that this increase in moderator temperature results in an increase in the dose rate by approximately 4% and an increase in the decay heat by approximately 2%.

The fuel qualification, however, ensures that the design basis fuel assembly utilized in the shielding calculations results in bounding dose rates (see Response to A7.1).

Note that the source terms calculations are performed using a moderator temperature of 558 K while the corresponding moderator density employed (0.733 g/cm^3) is representative of a moderator temperature of 570 K (566 ° F). The use of a density that corresponds to a moderator temperature of 570 K is justified because it is representative of a core average moderator temperature.

Section A7.2.1 of the SAR (Page A7.2.3) that discusses the "Reactor Coolant System Temperature" will be modified to include the results of the sensitivity evaluation discussed above. The following text will be added after the first paragraph.

> *The results of the fuel qualification sensitivity calculations with moderator temperature are shown in Table A7.2-11. Cases D1 through D8 are sensitivity calculations where the moderator temperature is increased from 558 K (representative of a core average moderator temperature) to 590 K (representative of an average hot leg moderator temperature) and the moderator density is correspondingly reduced from 0.733 g/cm^3 to 0.690 g/cm^3. The results of these evaluations show that this increase in moderator temperature results in an increase in the dose rate by approximately 4% and an increase in the decay heat by approximately 2%. The fuel qualification, however, ensures that the design basis fuel assembly utilized in the shielding calculations results in bounding dose rates. In addition, the use of a moderator density of 0.733 g/cm^3 (which corresponds to a moderator temperature of 566ºF)* for the design basis *is justified because 566ºF is representative of a core average moderator temperature.*

.

DOSE ANALYSIS

RAIs
A7A.1 THROUGH A7A.5, & ED-2

DOSE ANALYSIS

RAI: A7A.1

Identify the dimensions, conservatisms, and assumptions used in the TN-40HT cask model and the justification for all assumptions used in the shielding evaluation. Include all relevant dimensions, conservatisms, and assumptions used to generate the SAS2H and MCNP models, along with the justification for any differences between the TN-40HT cask design and the models used in the shielding evaluation.

Section A7A.4.1 states, in part, that the MCNP model used for normal and off-normal conditions is essentially based on the design details from the TN-40HT cask drawings, shown in Section A1.5, except for some conservative representations. The SAR must describe the computational models, data, and assumptions used in evaluating shielding effectiveness. More detail needs to be included (i.e., the distinct dimensions used in the models) in order for staff to confirm the adequacy of the shielding evaluation.

This information is needed to determine compliance with 10 CFR 72.24.

Response: A7A.1

A description of the MCNP model for shielding calculations is provided in Section A7A.4.1 of the SAR. The MCNP model of the TN-40HT cask is based on a “same or similar” representation of the cask from the drawings and is described in the SAR as being “essentially” the same. This implies that the MCNP model is an exact representation of the TN-40HT cask as designed within the limitations of the code geometry modeling options.

The details of the MCNP models as described on Page A7A.4-2 also include the conservative simplifications (differences from the actual design) in the model. Table A7A.1-1 provides the cask material densities and thicknesses as designed and employed in the MCNP models. Figure A7A.1-1 is a sketch of the TN-40HT cask containing the modeled dimensions in the shielding evaluation models.

The MCNP models plots are shown in Figure A7A.4-1 and Figure A7A.4-2 also contain important details that are consistent with the physical design of the TN-40HT cask. The MCNP input file listing is also included in Section A7B and provides further information.

SAR Section A7A.4 will be modified to include the following key assumptions:

- The condition of the cask during and after an accident assumes the side neutron shield and steel shell, the protective cover and the top neutron shield (polypropylene) are lost.

- The borated neutron absorber sheets in the TN-40HT basket are modeled as aluminum.
- Fuel is homogenized into 4 zones within the fuel assembly perimeter, although the TN-40HT basket is modeled explicitly.
- The basket is modeled as discrete stainless steel boxes surrounded by aluminum plates. The stainless steel support bars are conservatively neglected.
- The spatial distribution of the source is assumed to be uniform within each non-fuel hardware zone and within each axial burnup segment in the active fuel. Isotropic angular distribution is assumed for all sources.

The second paragraph of Section A7A.4.1 will be modified as shown below (the additions are in bold).

> *The MCNP model for these shielding configurations is based on a discrete basket with the homogenized fuel assemblies (with an active height of 144 inches) positioned within fuel compartments. The MCNP model developed in this calculation is ~~essentially~~ based on the design details from the TN-40HT cask* drawings **(within the limitations of the code geometry modeling options)**, shown in Section A1.5, except for some conservative representations. **Table A7A.1-1 provides the cask material densities and thicknesses as designed and employed in the MCNP models. Figure A7A.1-1 is a sketch of the TN-40HT cask containing the modeled dimensions in the shielding evaluation models.** The cells 2051 through *2133 represent the discrete basket and fuel assembly zones.*

RAI: A7A.2

Identify the differences between the gamma and neutron models used for the normal and off-normal conditions of the shielding evaluations and the justification for any assumptions and conservatisms used in the models.

Section A7A.4.1 states that two models were developed for determining the normal and off-normal dose rates. The gamma model containing a detailed segmentation of the thicker cask steel body is utilized to calculate the primary gamma dose rates. The neutron model is utilized to calculate the neutron and secondary gamma dose rates.

Staff finding is that although two different models were developed, no justification was included for the differences in the two models (e.g., why a thicker cask steel was used). In addition, the SAR states that the thickness of the gamma shield was reduced but the neutron shield thickness was increased as a consideration of the overall weight. More detail regarding the differences in dimensions need to be provided as part of this analysis.

This information is needed to determine compliance with 10 CFR 72.24.

Response: A7A.2

There are no differences in the modeled material thicknesses between the gamma and neutron shielding evaluation models. The description of "detailed segmentation" of the cask body for the gamma dose rate calculations pertains to the geometry splitting variance reduction technique implemented in the MCNP model. Because the steel body has a larger impact on the primary gamma attenuation compared to the neutron attenuation, geometry splitting was only utilized in the gamma MCNP models. Therefore, the cask body steel was modeled with 10 layers for the gamma model while it was modeled with an equivalent single layer for the neutron model. Utilizing this modeling improves the MCNP computational performance.

This segmentation modeling is also described in the first paragraph of Page A7A.4-2 where it states "A simple analog model is used for calculating the neutron dose. For the primary gamma dose rates, a multiple cell sub-layer model is used."

The statement "the thickness of the gamma shield was reduced but the neutron shield thickness was increased as a consideration of the overall weight" could not be found in the SAR. However, this change in thickness would be relative to the TN-40 cask design and not in the MCNP models used to analyze the TN-40HT cask.

The SAR text in the fifth paragraph on page A7A.4-2 will be modified as shown below (note the changes are in bold):

"Two MCNP models are developed for determining the normal and off-normal dose rates. The gamma model containing a detailed segmentation of the thicker cask steel body **(for variance reduction purposes - implemented employing multiple cell sub-layers)** is utilized to calculate the primary gamma dose rates. The neutron model is utilized to calculate the neutron and secondary gamma dose rates."

RAI: A7A.3

Provide relevant calculations and assumptions regarding the exponential function and decay constant used in specifying the total neutron and gamma source term strengths as described in Section A7A.7.1.

The discussion in Section A7A.7.1 addresses how the source term strengths can be approximated with an exponential function as a function of decay time. However, it is not clearly defined how the exponential function is used to approximate the source strength. No information was identified supporting the relationship between the source term strength and decay time. In addition, it is not clearly defined whether or not the relationship assumes that all nuclides decay at the same rate.

This is required for staff to determine whether appropriately detailed SAR calculations show that the radiation shielding features are sufficient to meet the requirements of 10 CFR 72.24, 10 CFR 72.104, and 10 CFR 72.106.

Response: A7A.3

The discussion of the calculation of the source terms for the ISFSI site dose calculations is provided in SAR Section A7A.7.1, Page A7A.7-2. As described in the second paragraph, the fuel assembly gamma and neutron source terms are calculated for cooling times ranging from 18 years to 40 years with 2-year increments.

The spectral distribution of the neutron source terms is due to Cm-244 and remains unchanged as a function of decay time. The spectral distribution of the fuel assembly hardware (including BPRAs and TPDs) is due to Co-60 and remains unchanged as a function of decay time. The spectral distribution of the fuel assembly in-core gamma source is the same as that of the 18 year cooled fuel (design basis fuel) and remains unchanged as a function of decay time. Since the source spectrum remains unchanged as a function of time and the total source strength (one value each for neutron and gamma (in-core and hardware) at decay time is known (calculated from SAS2H), a mathematical function can be utilized to fit the total source term as a function of time.

An exponential function was employed for this purpose. The source strength at any cooling time is expressed as:

$$A_t = A_0 * e^{(-\lambda(t-18))}$$

where

A_t is the Source Strength at time t ($18 \le t \le 40$)
A_0 is the source strength at 18 years
λ is a decay constant

The decay constants are calculated based on the above equation using the source strengths obtained from SAS2H calculations. The decay constants are shown in Page A7A.7-2 for the fuel gamma, hardware (fittings) gamma and neutron sources.

No attempt has been made to model the nuclide specific decay behavior of the sources. The purpose of this evaluation is to determine simple fitting functions for the total source strengths and to enable simplified input to the MCNP models.

DOSE ANALYSIS

As discussed in the SAR, the differences between the source strengths calculated by SAS2H and that calculated with the exponential function are within 1%.

A reference to the following table will be added to the second paragraph on SAR page A7A.7-2:

TABLE A7A.7-4
SAS2H SOURCE TERMS AS A FUNCTION OF COOLING TIME

Decay Time	Source Strength (particles/sec)				
(years)	Bottom Nozzle	In-Core	Plenum	Top Nozzle	Neutron
18	2.235E+12	3.303E+15	2.870E+12	1.314E+12	7.59E+08
20	1.718E+12	3.142E+15	2.206E+12	1.010E+12	7.05E+08
22	1.321E+12	2.989E+15	1.696E+12	7.763E+11	6.54E+08
24	1.015E+12	2.843E+15	1.304E+12	5.967E+11	6.08E+08
26	7.805E+11	2.704E+15	1.002E+12	4.587E+11	5.65E+08
28	6.000E+11	2.573E+15	7.705E+11	3.527E+11	5.25E+08
30	4.613E+11	2.447E+15	5.923E+11	2.711E+11	4.88E+08
32	3.546E+11	2.328E+15	4.553E+11	2.084E+11	4.54E+08
34	2.726E+11	2.214E+15	3.500E+11	1.602E+11	4.22E+08
36	2.095E+11	2.106E+15	2.691E+11	1.232E+11	3.93E+08
38	1.611E+11	2.004E+15	2.068E+11	9.468E+10	3.65E+08
40	1.238E+11	1.906E+15	1.590E+11	7.278E+10	3.40E+08

In addition the following will be added to the third paragraph on page A7A.7-2:

The source strength at any cooling time is expressed as:

$A_t = A_0 * e^{(-\lambda(t-18))}$
where
A_t is the Source Strength at time t ($18 \le t \le 40$)
A_0 is the source strength at 18 years
λ is a decay constant

The decay constants are calculated based on the above equation using the source strengths obtained from SAS2H calculations.

RAI: A7A.4

Table A7.2-6

Clarify if the light elements Co-60 and Ni-63 should be included in the radioactive inventory for the 14x14 design basis fuel assembly.

Section A7A.8.5.1 of the SAR states that Table A7.2-6 lists the activity representing the fission gases, volatiles, and fines contributing more than 0.1% of the activity contained in the design basis fuel, plus Iodine-129. It appears that the light elements Co-60 and Ni-63 contribute more than 0.1% of the activity contained in a design basis fuel (based on SAS2H results, 0.39% and 0.21% respectively), but they were not included in Table A7.2-6.

This information is required to determine compliance with 10 CFR 72.24(l)(1).

Response: A7A.4

The light elements Co-60 and Ni-63 are not included in the radioactive inventory for the confinement evaluation because they do not fall under the category of "fission products" (0.1% of activity) or "actinides" (0.01% of activity) as described in Section V.3 of the Attachment to ISG-5, Revision 1.

The light elements are not included in the radioactive inventory because they are not part of the fuel pellet matrix and as such are not classified as fission products. Thus they do not contribute to the confinement source term as gases, volatiles or fines. The light element activities are as a result of the irradiation of the fuel assembly hardware / cladding materials (other than the fuel pellet) and are thus are not available for release.

There are no SAR or TS changes proposed as part of the response to this RAI.

RAI: A7A.5

Justify the use of a 45 day exposure period for off-normal conditions in Section A7A.8.5.2.

NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities," states that for off-normal conditions, the bounding exposure duration should be the same as those for normal conditions which assumes that an individual is present at the controlled area boundary for one full year (8760 hours). Alternative exposure duration may be considered by the staff if the applicant provides justification.

This information is required to determine compliance with 10 CFR 72.104(a).

Response: A7A.5

Note that there were two subsections in A7A.8 numbered A7A.8.1. This has been corrected. Subsection A7A.8.5 has become A7A.8.6 and will be referred to as such in the discussion below.

DOSE ANALYSIS

As stated in proposed SAR Section A7A.8.6.1, “Under off-normal conditions, it is assumed that the OP system is not functioning properly”. This means that the inter-seal pressure can not be maintained and there is the potential for leakage out of the cask cavity. Under these assumed conditions, proposed Technical Specification (TS) 3.1.5 Condition A would be entered within in 1 day of the inter-seal pressure falling below the 30 psig setpoint. TS 3.1.5, Action A.1 allows 7 days to increase the inter-seal pressure above the 30 psig setpoint, i.e. return to normal conditions. If the OP system cannot be returned to normal condition within the 7 days, TS 3.1.5, Condition B would be entered and Required Action B.1 would allow 30 days to return the cask to the spent fuel pool and reflood. This action prevents any further off-normal release. Based on these TS allowed times, the maximum duration the cask would be in the off-normal condition would be 1+7+30=38 days. Therefore assuming a 45 day exposure period for off-normal conditions in Section A7A.8.6.2 is conservative.

The following information will be added to the description of the off-normal condition in the proposed SAR Section A7A.8.6.2.

> “The 45 day exposure duration will serve as the bases for the allowed completion times for the Cask Interseal Pressure Technical Specification.”

The following information will be added to TS Bases for TS 3.1.5 Action B.1.

> “The allowed completion times are bounded by the 45 day exposure duration for off-normal conditions in Reference 3”
>
> Reference 3. SAR Section A7A.8

RAI: ED-2

Clarify if in Table A7A.8-3 through A7A.8-6 of the SAR, Np237 should be Np239. Also clarify if in Table A7A.8-6 of the SAR, Cm243 should be Cm244.

Response: ED-2

SAR Tables A7A.8-3 through A7A.8-6 will be changed to show the correct isotopes, i.e. Np239 and Cm244.
