Enclosures 4, 6, 9, 11, and 12 Contain Proprietary Information

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JUN 2 6 2009

L-PI-09-071
10 CFR 72.56

U S Nuclear Regulatory Commission
ATTN: Document Control Desk,
Director, Spent Fuel Project Office,
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

Prairie Island Independent Spent Fuel Storage Installation
Docket No. 72-10
Materials License No. SNM-2506

Supplement to License Amendment Request (LAR) to Modify TN-40 Cask Design (Designated as TN-40HT) (TAC No. L24203)

References:
1. Nuclear Management Company, LLC (NMC) letter to US Nuclear Regulatory Commission (NRC), "License Amendment Request (LAR) to Modify TN-40 Cask Design (Designated as TN-40HT)", dated March 28, 2008, Accession Number ML081190039.
2. NMC letter to NRC, "License Amendment Request (LAR) to Modify TN-40 Cask Design (Designated as TN-40HT) (TAC No. L24203)", dated June 26, 2008.
3. NMC letter to NRC, "Supplement to License Amendment Request (LAR) to Modify TN-40 Cask Design (Designated as TN-40HT) (TAC No. L24203)", dated August 29, 2008.
4. NRC letter to NMC, "Request For Additional Information for Review of Nuclear Management Company, LLC Amendment Request To Modify the TN-40 Cask Design at the Prairie Island Independent Spent Fuel Storage Installation (ISFSI)", dated January 24, 2009.

In Reference 1, NMC* submitted an LAR to revise the Special Nuclear Materials (SNM) license and Technical Specifications (TS) for the Prairie Island Independent Spent Fuel Storage Installation (PIISFSI), to modify the TN-40 cask for storage of higher

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

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NMSS01

enrichment and burnup fuel. References 2 and 3 provided supplemental information for the LAR. In Reference 4, the NRC Staff requested additional information to support their review of Reference 1. As discussed below, this letter and its enclosures provide the responses to the NRC Staff requests for additional information. NSPM submits this supplement in accordance with the provisions of 10 CFR 72.56.

Enclosure 1 to this letter contains the oath or affirmation statement for this supplement required pursuant to 10 CFR 72.16(b).

Enclosure 2 to this letter contains the affidavit and withholding request, pursuant to the requirements in 10 CFR 2.390(b)(1)(iii), of trade secret information contained in Enclosures 4, 6, 9, 11, and 12.

Enclosure 3 to this letter contains the NSPM responses to the non-proprietary additional information requested by NRC Staff.

Enclosure 4 to this letter contains the NSPM responses to the proprietary additional information requested by NRC Staff. This enclosure contains trade secret information that is proprietary to Transnuclear, Inc.

Enclosure 5 to this letter contains the instructions for the page updates to the Technical Specifications (TS), Technical Specification Bases, and Safety Analysis Report (SAR).

Enclosure 6 to this letter contains the updates to the SAR sections that are required in support of the NSPM response to the request for additional information by NRC Staff. This enclosure contains trade secret information that is proprietary to Transnuclear, Inc.

Enclosure 7 to this letter contains the updates to the TS that are required in support of the NSPM response to the request for additional information by NRC Staff.

Enclosure 8 to this letter contains the updates to the TS Bases that are required in support of the NSPM response to the request for additional information by NRC Staff.

Enclosure 9 to this letter contains changes to the TS, TS Bases and SAR sections page mark-ups from the Reference 1 submittal. Please note that it does not include pages where information on a previous (or subsequent) page has rolled over resulting in a page number change. This enclosure contains trade secret information that is proprietary to Transnuclear, Inc.

Enclosure 10 to this letter contains the non-proprietary updates to the SAR sections that are required in support of the NSPM response to the request for additional information by NRC Staff.

Enclosure 11 to this letter contains computer input / output files that were used in support of the NSPM response to the request for additional information by NRC Staff. These files are provided to aid NRC review. This enclosure contains trade secret information that is proprietary to Transnuclear, Inc.

Enclosure 12 to this letter contains a copy of the references for the guide tube and instrument tube dimensions for the Exxon fuel assemblies. These references are proprietary and contain trade secret information.

The supplemental information provided in this letter does not impact the conclusions presented in the June 26, 2008 submittal as supplemented on June 26, 2008 and August 29, 2008.

If there are any questions or if additional information is needed, please contact Mr. Dale Vincent, P.E., at 651-388-1121.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

Michael D. Wadley

Michael D. Wadley
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Northern States Power Company - Minnesota

Enclosures (12)

cc: Administrator, Region III, USNRC (letter only)
NMSS Project Manager, TN-40HT LAR, USNRC (8 copies Enclosure 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, and 12; 4 copies Enclosure 11)
NRR Project Manager, Prairie Island Nuclear Generating Plant, USNRC (letter only)
Resident Inspector, Prairie Island Nuclear Generating Plant, USNRC (letter only)
State of Minnesota (letter only)

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY - MINNESOTA

PRAIRIE ISLAND INDEPENDENT SPENT FUEL STORAGE FACILITY
DOCKET NO. 72-10

REQUEST FOR AMENDMENT TO
MATERIALS LICENSE No. SNM-2506

SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR)
TO MODIFY TN-40 CASK DESIGN (DESIGNATED AS TN-40HT)

Northern States Power Company - Minnesota, provides the requested additional information that supports the request for changes to the Prairie Island Independent Spent Fuel Storage Facility Material License.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY - MINNESOTA

By Michael D. Wadley

Michael D. Wadley
Site Vice President,
Prairie Island Nuclear Generating Plant
Northern States Power Company - Minnesota

State of Minnesota

County of Goodhue

On this 26 day of June 2009 before me a notary public acting in said County, personally appeared Michael D. Wadley, Site Vice President, Prairie Island Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company - Minnesota, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true.

Carla A. McEathron

CATHERINE S MCEATHRON
Notary Public
Minnesota
My Comm. Expires Jan 31, 2010

ENCLOSURE 2

Proprietary Affidavit Pursuant to 10 CFR 2.390

2 Pages Follow

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

Transnuclear, Inc.)
State of Maryland) SS.
County of Howard)

I, Jayant Bondre, depose and say that I am a Vice President of Transnuclear, Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosures 4, 6, 9, 11, and 12 and as listed below:

1. Proprietary Request for Additional Information (RAI) responses.
2. Portions of SAR Section A1.5, SAR Drawings TN40HT-72 series, as follows:
   - Transnuclear Drawing TN40HT-72-2, Revision 1
   - Transnuclear Drawing TN40HT-72-21, Revision 2
3. Portions of updates to SAR Chapters A3 and A4.
4. Portions of SAR section page mark-ups.
5. Computer analysis input and output files associated with Transnuclear's analysis of the TN-40HT cask.
6. XN-NF-83-87, "Mechanical Design Report Supplement for Margin Upgrade of Prairie Island Units 1 and 2 TOPROD Fuel" October 1983.
7. XN-NF-78-34 (P), "Generic Mechanical and Thermal Hydraulic Design for Exxon Nuclear 14x14 Reload Fuel Assemblies with Zircaloy Guide Tubes for Westinghouse 2-Loop Pressurized Water Reactors," November 1978.

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Transnuclear, Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure are portions of certain TN-40HT spent fuel storage cask design drawings and analyses, plus certain computer files associated with those analyses, all of which are owned and have been held in confidence by Transnuclear, Inc., plus certain reports which were obtained under a proprietary agreement with others, and have been held in confidence by Transnuclear, Inc.

2) The information is of a type customarily held in confidence by Transnuclear, Inc. and not customarily disclosed to the public. Transnuclear, Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.390 with the understanding that it is to be received in confidence by the Commission.

4) The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

5) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. because:

   a) A similar product is manufactured and sold by competitors of Transnuclear, Inc.

   b) Development of this information by Transnuclear, Inc. required expenditure of considerable resources. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.

   c) In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of a design and analysis of a dry spent fuel storage system.

   d) The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

   e) The information consists of descriptions of the design and analysis of dry spent fuel storage systems, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

   f) In pricing Transnuclear, Inc.'s products and services, significant research, development, engineering, analytical, licensing, quality assurance and other costs and expenses must be included. The ability of Transnuclear, Inc.'s competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

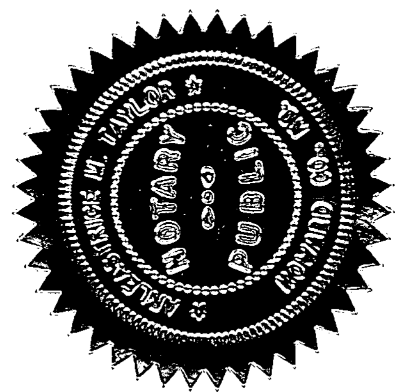
Further the deponent sayeth not.

Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this23rd day of June, 2009.

Notary Public

My Commission Expires 10 / 14 / 2012



ENCLOSURE 3

NON-PROPRIETARY RAI RESPONSES

97 pages follow

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 safety analysis report but also essentially all other Commission requirements governing the operation of nuclear power plants. ... It has diverted both 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 Site Specific 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).

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, NSPM proposes to use the default definition of Damaged Fuel. This will be accomplished by revising SAR Section A3.3.7.1 and the proposed 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 SAR and proposed 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 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: 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 has 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.

The first paragraph in SAR Section A4.2.3.6.3 will be revised as follows (the additions are in Bold):

> Neolube, Loctite N-5000, or equivalent may be used to coat the threads and bolt shoulders of the TN-40HT cask closure bolts. Loctite N-5000 or equivalent may be used to coat the contact areas of the top and bottom trunnions prior to lifting operations. **The lubricant shall be removed prior to immersing the cask into the spent fuel pool unless the lubricant has been approved for compatibility with the spent fuel pool water.**

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).

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 dioxide rather than with enriched uranium dioxide. Since the uranium dioxide 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 what is meant by uranium rods, the sentence in Section A3.1.1 will be changed to read as follows.

> "Reconstituted assemblies, (natural uranium dioxide 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

	Exxon STD	Exxon High Burnup	Exxon TOPROD	West STD	WEST OFA
Instrument Tube ID	0.374 in. See Note 3	0.374 in. See Note 3	0.374 in. Table 4.1 of XN-NF-83-87	0.374 in. Table 3-1 of WCAP 16517-NP	0.352 in. Table 3-1 of WCAP 16517-NP
Instrument Tube OD	0.424 in. See Note 3	0.424 in. See Note 3	0.424 in. Table 4.1 of XN-NF-83-87	0.422 in. Table 3-1 of WCAP 16517-NP	0.399 in. Table 3-1 of WCAP 16517-NP
Guide Tube ID	0.507 in. Table 2.1 of XN-NF-78-34	0.507 in. Table 2.1 of XN-NF-78-34	0.507 in. Table 4.1 of XN-NF-83-87	0.505 in. Table 3-1 of WCAP 16517-NP	0.492 in. Table 3-1 of WCAP 16517-NP
Guide Tube OD	0.541 in. Table 2.1 of XN-NF-78-34	0.541 in. Table 2.1 of XN-NF-78-34	0.541 in. Table 4.1 of XN-NF-83-87	0.539 in. Table 3-1 of WCAP 16517-NP	0.526 in. Table 3-1 of WCAP 16517-NP

XN-NF-78-34 (P) "Generic Mechanical and Thermal Hydraulic Design For Exxon Nuclear 14x14 Reload Fuel Assemblies With Zircaloy Guide Tubes for Westinghouse 2-Loop Pressurized Water Reactors", November 1978. See Enclosure 12

XN-NF-83-87 "Mechanical Design Report Supplement for Margin Upgrade of Prairie Island Units 1 and 2 TOPROD Fuel" October 1983. See Enclosure 12.

WCAP-16517-NP "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis November 2005". See Accession Number ML053390121.

Notes:

1. WCAP-16517-NP was sent to the NRC as Enclosure 2 of Nuclear Management Letter L-PI-05-110 dated 12/02/2005, Subject: "Supplement to License Amendment Request (LAR) to Revise Spent Fuel Pool Criticality Analyses and Technical Specifications (TS) 3.7.17, 'Spent Fuel Pool Storage' and 4.3 'Fuel Storage' (TAC Nos MC5811 and MC5812)", see Accession Number ML053390121.
2. Table 3-1 of WCAP-16517-NP lists the Guide Tube ID and OD for OFA fuel as 0.492 and 0.526 inches respectively. These values are slightly different than those listed in Table 6-2 of the TN-40 Transport SAR (i.e. 0.490 and 0.528 inches). NSPM has confirmed that the 0.490 and 0.528 values are consistent with the fuel drawings for OFA fuel using Zircaloy-4 material. The values listed in the WCAP correspond to guide tubes made with the ZIRLO material. In any case this small difference in diameter of the guide tubes has an insignificant affect on the criticality analysis.
3. Exxon Report XN-NF-78-34 does not list the dimensions of the instrument tube for the Exxon standard and High Burnup fuel design. However, NSPM has confirmed that the instrument tube dimensions for the Exxon standard and High Burnup fuel assembly design are the same as those for the Exxon TOPROD fuel assembly design.

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.

Response: M4

The correct value for the loading of a Westinghouse Standard Fuel type should be 0.410 MTU.

SAR 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.

The second sentence of Note 3 on SAR Table A7A.8-1 will be modified to say that the 140 $\mu Ci/cm^2$ value is "per Table 7.1 of Reference 8". Reference 8 is ISG-5 Revision 1.

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 1.1 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 $2.2x10^6$ 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 $2x10^7$ rads (see first 2 paragraphs on page 2 of 4, of Attachment 1 to NRC Information Notice No. 86-57, Accession # ML031220718). 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 CFR 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 CFR 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 Site Specific 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 CFR 72.124(a) is already in 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.

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

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 provided in new SAR Section A9.7.3 with acceptance testing located in new SAR Section A9.7.4. Qualification testing of new Metal Matrix Composites will be located in new SAR Section A9.7.5 and process controls for Metal Matrix Composites will be located in new SAR Section A9.7.6.

The information in these new sections may be used to determine compliance with 10 CFR 72.124(a).

These new sections are based on Transnuclear's response to NRC RAI questions 9.1 through 9.11 for the NUHOMS HD CoC 1030 Amendment 1 application (TN Letter E-27377, Dated December 15, 2008, TAC NO. L24153).

Note there is no hydrogen in the metallic neutron absorbers. Therefore the acceptance testing does not determine the percentage or density of H in the absorber material.

Regulation 10 CFR 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 CFR 72.44(c), concluded that the regulation does not require fabrication acceptance testing of the neutron absorber plates to 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 fabrication acceptance testing of the neutron absorber plates in the Technical Specifications. The review concluded that NUREG-1745 did not call for fabrication acceptance testing of the neutron absorber plates 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 Site Specific 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 fabrication acceptance testing of the neutron absorber plates.

Since the information needed to demonstrate compliance with 10 CFR 72.124(a) will be added in SAR Sections A9.7.3 through A9.7.6, and for the reasons above, NSPM does not propose to include fabrication acceptance testing of the neutron absorber plates into the proposed Technical Specifications.

The following will be added to the SAR:

A9.7.3 NEUTRON ABSORBER REQUIREMENTS

The neutron absorber used for criticality control in the TN-40HT basket may consist any of the following types of material:

(a) Boron-aluminum alloy (borated aluminum)

(b) Boron carbide / aluminum metal matrix composite (MMC)

(c) Boral®

The TN-40HT safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials.

To assure performance of the neutron absorber's design function only visual inspections, thermal conductivity testing, and the presence / uniformity of B10 need to be verified with testing requirements specific to each material.

References to metal matrix composites throughout this chapter are not intended to refer to borated aluminum or Boral®.

A9.7.3.1 Boron Aluminum Alloy (Borated Aluminum)

Description

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating as a uniform fine dispersion of discrete aluminum diboride (AlB_2) or Titanium diboride (TiB_2) particles in the matrix of aluminum or aluminum alloy. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The boron may have the natural isotopic distribution or may be enriched in B10.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section A9.7.4.3.

Requirements

The boron content in the aluminum or aluminum alloy shall not exceed 5% by weight.

The neutron absorbers shall be 100% visually inspected in accordance with the inspection requirements described in Section A9.7.4.1.

The thermal conductivity of the material shall be tested in accordance with the testing requirements in Section A9.7.4.2.

The minimum B10 areal density specified in Table A3.3-17 shall be confirmed via neutron transmission testing as described in Section A9.7.4.3.

A9.7.3.2 BORON CARBIDE / ALUMINUM METAL MATRIX COMPOSITES (MMC)

<u>Description</u>

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. It is a low-porosity product, with a metallurgically bonded matrix.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section A9.7.4.3.

<u>Requirements</u>

For non-clad MMC products, the boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

Non-clad MMC products shall have a density greater than 98% of theoretical density , with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

Boron carbide particles for the products considered here typically have an average size in the range 10-40 microns, although the actual specification may be by mesh size, rather than by average particle size. No more than 10% of the particles shall be over 60 microns.

The neutron absorbers shall be 100% visually inspected in accordance with the inspection requirements described in Section A9.7.4.1.

The thermal conductivity of the material shall be tested in accordance with the testing requirements in Section A9.7.4.2.

The minimum B10 areal density specified in Table A3.3-17 shall be confirmed via neutron transmission testing as described in Section

A9.7.4.3.

The MMCs material shall be qualified in accordance with the requirements specified in Section A9.7.5, and shall subsequently be subject to the process controls specified in Section A9.7.6.

A9.7.3.3 BORAL®

Description

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. The average size of the boron carbide particles in the finished product is approximately 50 microns after rolling.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®.

Requirements

The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The neutron absorbers shall be 100% visually inspected in accordance with the inspection requirements described in Section A9.7.4.1.

The thermal conductivity of the material shall be tested in accordance with the testing requirements in Section A9.7.4.2.

The minimum B10 areal density specified in Table A3.3-17 shall be confirmed via chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing described in Section A9.7.4.3. Areal density testing shall be performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

A9.7.4 NEUTRON ABSORBERS ACCEPTANCE TESTING

A9.7.4.1 VISUAL INSPECTIONS OF NEUTRON ABSORBERS

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data (Reference 6), Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings". Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be treated as non-conforming. Inspection of MMCs with an integral aluminum cladding shall also include verification that the matrix is not exposed through the faces of the aluminum cladding and that solid aluminum is not present at the edges.

For Boral®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet.

A9.7.4.2 THERMAL CONDUCTIVITY TESTING OF NEUTRON ABSORBERS

Testing shall conform to ASTM E1225 (Reference 7), ASTM E1461 (Reference 8), or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Previous testing of borated aluminum and metal matrix composite, Table A9.7-1, shows that thermal conductivity increases slightly with temperature. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.

If a thermal conductivity test result is below the specified minimum, additional tests may be performed on the material from that lot. If the mean value of those tests falls below the specified minimum the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the boron appearing in the same phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The thermal analysis in Chapter A3.3.2.2 considers a dual plate basket construction base model with 0.125" thick neutron absorber with a 0.312" thick aluminum 1100 plate. This model gives the bounding values for the maximum component temperatures. Either a dual plate basket construction or an alternate single plate (borated aluminum or MMC) construction basket may be utilized. For the dual plate construction, the specified thickness of the neutron absorber may vary, and the thermal conductivity acceptance criterion for the neutron absorber will be based on the nominal thickness specified. In either construction type, to maintain the thermal performance of the basket, the minimum thermal conductivity shall be such that the total thermal conductance (sum of conductivity * thickness) of the neutron absorber and the aluminum 1100 plate shall at least equal the conductance assumed in the analysis for the base model. Samples of the acceptance criteria for various neutron absorber thicknesses are highlighted in Table A9.7-2.

The aluminum 1100 plate does not need to be tested for thermal conductivity; the material may be credited with the values published in the ASME Code Section II part D. The neutron absorber material need not be tested for thermal conductivity if the nominal thickness of the aluminum 1100 plate is 0.359 inch or greater.

A9.7.4.3 Neutron Transmission Testing of Neutron Absorbers

Neutron Transmission acceptance testing procedures shall be subject to approval by Transnuclear . Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in a lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.
The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of boron 10 at that energy.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 0.75 sq. inch.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with Transnuclear 's QA procedures.

The following illustrates one acceptable method and is intended to be utilized as an example. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined as the mean value of B10 volume density for the sample less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence (Reference 9).

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness.

Non-conforming material shall be evaluated for acceptance in accordance with Transnuclear 's QA procedures.

A9.7.5 Qualification Testing of Metal Matrix Composites

A9.7.5.1 APPLICABILITY AND SCOPE

Prior to initial use in a spent fuel dry storage system, new MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section A9.7.6 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage system.

ASTM methods and practices are referenced below for guidance. Alternative methods may be used with the approval of Transnuclear .

A9.7.5.2 DURABILITY

There is no need to include accelerated radiation damage testing in the qualification. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842 °F (Reference 10), well above the basket temperature under normal conditions of storage or transport.

Corrosion testing is not required for MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear (Reference 11).

A9.7.5.3 DELAMINATION TESTING OF CLAD MMC

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage.

A9.7.5.4 REQUIRED TESTS AND EXAMINATIONS TO DEMONSTRATE MECHANICAL INTEGRITY

At least three samples, one each from approximately the two ends and middle of the test material production run shall be subjected to:

a) room temperature tensile testing (ASTM- B557 (Reference 12)) demonstrating that the material:

- has a 0.2% offset yield strength no less than 1.5 ksi;
- has an ultimate strength no less than 5.0 ksi; and
- has minimum elongation in two inches no less than 0.5%.

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290 (Reference 13). The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) testing by ASTM-B311 (Reference 14) to verify more than 98% theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a method to be approved by Transnuclear. The maximum interconnect porosity is 0.5 volume %.

And for at least one sample,

c) for MMCs with an integral aluminum cladding, thermal durability testing demonstrating that after a minimum 24 hour soak in either pure or borated water, then insertion into a preheated oven at approximately 825°F for a minimum of 24 hours, the specimens are free of blisters and delamination and pass the mechanical testing requirements described in test 'a' of this section.

A9.7.5.5 REQUIRED TESTS AND EXAMINATIONS TO DEMONSTRATE B10 UNIFORMITY

Uniformity of the boron distribution shall be verified either by:

(a) Neutron radioscopy or radiography (ASTM E94 (Reference 15), E142 (Reference 16), and E545 (Reference 17)) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum

B10 areal density, or

(b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section A9.7.4.3, or by chemical analysis for boron carbide content in the composite.

A9.7.5.6 APPROVAL OF PROCEDURES

Qualification procedures shall be subject to approval by Transnuclear.

A9.7.6 PROCESS CONTROLS FOR METAL MATRIX COMPOSITES

This section provides process controls to ensure that the material delivered for use is equivalent to the qualification test material.

A9.7.6.1 APPLICABILITY AND SCOPE

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. Transnuclear shall determine whether a complete or partial re-qualification program per Section A9.7.5 is required, depending on the characteristics of the material that could be affected by the process change.

A9.7.6.2 DEFINITION OF KEY PROCESS CHANGES

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduced density, reduced corrosion resistance, or reduce the mechanical strength or ductility of the MMC.

A9.7.6.3 IDENTIFICATION AND CONTROL OF KEY PROCESS CHANGES

The manufacturer shall provide Transnuclear with a description of materials and process controls used in producing the MMC. Transnuclear and the manufacturer shall identify key process changes as defined in Section A9.7.6.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change.

The following are examples of other changes that are established as key process changes, as determined by Transnuclear's review of the specific applications and production processes:

(a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns, or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit;

(b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering;

(c) Change in the nominal matrix alloy;

(d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for powder metallurgy or thermal spray MMCs that were qualified with extruded material, or a change to direct rolling from the billet;

(e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature;

(f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending; and

(g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

References Added to SAR Section A9.8

6. "Aluminum Standards and Data, 2003" The Aluminum Association.

7 ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"

8. ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

9. Natrella, "Experimental Statistics," Dover, 2005.

10. Pyzak and Beaman, "Al-B-C Phase Development and Effects on Mechanical Properties of B_4C/Al Derived Composites," J. Am. Ceramic Soc., 78[2], 302-312 (1995)

11. "Hydrogen Generation Analysis Report for TN-68 Cask Materials," Test Report No. 61123-99N, Rev 0, Oct 23, 1998, National Technical Systems.

12. ASTM B557, "Standard Test Methods of Tension Testing Wrought and Cast Aluminum- and Magnesium-Alloy Products"

13. ASTM E290, "Standard Test Methods for Bend Testing of Material for Ductility"

14. ASTM B311, "Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less Than Two Percent Porosity"

15. ASTM E94, "Recommended Practice for Radiographic Testing"

16. ASTM E142, "Controlling Quality of Radiographic Testing"

17. ASTM E545, "Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing"

18. Thermal Conductivity Measurements of Boron Carbide/Aluminum Specimens, Oct 1998, testing by Precision Measurements and Instruments Corp. for Transnuclear, Inc., Purchase Order Number 98037

19. Eagle Picher Report AAQR06, "Qualification of Thermal Conductivity, Borated Aluminum 1100", May 2001

Table A9.7-1
Thermal Conductivity for Sample Neutron Absorbers

Temperature °C	Material			
	1	2	3	4
20	193	170	194	194
100	203	183	207	201
200	208	-	-	
250	-	201	218	206
300	211	204	220	203
314	-	-	-	202
342	-	-	-	202

Units: W/mK

Materials:
1) Boralyn® MMC, aluminum 1100 with 15% B_4C
2) Borated aluminum 1100, 2.5% boron as TiB_2
3) Borated aluminum 1100, 2.0% boron as TiB_2
4) Borated aluminum 1100, 4.3% boron as AlB_2

Sources:

References 18 and 19

TABLE A9.7-2
SAMPLE DETERMINATION OF THERMAL CONDUCTIVITY ACCEPTANCE CRITERION

Single Plate Model	Al 1100	n absorber	total
thickness (inch)	0	0.437	0.437
conductivity at 70°F (Btu/hr-in-°F)	n/a	**9.11**	n/a
conductance (Btu/hr-°F)	0	3.98	3.98*

Dual Plate Construction	Al 1100	n absorber	total	
thickness (inch)	0.312	0.125	0.437	as modeled
conductivity at 70°F (Btu/h-.in-°F)	11.09	**4.17**	n/a	
conductance (Btu/hr-°F)	3.46	0.52	3.98	

thickness (inch)	0.187	0.250	0.437	thicker neutron absorber
conductivity at 70°F (Btu/hr-in-°F)	11.09	**7.62**	n/a	
conductance (Btu/hr-°F)	2.07	1.91	3.98	

thickness (inch)	0.359	0.078	0.437	thinner neutron absorber
conductivity at 70°F (Btu/hr-in-°F)	11.09	**0**	n/a	
conductance (Btu/hr-°F)	3.98	0	3.98	

The acceptance criterion is identified by boldface type for each thickness.

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.

Analyses of the thermal gradient and resulting stresses during reflooding will be added in new SAR Section A4.2.3.9.

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."

In addition, the following will be added in new SAR Section A4.2.3.9.

A4.2.3.9 THERMAL STRESS OF FUEL CLADDING DUE TO UNLOADING OPERATIONS

To evaluate the effects of the thermal loads on the fuel cladding during unloading operations, the following assumptions are made:

- A conservative high maximum fuel cladding temperature of 700 °F and quench water temperature of 50 °F are used.
- The fuel rod is assumed to be simply supported at both ends.
- The outer surface temperatures of the fuel cladding are conservatively assumed as shown in Figure A4.2-13. 50 °F (water), 212 °F (steam), and 700 °F (cladding) temperature occurs at three equal heights.
- The fuel cladding thickness and cladding outside diameter are reduced by 0.00270 inch to account for oxidation.

A4.2.3.9.1 FINITE ELEMENT MODEL

The finite element model is shown in Figure A4.2-14. ANSYS (Reference

3) finite element Plane 55 and Plane 42 (Axisymmetric) are used for thermal and structural analysis respectively. The fuel rod with the thinnest cladding (WE14 x 14 STD) is modeled, as this will result in the largest temperature gradient across the cladding (temperatures are kept constant at the inner and outer surfaces). The cladding thickness is 0.0216 inches and the rod outer diameter is 0.4166 inches. A tube length of 2 inches is considered for the analysis such that maximum stresses are not affected by the boundary conditions.

A4.2.3.9.2 MATERIAL PROPERTIES

The following material properties are used for the thermal and structural analysis:

Material Properties for Thermal Analysis

Temp °F	Conductivity Btu/hr-in- °F
212	0.655
392	0.689
572	0.732
752	0.790

Material Properties for Structural Analysis

Temp °F	E (psi)	α in/in- °F	γ	S_y (psi) at 750 °F
300	12.2×10^{6}	3.73×10^{-6}	0.404	126,102
400	11.7×10^{6}			116,272
500	11.2×10^{6}			108,921
600	10.7×10^{6}			102,512
700	10.2×10^{6}			95,793
750	9.93×10^{6}			92,000

A4.2.3.9.3 THERMAL ANALYSIS

Steady state thermal analysis was conducted using the surface nodal temperatures as shown in Figure A4.2-13. The inside surface nodal temperatures are all assumed to be 700 °F , and the outside surface temperatures to conservatively represent the quench water temperature. The temperature distribution resulting from this analysis is shown in Figure A4.2-15.

A4.2.3.9.4 THERMAL STRESS ANALYSIS AND RESULTS

A thermal stress analysis using the same model was conducted using the nodal temperatures obtained from the thermal analysis. The resulting nodal stress intensity distribution is shown in Figure A4.2-16. The maximum nodal stress intensity in the fuel cladding is 24.0 ksi. This stress is less than the yield strength of Zircaloy, which is 92 ksi at 750 °F.

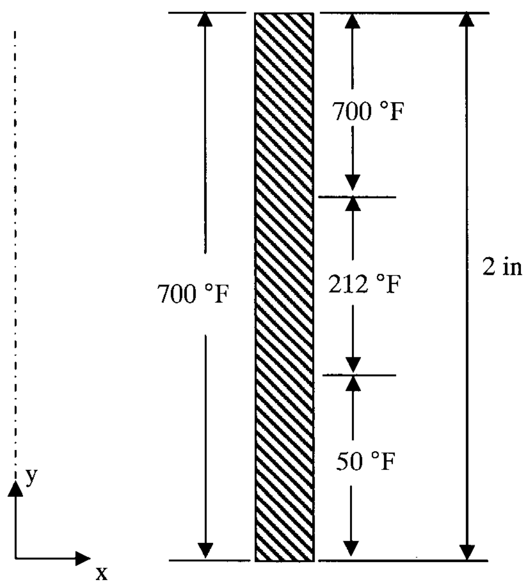


FIGURE A4.2-13
TN40HT FUEL CLADDING OUTER SURFACE TEMPERATURES

FIGURE A4.2-14
TN40HT FUEL CLADDING FINITE ELEMENT MODEL

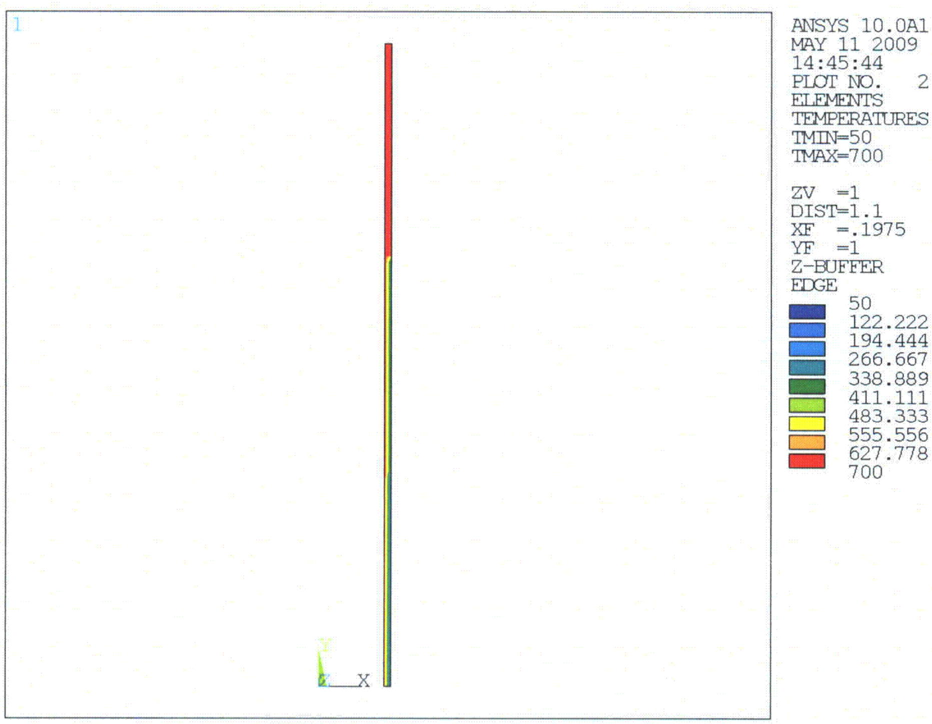


FIGURE A4.2-15
TN40HT FUEL CLADDING TEMPERATURE DISTRIBUTION

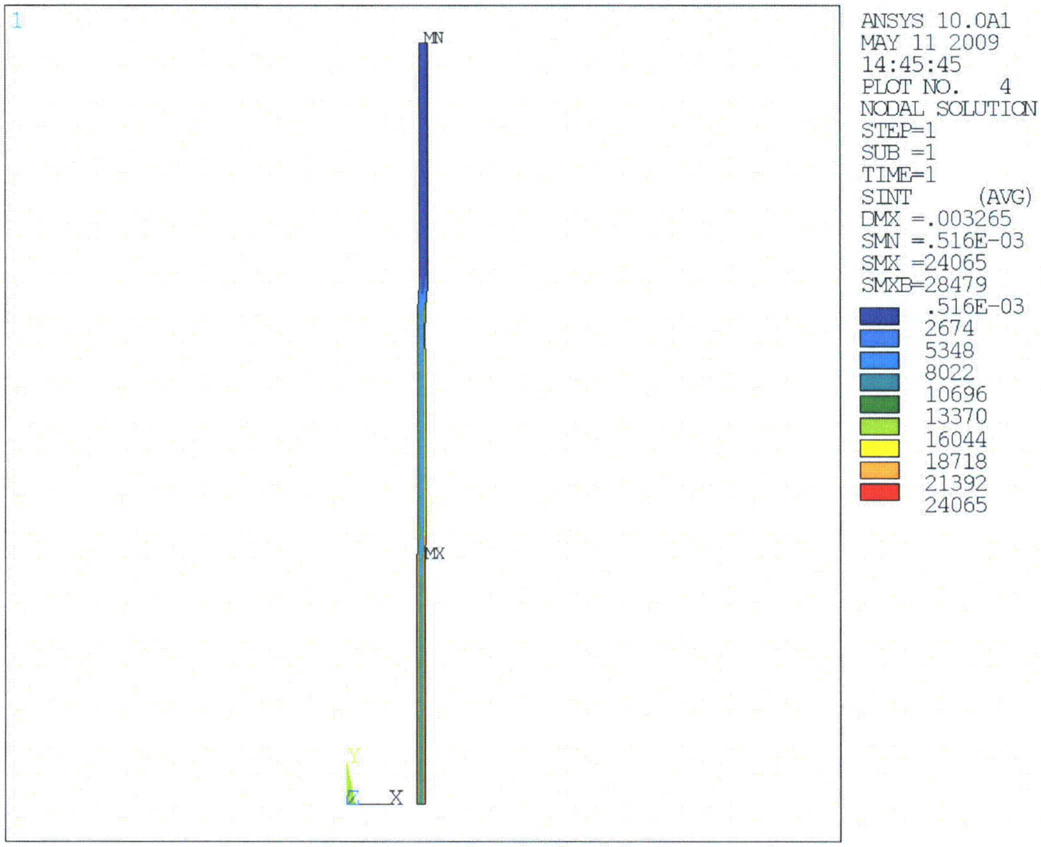


FIGURE A4.2-16
TN40HT FUEL CLADDING STRESS INTENSITY

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 provided in new SAR Section A9.7.7.

The information in these new sections may be used to determine compliance with 10 CFR 72.126(6).

Acceptance of neutron shielding materials

Because the top polypropylene neutron shield is a standard industrial plastic plate, the only acceptance planned is verification of supplier certification to confirm that the material is polypropylene.

Demonstration of durability for neutron shielding materials

Both the polypropylene and the proprietary polyester resin proposed for use in the TN-40HT have been used since 1995 in the TN-40, TN-32, and TN-68 casks with no evidence of degradation of their shielding functions, i.e., no reported increase in dose rates on the cask exterior.

Radiation:

Radiation can cause degradation of polymers by cross-linking or by chain scission. Radiation can also cause radiation-assisted oxidation, which can facilitate chain scission in the polymer. In oxygen-starved conditions, bond repair or crosslinking can be significant mechanisms that prevent chain scission; for this reason, on thick sections such as the TN-40HT neutron shields, most of the damage is confined to a surface layer. The anti-oxidant additive in the radial neutron shield further limits radiation-assisted oxidation.

The threshold for radiation dose damage for polymers is typically greater than 1×10^6 rad. (See page 11 of NASA SP8053, "Nuclear and Space Radiation Effects on Materials", June 1970). To evaluate the radiation damage to the neutron shield, note that the energy absorption of polymers and tissue is similar. Therefore, the gamma radiation energy absorbed by the polypropylene shield may be approximated as the rad equivalent of the surface dose in rem. The absorbed neutron energy may be estimated as half the neutron dose rate to account for the tissue quality factor. Based on SAR Table A7A.2-1, the accident dose rate at the radial surface of the gamma shield is 116 mrem/hr gamma, and 1980 mrem/hr neutron. This is approximately equivalent to 1.1 rad/hr for the radial neutron shield and less for the less for the top shield. At the end of 40 years, assuming that the radiation field remains constant, this would result in absorbed energy in the radial shield of about 3.9×10^5 rad. This is well below the threshold of 1×10^6 rad.

Thermal durability:

> Public sources of information on polypropylene generally establish the melting point of polypropylene near 327° F (for example, CRC Handbook of tables for Applied Engineering Science, 2nd Edition, Table 1-80) and the long term maximum service temperature at 170° F. Published continuous use temperatures apply to typical industrial applications where the material must retain most of its mechanical properties and dimensional stability. In the case of the top neutron shield, because the material is entirely enclosed in a steel shell, the polypropylene could in fact melt, and still perform its shielding function. On this basis, Table A3.3-3 of the SAR assigns a maximum use temperature of 300° F for the polypropylene, 27° F below its melting point. According to the same table, the normal temperature of the top neutron shield is 191° F, well below the limit.

More information on the durability of the radial shielding resin may be found in Appendix 9A of the TN-68 Storage Safety Analysis Report, Docket 72-1027

The RAI questions says that the "plan should be included by reference to the SAR in the proposed CoC". However, the plan is being submitted as part of a License Amendment Request to the site specific License SNM-2506. Therefore there is no CoC. In the event that the intent of the RAI question was that the plan should be included into the Technical Specification by reference, the following discussion is provided.

Regulation 10 CFR 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 CFR 72.44(c), concluded that the regulation does not require the radial neutron shield acceptance plan be included in the Technical Specifications.

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 radial neutron shield acceptance plan in the Technical Specifications. The review concluded that NUREG-1745 did not call for the radial neutron shield acceptance plan 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 Site Specific 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 the radial neutron shield acceptance plan.

Since the information needed to demonstrate compliance with 10 CFR 72.126(6) will be added in SAR Section A9.7.7, and for the reasons above, NSPM does not propose to include the radial neutron shield acceptance plan into the proposed Technical Specifications.

The following will be added to the SAR:

A9.7.7 Radial Neutron Shielding Tests

The shielding performance of the radial polyester resin can be verified adequately by chemical analysis and verification of density. Uniformity is assured by installation process control.

Testing Requirements

Chemical analysis shall be performed on the first batch mixed with a given set of components, and thereafter whenever a new lot of one of the major components is introduced. The acceptance values for the chemical composition of the polyester resin are listed in the following table. Note that the chemical composition used in the shielding models (i.e. listed in Table A7A.4-3) are included in the following table for comparison.

Table A7A.4-3 values		Acceptance Testing Values		
Element	nominal wt %	Element	wt %	acceptance range (wt %)
H	5.05	H	5.05	-10 / +20
B	1.05	B	1.05	± 20
C	35.13	C	35.13	± 20
Al	14.93	Al	14.93	± 20
O	41.73	O+Zn (balance)	43.84	± 20
Zn	2.11			
Total	100.0%		100%	

A density measurement shall be performed on every mixed batch of the polyester resin The minimum polymer density measured shall be greater than 1.547 g/cm^3.

Process Controls

Qualification tests of the personnel and procedure used for mixing and pouring the polyester resin shall be performed. Qualification testing shall include verification that the chemical composition and density is achieved, and the process is performed in such a manner as to prevent voids.

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 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.

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-72-22 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 CFR 72.122(b)(2).

Proposed Technical Specification 4.4 states the following:

> "The TN-40HT basket is designed, 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 CFR 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 CFR 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 Site Specific 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 CFR 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 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).

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 CFR 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 CFR 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 CFR 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 Site Specific 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 CFR 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 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.
>
> Non-confinement 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 CFR 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 CFR 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 CFR 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 NSPM's understanding that Technical Specifications (both the Prairie Island Nuclear Generating Plant's and the Site Specific 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 CFR 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 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 previously 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

While the hemispherical emissivity of 0.46 is reported in Reference 34 for the temperature range from room to 1100 ^{o}F, 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 as described in the SAR Section A.3.3.2.2.3.6.3.1.

Reference 34 will be removed from the reference list, instead the Baumeister and Marks Handbook (SAR, Reference 6) will be used as the basis for the stainless steel emissivity. The emissivities for stainless steel plates are provided in Table 2, page 4-111 of Baumeister and Marks Handbook.

SAR Section A.3.3.2.2.3.6.2.3 will be modified to read as follows:

> An emissivity of 0.3 for the stainless steel plates is used for the fuel compartments in calculating the transverse effective fuel conductivity. This value is conservative relative to the values provided in Reference 6.

And;

> Reference 34 in Section A3.6 will be deleted.

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. The values are consistent with those used for the TN-40 storage cask, see SAR Table 3.3-2. Note that the unit of the values in the TN-40 SAR is Btu/hr-ft-°F.

Note that while preparing this response, a typographical error was identified in SAR Table A3.3-8 for the thermal conductivity of the Solid Neutron Shield Resin. The value of 0.0833 Btu/hr-in-°F should be 0.0083 Btu/hr-in-°F. 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 were studied by Amaya et al. (Reference M19-1) and Ronchi et al. (Reference M19-2). Based on the study by Ronchi et al., 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.) can drop significantly (more that 50%) compared to un-irradiated UO_2.

The thermal conductivity values of UO_2 used to calculate the effective fuel conductivity are listed in SAR Section A3.3.2.2.3.6.2.1 and are for un-irradiated pellets. Figure M19-1 below compares these values to those obtained from the study by Ronchi et al. The comparison shows that in the temperature range of interest (600°F to 750°F) the SAR conductivity values are higher by approximately a factor of two compared to values obtained from Ronchi et al.

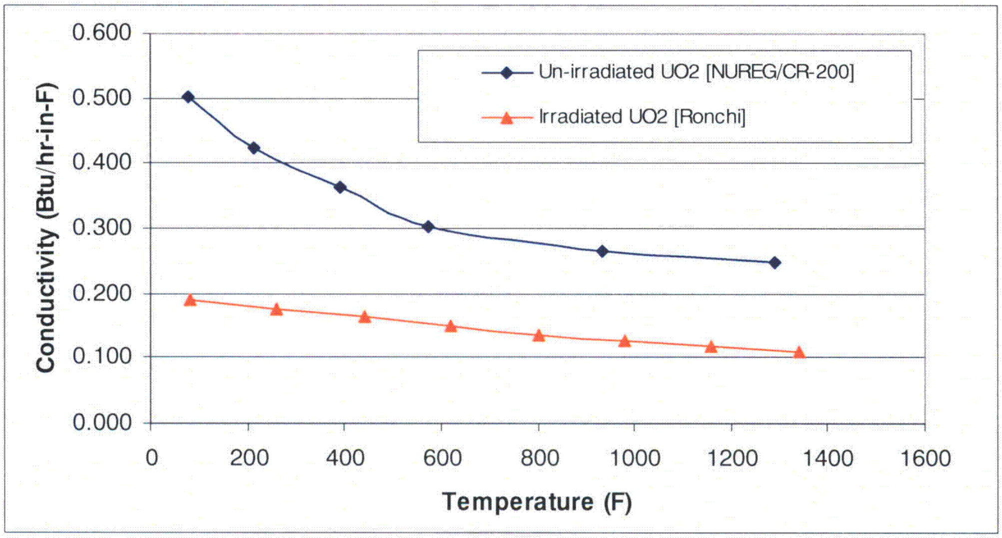


Figure M19–1 UO_2 Thermal Conductivity

The use of irradiated UO_2 conductivity would decrease the calculated effective fuel conductivity in the transverse direction. (Note that as discussed in SAR Section A3.3.2.2.3.6.3.2, 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.) However, the transverse effective conductivities in the SAR were calculated using unirradiated UO_2 and the results are summarized in SAR Table A3.3-9.

A study performed by Transnuclear (TN) and provided to the NRC via a RAI response to NUHOMS® HD System, Amendment 1 (Reference M19-3) 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 showed that the fuel cladding temperature changes by approximately 1°F when using irradiated UO_2 conductivity. Since a cladding temperature change of 1°F is negligible, the results of the study show that the fuel cladding temperature is 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.

In addition, the transverse effective fuel conductivities used in the SAR ANSYS thermal models and presented in Table A3.3-8 are at least 20% lower than the calculated transverse effective conductivities presented in 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 Tables A3.3-8 and A3.3-9 is discussed in

SAR Section A3.3.2.2.3.6.5 and depicted in SAR Figure A3.3-19.

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.

References to RAI-M19:

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.

M19-2) 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.

M19-3) 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.

The following summary of the above discussion will be added to end of SAR Section A3.3.2.2.3.6.2.1.

> The thermal conductivities shown above represent values for un-irradiated UO_2 pellets. A study performed by Transnuclear (TN) and provided to the NRC in Reference 37 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 a temperature of 700°F.
>
> The sensitivity runs in the TN study showed that the fuel cladding temperature changes by approximately 1°F when using irradiated UO_2 conductivity. Since a cladding temperature change of 1°F is negligible, the results of the study show that the fuel cladding temperature is not sensitive to the conductivity of UO_2. Therefore, use of un-irradiated UO_2 fuel pellet conductivity from NUREG/CR-0200 (Reference 14) is reasonable for irradiated UO_2.

And, Reference 37 below will be added to Section A3.6.

37. 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.

RAI: A1.1

PROPRIETARY
See Enclosure #4

Response: A1.1

See Enclosure #4

RAI: A1.2

PROPRIETARY
See Enclosure #4

Response: A1.2

See Enclosure #4

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.

SAR 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.

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 the 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 the 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, see SAR Tables A3.3-28 and A3.3-29. Most importantly, the calculated effective multiplication factor (k_{eff}) is based on the "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, see SAR Figure A3.3-26. 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 dioxide 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 dioxide replacement rods are identical to the fuel pins they are replacing except that natural uranium dioxide pellets are used instead of enriched uranium dioxide 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 dioxide 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 dioxide 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 dioxide 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 dioxide 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, the 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: 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 ASME reference should have been "Appendix F, Section F-1342(c)".

The third paragraph of Section A4.2.3.3.3 will be revised to correct a typographical error and to make an editorial change.

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 where the gamma shield cylinder contacts the corner of the Bottom Shield Plate (see Figure A4.3-1). In the remaining large portion of the cross section, the 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 the 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.

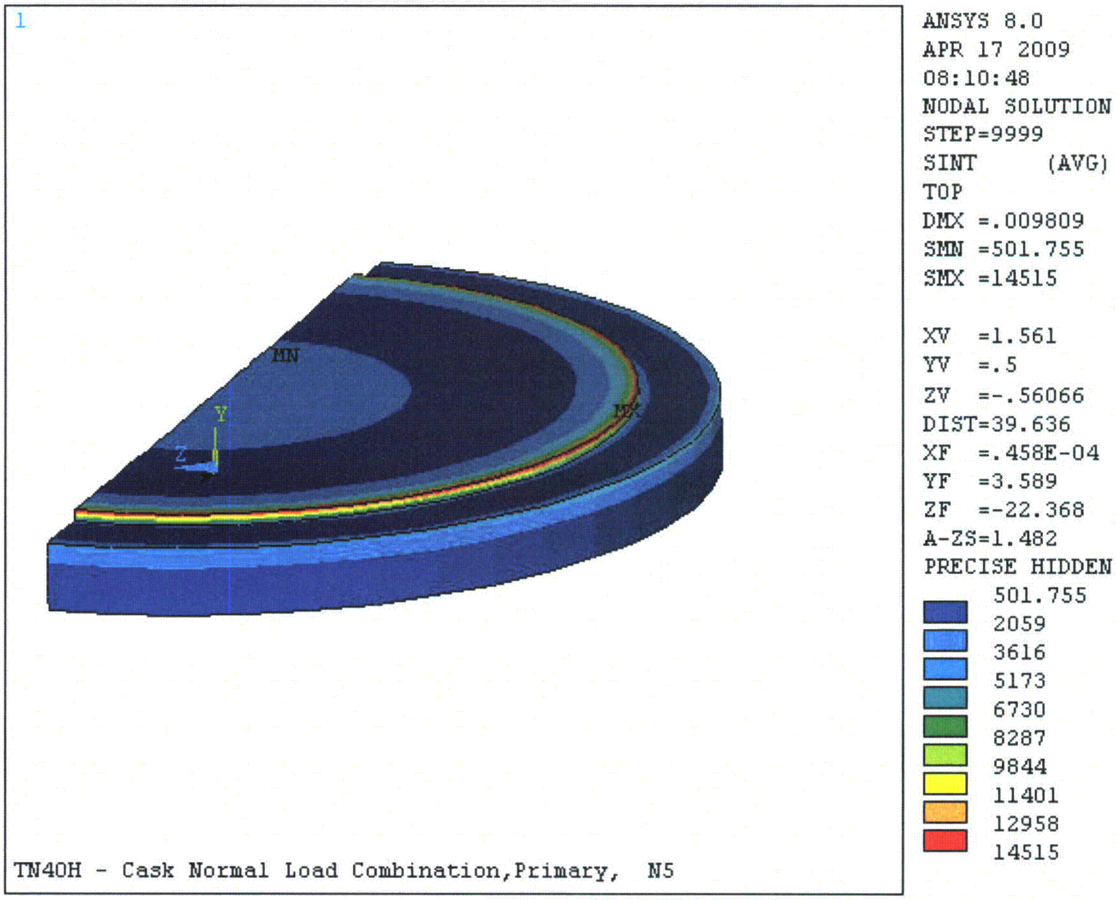


Figure A4.3-1 Bottom Shield Plate, Nodal Stress Intensity Distribution

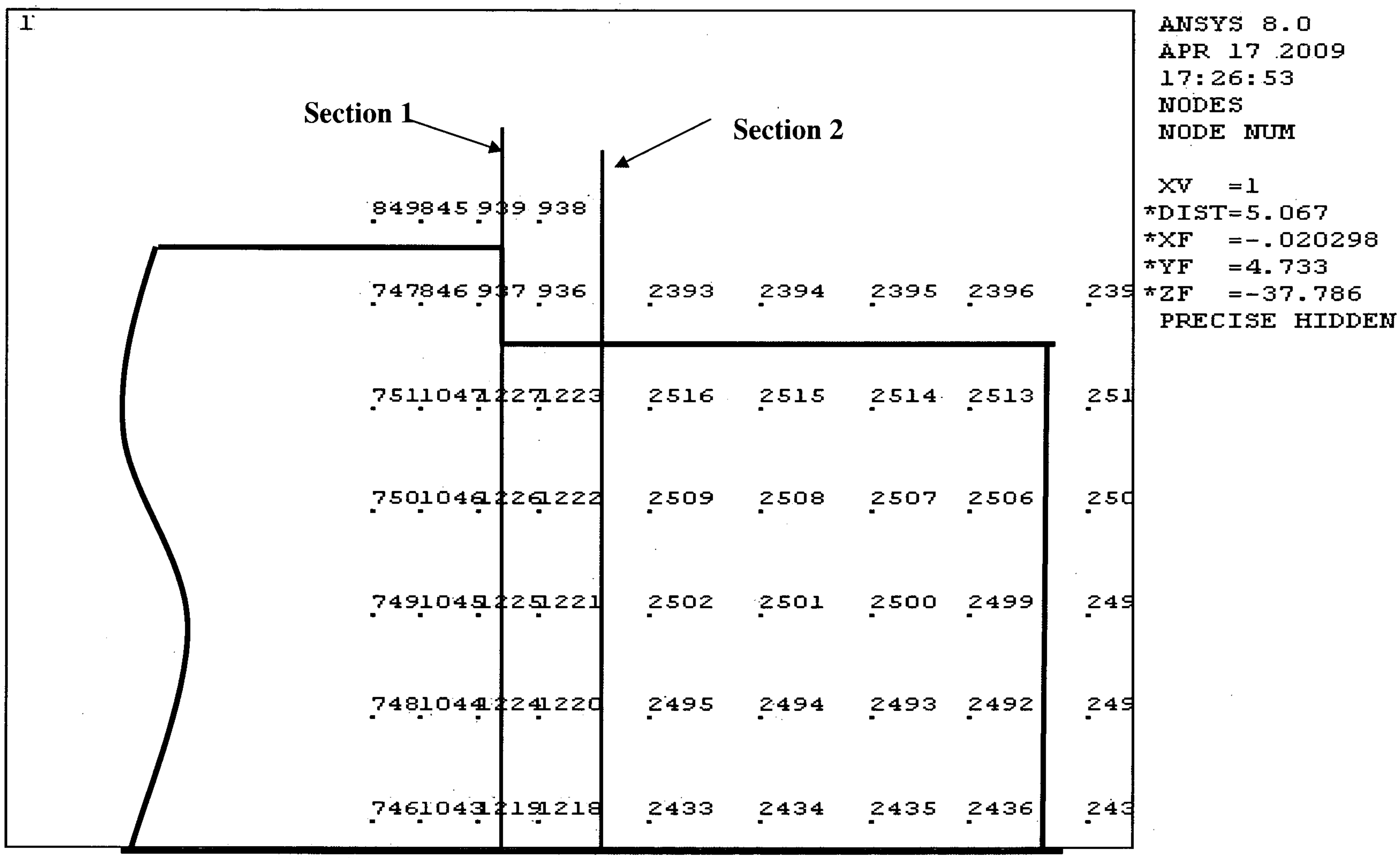


Figure A4.3-2 Bottom Shield Plate, Cross Section Locations

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
```

RAI: A4.4

PROPRIETARY
See Enclosure #4

Response: A4.4

See Enclosure #4

RAI: A4.5

PROPRIETARY
See Enclosure #4

Response: A4.5

See Enclosure #4

RAI: A4.6

PROPRIETARY
See Enclosure #4

Response: A4.6

See Enclosure #4

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 SAR 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 and A4B.1-22 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. For the transition rails and cask interface the gap is varied in the circumferential direction such that it is zero at the point

of contact, which depends on the orientation analyzed, and maximum 180 degrees from the point of contact."

The last paragraph in Section A4B1.5.2.1 will be changed to read as follows.

The boundary conditions and interfaces for a typical fuel compartment are shown in Figure A4B.1-21 and Figure A4B.1-22.

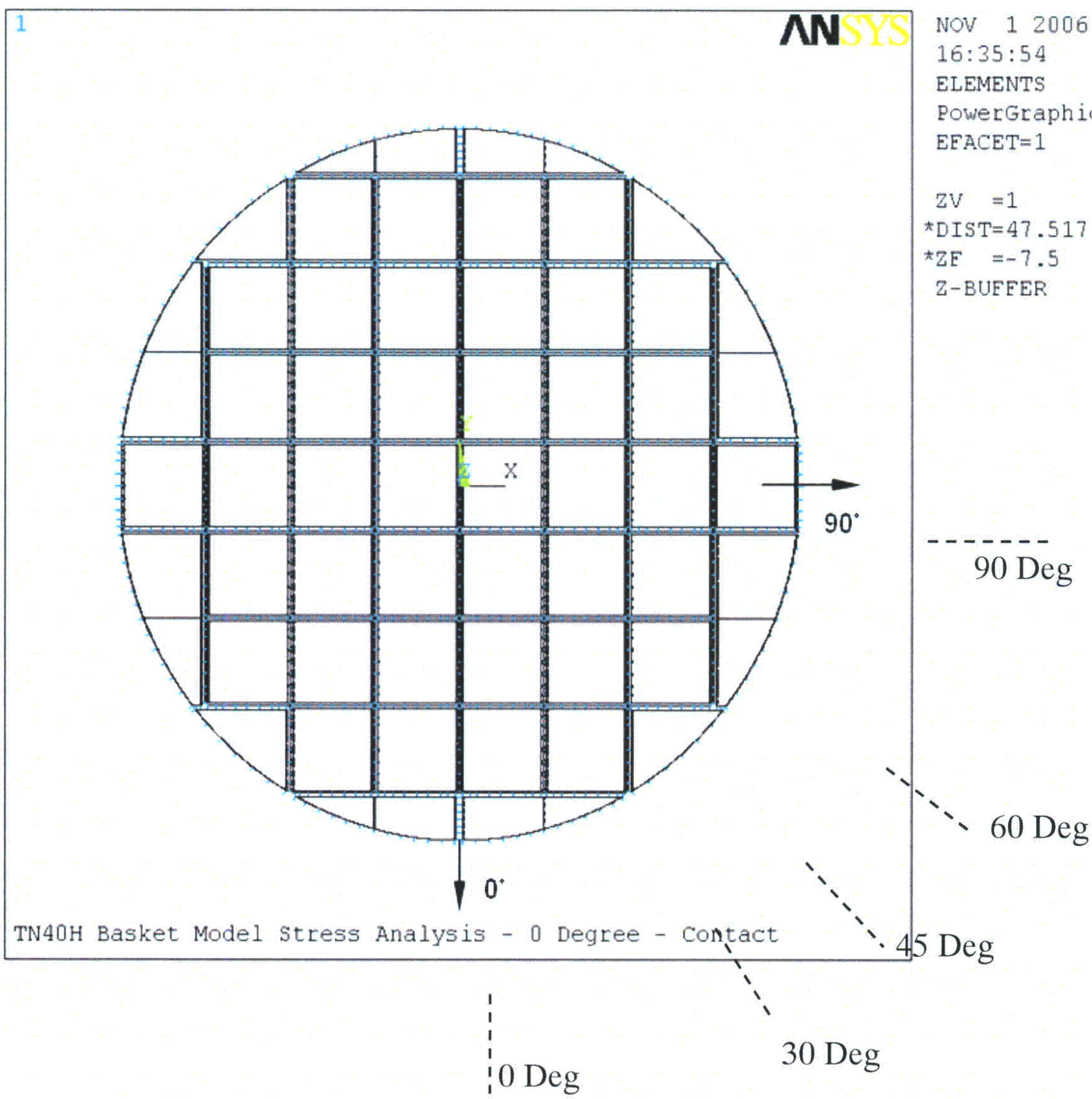


FIGURE A4B.1-2
<u>TN-40HT BASKET LOADING ORIENTATION DEFINITIONS</u>

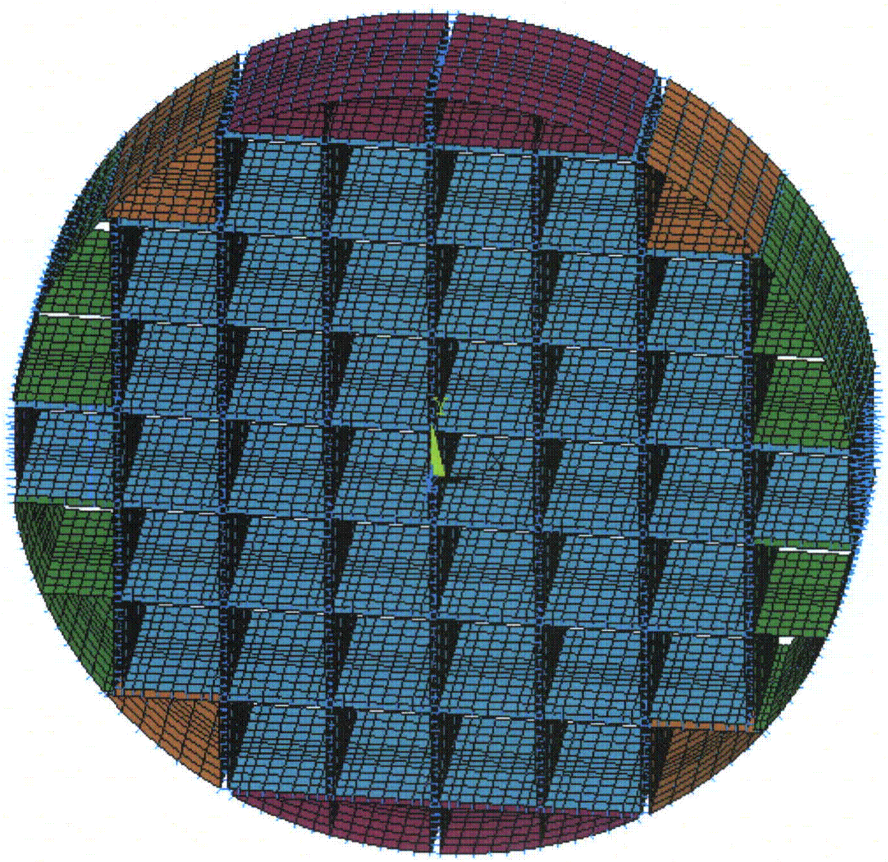


FIGURE A4B.1-3
BASKET FINITE ELEMENT MODEL

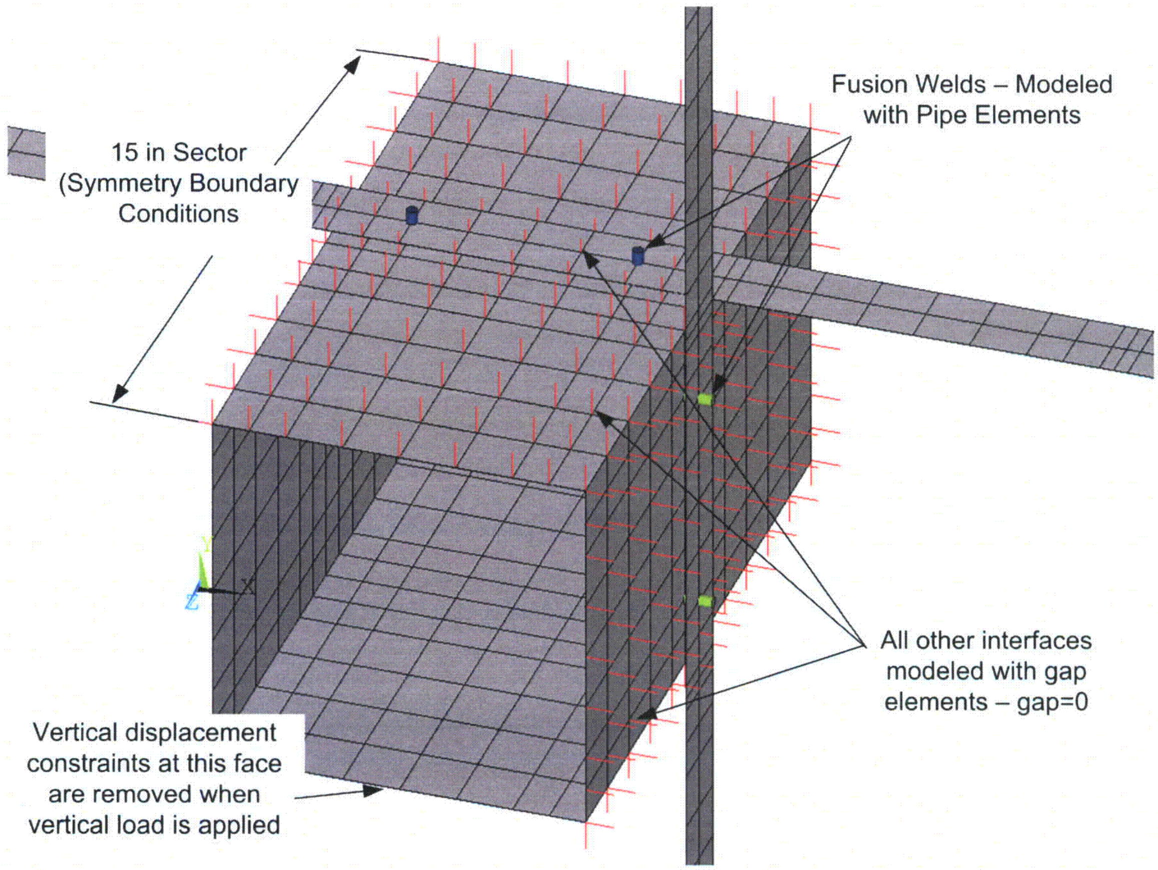


FIGURE A4B.1-21
INTERFACES FOR TYPICAL FUEL COMPARTMENT TUBE

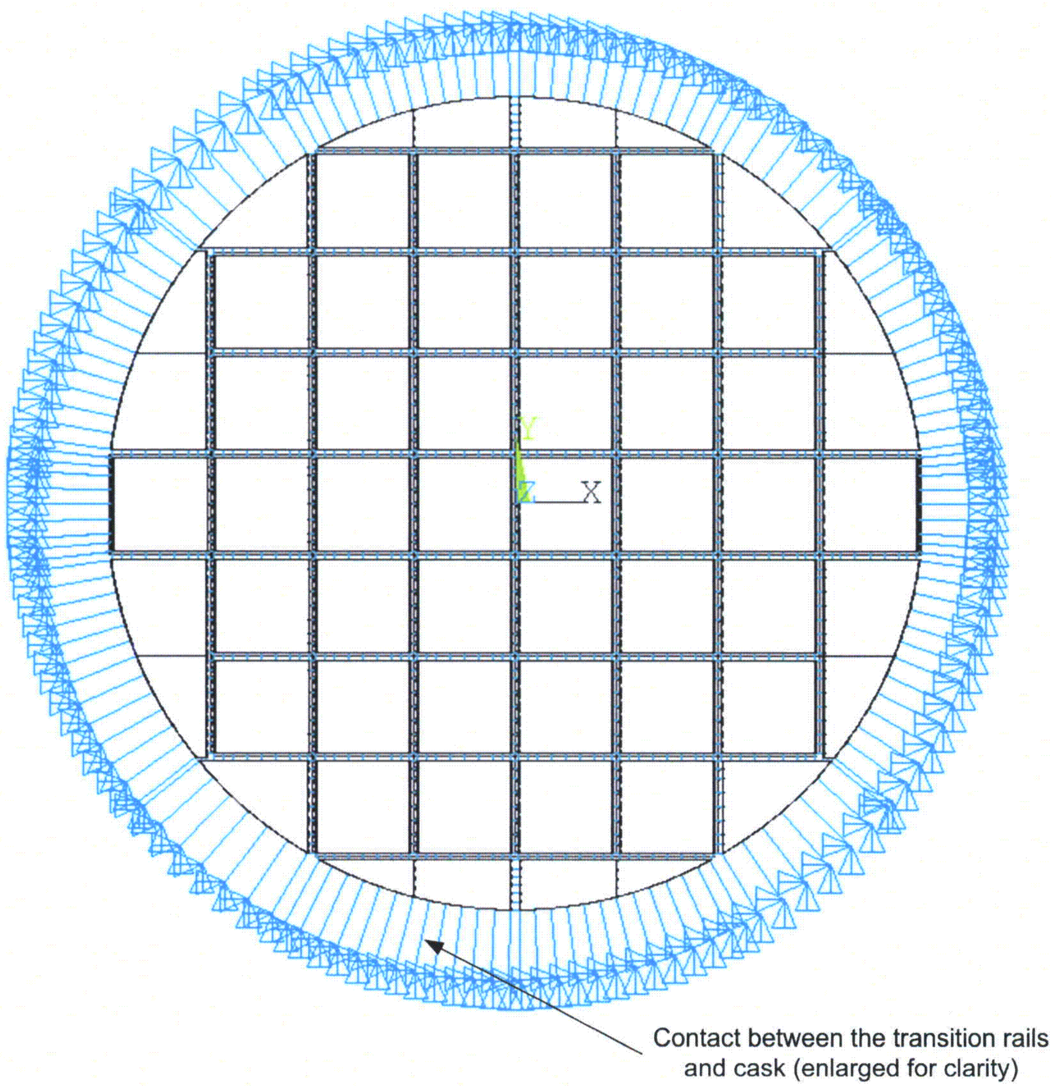


FIGURE A4B.1-22
BOUNDARY CONDITIONS FOR THE BASKET ANALYSIS
(SYMMETRY BOUNDARY CONDITIONS ARE NOT SHOWN FOR CLARITY)

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.

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 5×10^{-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 SAR Section A7A.8.2. (Note that there were two subsections in A7A.8 numbered A7A.8.1. This has been corrected).

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

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

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

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-40HT 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 more limiting radiation source terms. For the TN-40HT cask, a minimum enrichment of 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-40HT cask is employed for this purpose. The response function is utilized to determine the dose rate at 2 meters from the surface of the TN-40HT 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.

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 and Table A7.2-12. 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 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 B9 and B10 represent BECT combinations at enrichments of 2.1 wt. % U-235 and 1.0 wt. % U-235. Their results are also bounded by A7.

The results of the sensitivity calculations – "C" and "D" cases are shown in Table A7.2-12. 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 (545°F) to 590 K (602°F, 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.

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

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

Primary Gamma Energy Range (MeV)	Response Function ((mrem/hour) per particle) per cask
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 (1)	10.10	9.10	19.20
Ratio	0.75	0.80	0.77

(1) The neutron, gamma and total dose rates are obtained as an average of the dose rates shown in Table A7A.5-2. The dose rates at axial height ranging from -22.9 cm to 27.8 cm are included in the average calculations.

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
B9	44	2.1	12.0	687	10.03	14.39	24.42
B10	19	1.0	12.0	272	1.25	7.30	8.55

Table A7.2-12 Fuel Qualification Sensitivity 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
	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

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 2.7778 (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, (note changes are in Bold):

> <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 **~~each~~** 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 ~~three~~** cycles of operation. The model assumes that the BPRA is irradiated in an assembly **~~each~~** with an initial enrichment of 3.85 weight % U-235. The fuel assembly containing the BPRA is burned for **two ~~three~~** cycles with a burnup of 15**~~0~~** 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: A7.3

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 A7.2-12. Cases C1 through C8 are sensitivity calculations where the soluble boron concentration in Cases B1 through B7 on Table A7.2-11 was 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, i.e. Case A7 on Table A7.2-11, 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 a part of the 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 (545°F, representative of a core average moderator temperature) to 590 K (602°F, 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 A7.2-12. Cases D1 through D8 are sensitivity calculations where the moderator temperature in Cases C1 through C7 on Table A7.2-12 was increased from 558 K (545°F, representative of a core average moderator temperature) to 590 K (602°F, representative of an average hot leg moderator temperature) and the moderator density was 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 and boron concentration results in an increase (when compared to corresponding cases B1 through B7) 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, i.e. Case A7 on Table A7.2-11, 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 566°F) for the design basis is justified because 566°F is representative of a core average moderator temperature.

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 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. One of the last steps in the cask draining process calls for the removal of the Hansen fitting in the drain port (Item 35 on Section D-D on Drawing TN40HT-72-6) 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. The streaming path (approximately ¾ inch in diameter) is directly vertical and has little affect on the general area dose rates around the cask lid and flange area, which is where workers would be located. 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 of 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 procedures call for the Radiation Protection Department to 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 procedures and 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.

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

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 to be in the vicinity of the top and bottom ends of the fuel assembly and not confined to the central region (see SAR Table A7A.2-1 and Table A7A.5-1). 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 <u>Additional Fuel Characteristics for Fuel Stored in a TN-40HT Cask</u>

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 be ≥ 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

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 c~~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. 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 ($18 \leq t \leq 40$)
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.

As stated in 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 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 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: 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

Transient dynamic analysis of the cask for the 18-inch end-drop using LS-DYNA was performed to determine decelerations and will be added to new SAR Section A4A.10. The calculation to determine decelerations using EPRI NP-7551 will be removed from Section A8.2.8.2.1.

SAR Section A8.2.8.2.1 will be replaced with the following:

> The peak decelerations in the cask and basket during the 18 inch end drop were calculated by a dynamic nonlinear analysis described in Section A4A.10. The analysis showed a maximum acceleration in the TN40HT cask body of 44.1g. This occurred in the bottom plate. The highest acceleration in the basket and fuel was 28.8g. However, since the basket and fuel were not modeled explicitly, the maximum acceleration (28.8g) must be multiplied by the dynamic load factor of 1.52 resulting in a maximum loading of 43.8g.

The following will be added to new SAR Section A4A.10.

A4A.10 TN-40HT STORAGE CASK END DROP ANALYSIS

The purpose of this section is to determine the rigid body accelerations for the TN-40HT Cask during a vertical drop height of 18 inches on concrete.

The rigid body transfer cask accelerations were predicted numerically by the LS-DYNA 3D explicit nonlinear dynamic analysis finite element solver, Version 9.71s (Reference 18). The methodology used in performing this analysis is based on work conducted at the Lawrence Livermore National Laboratory (LLNL), where an analysis methodology was developed and validated through comparisons with test data (Reference 19 and Reference 20).

The results of these analyses are used as input to the detailed analyses for the cask body, internal basket and fuel assemblies.

A4A.10.1 FINITE ELEMENT MODEL DESCRIPTION

The ANSYS finite element model of the TN-40HT Cask developed for the cask stress analysis (Appendix A4A.3) was simplified for use in the dynamic impact analysis. The TN-40HT Cask model consists of the cask body, simplified basket structure, concrete pad and soil. Each of these components was modeled using 3D 8-node brick elements. Fully integrated selectively-reduced solid elements were used for all elements to reduce the risk of hourglassing problems.

The finite element model was developed with ANSYS and transferred to LS-DYNA. Modifications were made to the LS-DYNA input file to add the material definitions, non-reflecting boundaries and equation of state into LS-DYNA. Features of the cask, such as the trunnions and neutron shield were neglected in terms of stiffness but their weight was lumped into the density of the cask.

The fuel and basket were modeled as a solid cylinder inside the cask walls with elastic material properties approximately equivalent to that of the structure as a whole.

The geometry of the cask finite element model including the cask internals, concrete and base soil is shown in Figure A4A.10-1 and Figure A4A.10-2.

Only ½ of the cask, internals, concrete and soil were modeled, because the entire arrangement is symmetric about the *x-y* plane. The concrete modeled was 16'-8" long, 6'-8" wide, and 3' thick, and the soil modeled was 66'-8" long, 18'-9" wide, and 39'-2" deep.

A4A.10.2 MATERIAL PROPERTIES

The material properties required to perform the analysis include modulus of elasticity, E, Poison's Ratio, ν, and material density (ρ) for the cask body, basket, concrete, and soil. The concrete pad requires a more detailed material model since all of the significant nonlinear deformations occur in the concrete. Material properties used for the concrete and soil were based on those developed at Lawrence Livermore National Labs (Reference 19 and Reference 20).

All material properties were taken at room temperature. This is considered conservative because the cask loaded with spent fuel will typically reach temperatures higher than room temperature, and the lower modulus of elasticity at higher temperatures tends to soften the impact and consequently lower the computed g-loads.

TN-40HT Cask Material

The cask material properties were the same at those used in Appendix A4A.3. All cask materials were modeled as elastic.

Cask Component	Elastic Modulus (psi)	Density (lb-sec^2/in^4)	Poisson's Ratio
Lid Outer Plate	$27.8X10^6$	$8.230x10^{-4}$	0.3
Shield Plate	$29.0X10^6$	$8.230x10^{-4}$	0.3
Shell Flange	$27.8X10^6$	$7.324x10^{-4}$	0.3
Shell	$29.0X10^6$	$9.394x10^{-4}$	0.3
Bottom Plate	$29.0X10^6$	$7.324x10^{-4}$	0.3
Inner Liner	$27.8X10^6$	$7.324X10^{-4}$	0.3

Fuel and Basket Material

The basket structure material properties were the same as those used in Reference 20 except for density. The density of the basket was adjusted to calibrate the overall weight of the cask and basket assembly. The basket was modeled as elastic.

$E = 2.8X10^6$ psi
$v = 0.3$
$\rho = 3.215X10^{-4}$ lb sec^2/in^4

Total modeled weight of the cask and basket is 121,174 lbs since it is a half model. Therefore the total modeled weight is 242,348 lbs. Total actual weight of the cask and basket is 242,400 lbs.

Concrete Material

The concrete was modeled using material law 16 in LS-DYNA, which was developed specifically for granular type materials. The concrete data used in the analysis was originally designed by LLNL for the Shippingport Station Decommissioning Project in 1988. This model was also used in the LLNL (Reference 19) cask drop analysis. Material constants were implemented into Material Model 16, Mode II.B in LS-DYNA. The material represents 4,200 psi compressive strength concrete. A summary of the input used in the analysis is as follows.

$\rho = 2.09675 \times 10^{-4}$ lb sec^2 / in^4
$v = 0.22$
$a_0 = 1606$
$a_1 = 0.418$
$a_2 = 8.35 \times 10^{-5}$ psi^{-1}
$b_1 = 0$
$a_{0f} = 0.0$ psi
$a_{1f} = 0.385$

Effective Plastic Strain versus Scale Factor for Concrete Material

Effective Plastic Strain	Scale Factor, *v*
0	0
0.00094	0.289
0.00296	0.465
0.00837	0.629
0.01317	0.774
0.0234	0.893
0.04034	1.0
1.0	1.0

The maximum principal stress tensile failure cutoff was set at 870 psi. Strain rate effects were neglected in the analysis. Dilger (Reference 21) suggests that the major impact of strain rate effects is in the softening part of the stress-strain curve. Since the purpose of these analyses is primarily to predict the peak accelerations, the strain rate effects on the material behavior may be neglected.

The pressure-volume behavior of the concrete was modeled with the following tabulated pressure versus volumetric strain rate relationship using the equation of state feature in LS-DYNA.

Tabulated Pressure versus Volumetric Strain Rate for the Concrete Material

Volumetric Strain, ε	Pressure (psi)
0	0
-0.006	4,600
-0.075	5,400
-0.01	6,200
-0.012	6,600
-0.02	7,800
-0.038	10,000
-0.06	12,600
-0.0755	15,000
-0.097	18,700

An unloading bulk modulus of 700,000 psi was assumed to be constant at any volumetric strain, as was assumed in Reference 19.

One percent deformation was assumed in the concrete pad to account for the pad reinforcement.

The material properties used for the reinforcing bar are as follows.

$E = 30\times10^6$ psi
$v = 0.3$
$S_y = 30{,}000$ psi
Tangent Modulus, $E_T = 30\times10^4$ psi

Soil Material

The Lawrence Livermore National Labs report (Reference 20) and Brookhaven National Laboratory report (Reference 23) indicates that the stiffness of the soil has little impact on the peak accelerations predicted in the cask. Thus the same soil model was assumed as that used in the Livermore report. The soil material properties assumed for the analysis are:

$E = 6{,}000$ psi
$v = 0.45$
$\rho = 2.0368 \times 10^{-4}$ lb-sec^2 / in^4

A4A.10.3 BOUNDARY CONDITIONS

Only ½ of the cask was modeled with symmetry boundary conditions used to simulate the full structure. Non-reflecting boundaries were applied to the bottom and sides of the modeled soil not aligned with the plane of symmetry (bottom, left side, right side, and back) to prevent artificial stress waves from reflecting back into the model. Both dilatation and shear waves were damped as described in the LS-DYNA *BOUNDARY command.

An automatic surface to surface (contact_automatic_single_surface) contact definition was applied between all parts except the soil. The contact definition has a 0.5 penalty stiffness scale factor to prevent excessive contact stiffness leading to unrealistic part accelerations. A surface to surface (contact_surface_to_surface) contact definition was applied between the concrete and the soil with soft contact option 2. Soft contact option 2 was necessary between the soil and concrete as the materials have very different material stiffness. A conservatively low coefficient of friction (static and kinetic) of 0.25 was applied between all contact surfaces. It is conservative to use a low value for the coefficient of friction because less energy is absorbed due to friction resulting in greater impact acceleration forces.

A4A.10.4 INITIAL CONDITIONS AND LOADING

The analysis begins with a 1" gap between the cask and concrete to allow for at least 5 ms of zero acceleration other than gravity. An initial velocity was applied to all parts of the cask model. The initial velocity was computed by equating potential and kinetic energies. Due to the initial 1" gap and gravitational acceleration, initial velocities were computed 1" shorter than the drop heights.

V = potential energy = mgh
T = kinetic energy = $\frac{1}{2}mv^2$

For an 18" Drop:
$mgh = \frac{1}{2}mv^2$
$\Rightarrow v = \sqrt{2gh} = \sqrt{2(386.4)(18-1)} =$ 114.62 in./sec.

A gravitational acceleration of 386.4 in/sec^2 was applied to the cask and basket model.

A4A.10.5 RESULTS OF LS-DYNA ANALYSES

The resulting rigid body acceleration time histories were computed by LS-DYNA. The rigid body accelerations were computed for the bottom plate, circumferential shell, and basket representation. The parts can be seen in Figure A4A.10-3.

The peak filtered accelerations and corresponding time history plot for different parts of the TN-40HT cask 18" end drop are listed below. All results were filtered with a 4th order low pass butterworth filter with a 350Hz cutoff frequency.

Results Summary

Part	Peak Acceleration (g)	Time History Figure Number
Shell	41.5	A4A.10-4
Bottom Plate	44.1	A4A.10-5
Basket Representation	28.8	A4A.10-6

Based on the Results shown in the above table, the maximum acceleration in the TN-40HT cask during the 18 inch accident condition end drop event is 44.1g and occurs in the bottom plate. Also from this table, the highest acceleration in the basket and fuel is 28.8g. However, since the basket and fuel were not modeled explicitly, the maximum acceleration (28.8g) must be multiplied by the appropriate dynamic load factor (DLF). The maximum DLF for a triangular load is 1.52 (Reference 24). This results in a maximum loading of 43.8g.

References:

18. LS-DYNA Keyword User's Manual, Volumes 1 & 2, Version 9.71s, Rev. 7600.398 August 17, 2006, Livermore Software Technology Corporation.

19. Witte, M. et. Al. Evaluation of Low-Velocity Impact Testing of Solid Steel Billet onto Concrete Pads and Application to Generic ISFSI Storage Cask for Tipover and Side Drop. Lawrence Livermore National Laboratory. UCRL-ID-126295, Livermore, California. March 1997.

20. NUREG/CR-6608, UCRL-1D-12911," Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel/Billet onto Concrete Pad," LLNL, February, 1998

21. Dilger, etc., Ductility of Plain and Confined Concrete under Different Stain Rates, ACI Journal, January-February, 1984.

22. Not Used.

23. BNL-NUREG-71196-2003-CP, "Impact Analysis of Spent Fuel Dry Casks Under Accidental Drop Scenarios," BNL, 2003.

24. Methods for Impact Analysis of Shipping Containers, NUREG/CR-3966, UCID-20639, LLNL, 1987.

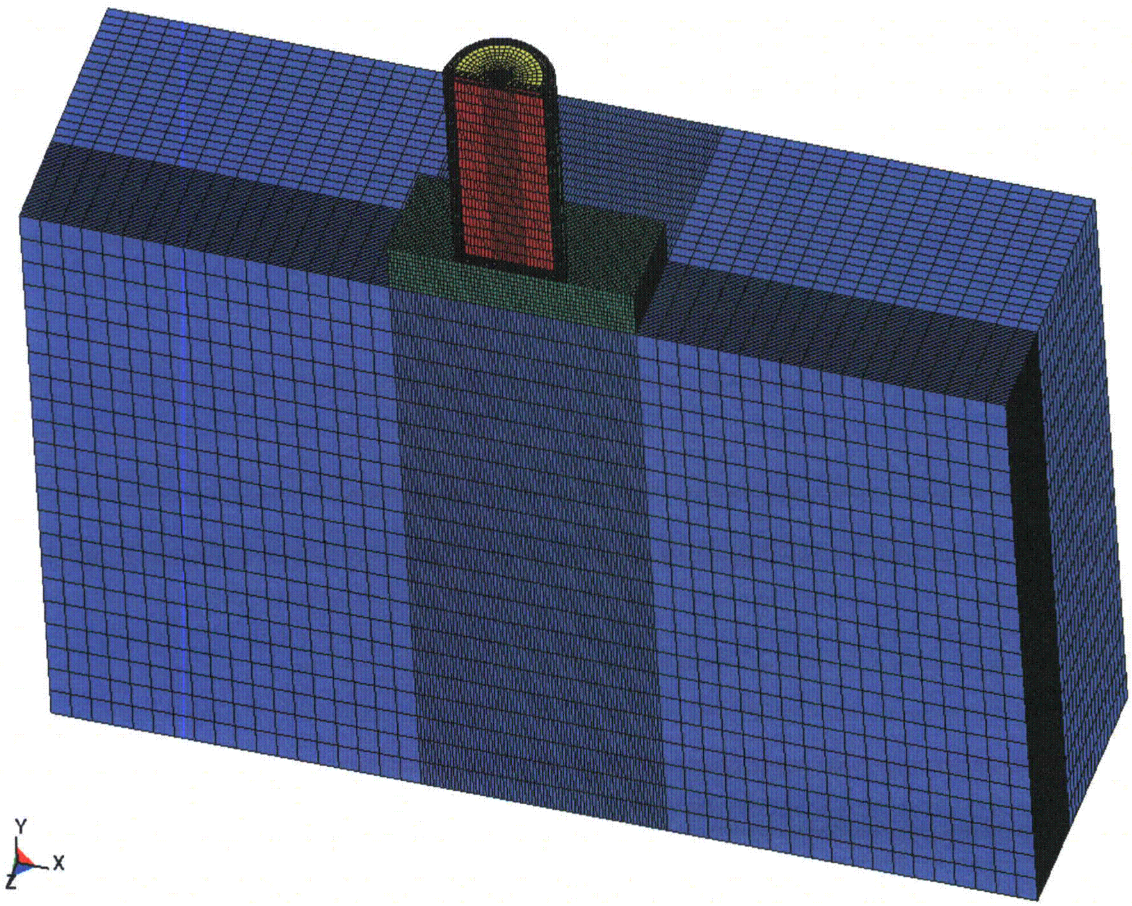


FIGURE A4A.10-1
OVERVIEW OF FINITE ELEMENT MODEL

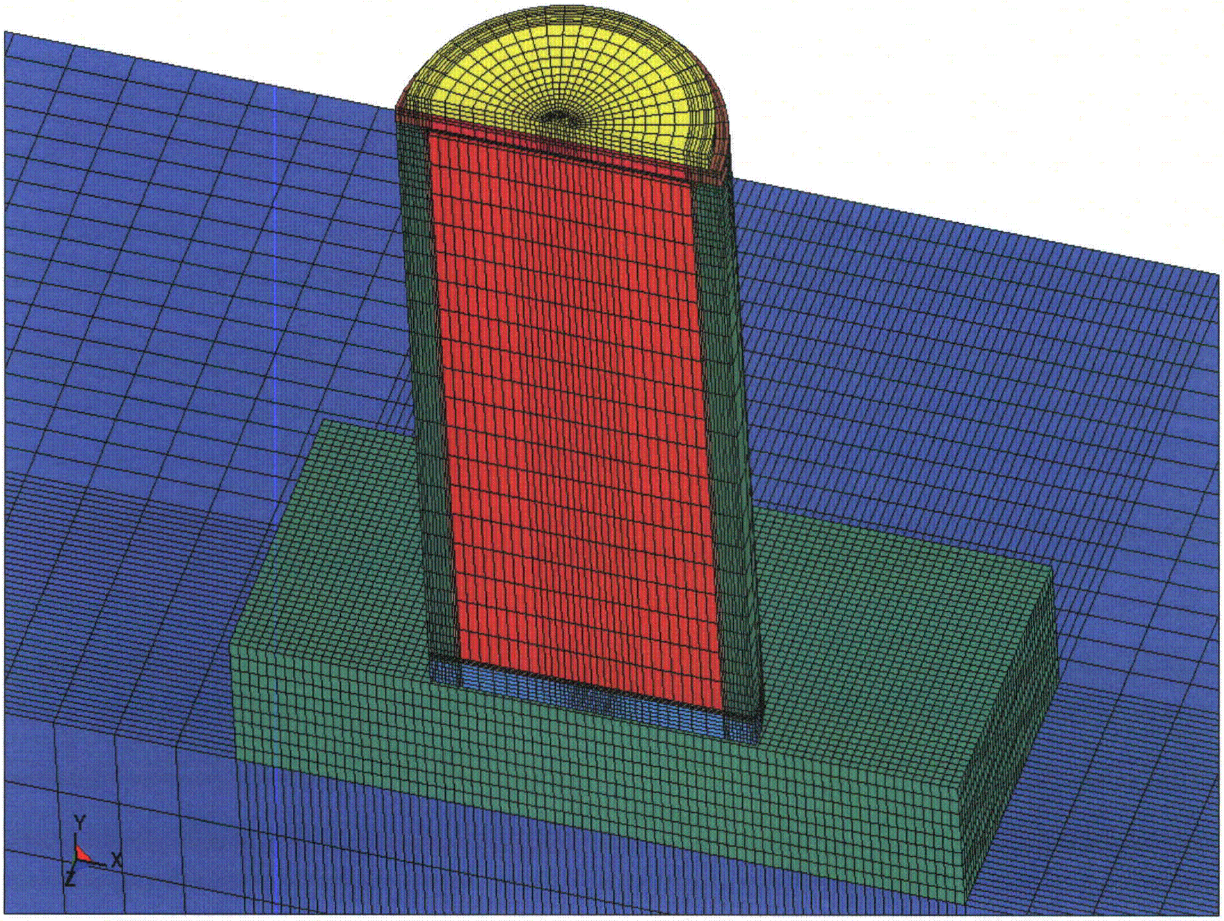


FIGURE A4A.10-2
OVERVIEW OF TN-40HT CASK FINITE ELEMENT MODEL

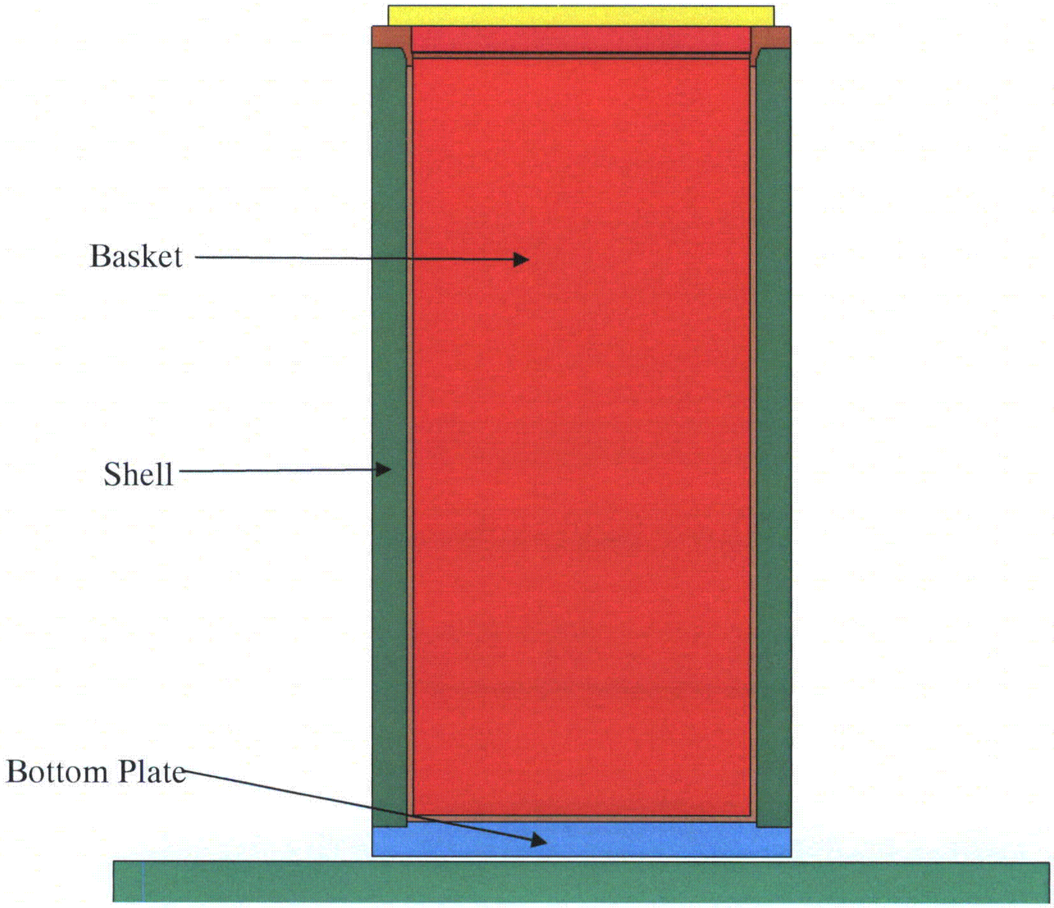


FIGURE A4A.10-3
PARTS ANALYZED FOR ACCELERATION TIME HISTORY

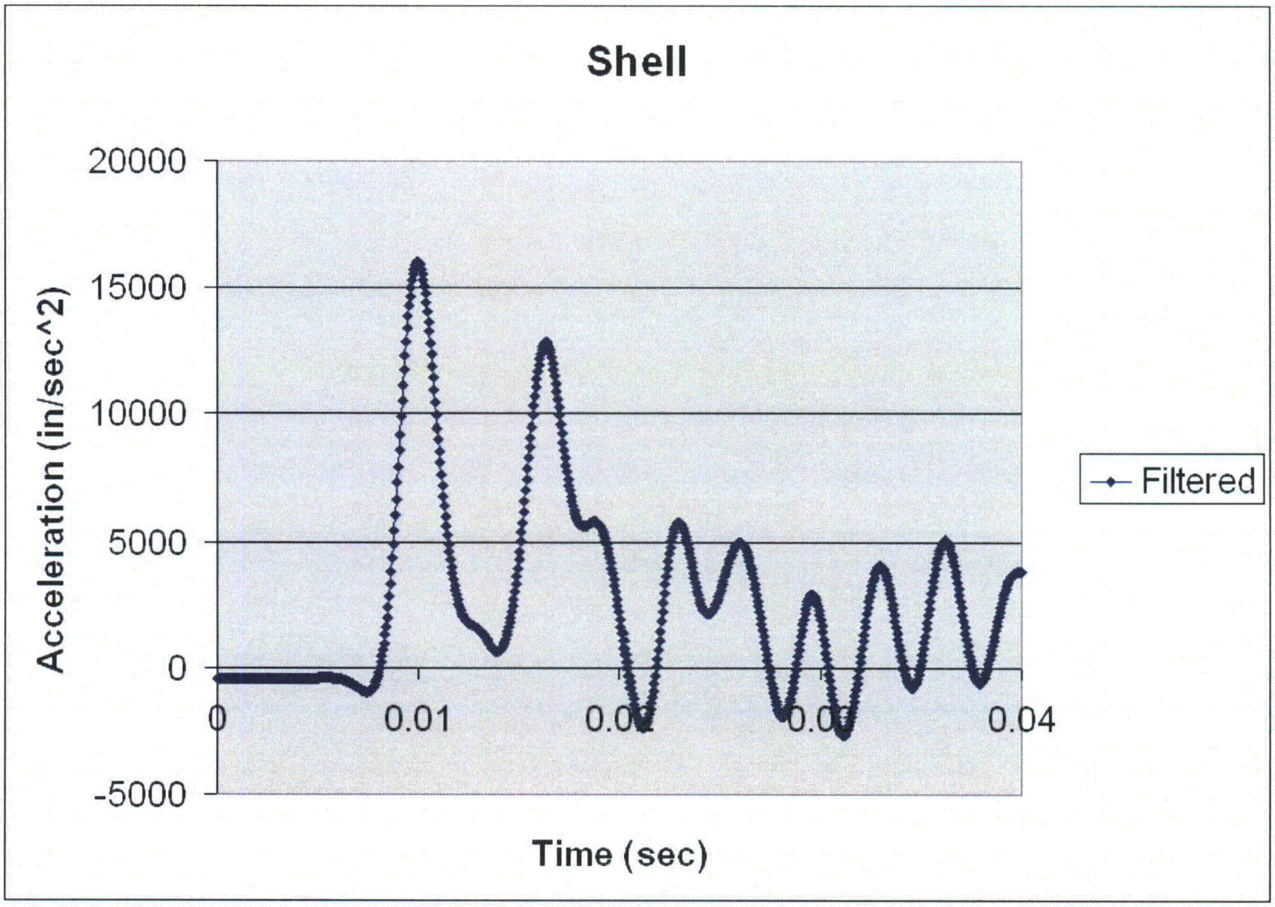


FIGURE A4A.10-4
CASK SHELL ACCELERATION TIME HISTORY (350HZ FILTER)

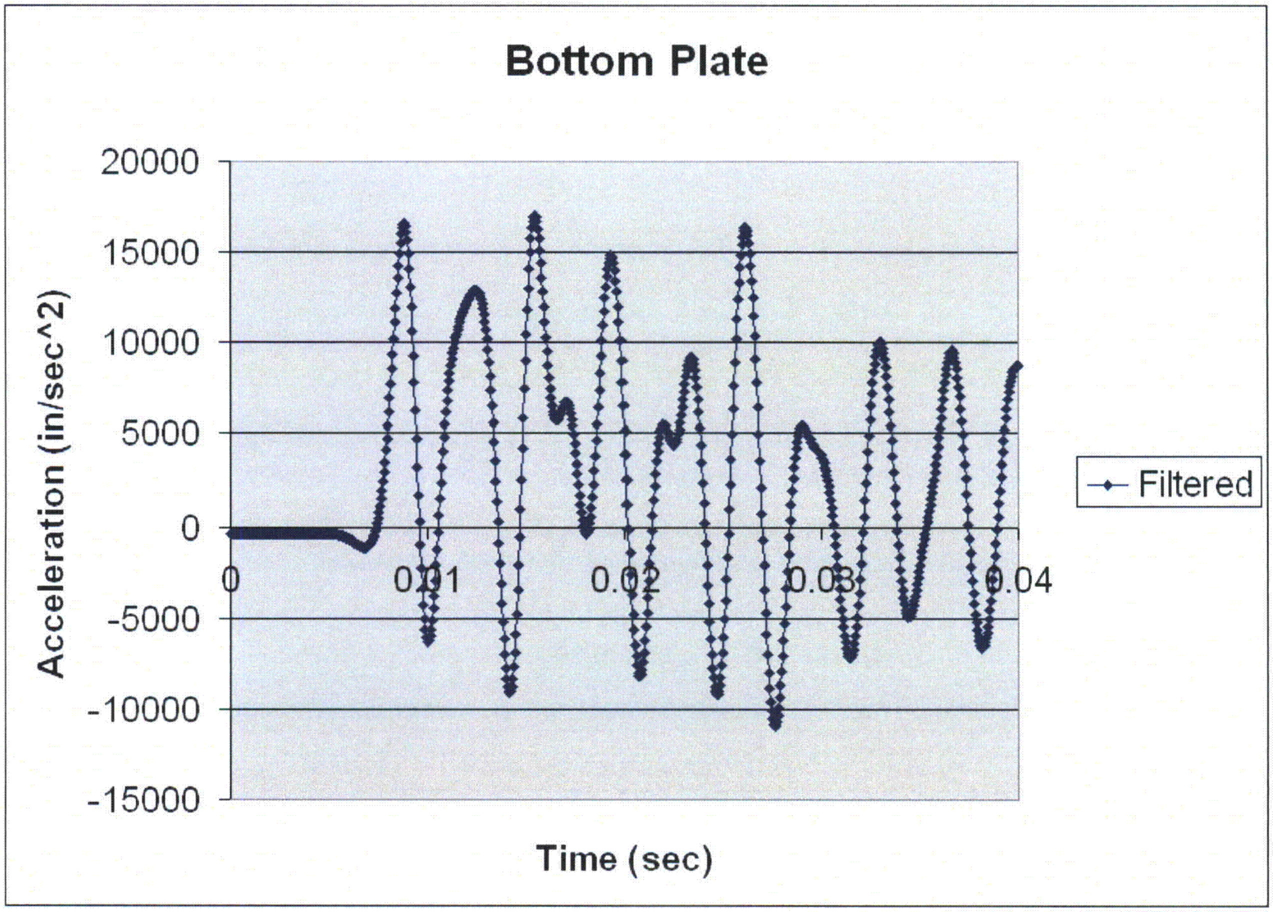


FIGURE A4A.10-5
CASK BOTTOM PLATE ACCELERATION TIME HISTORY (350HZ FILTER)

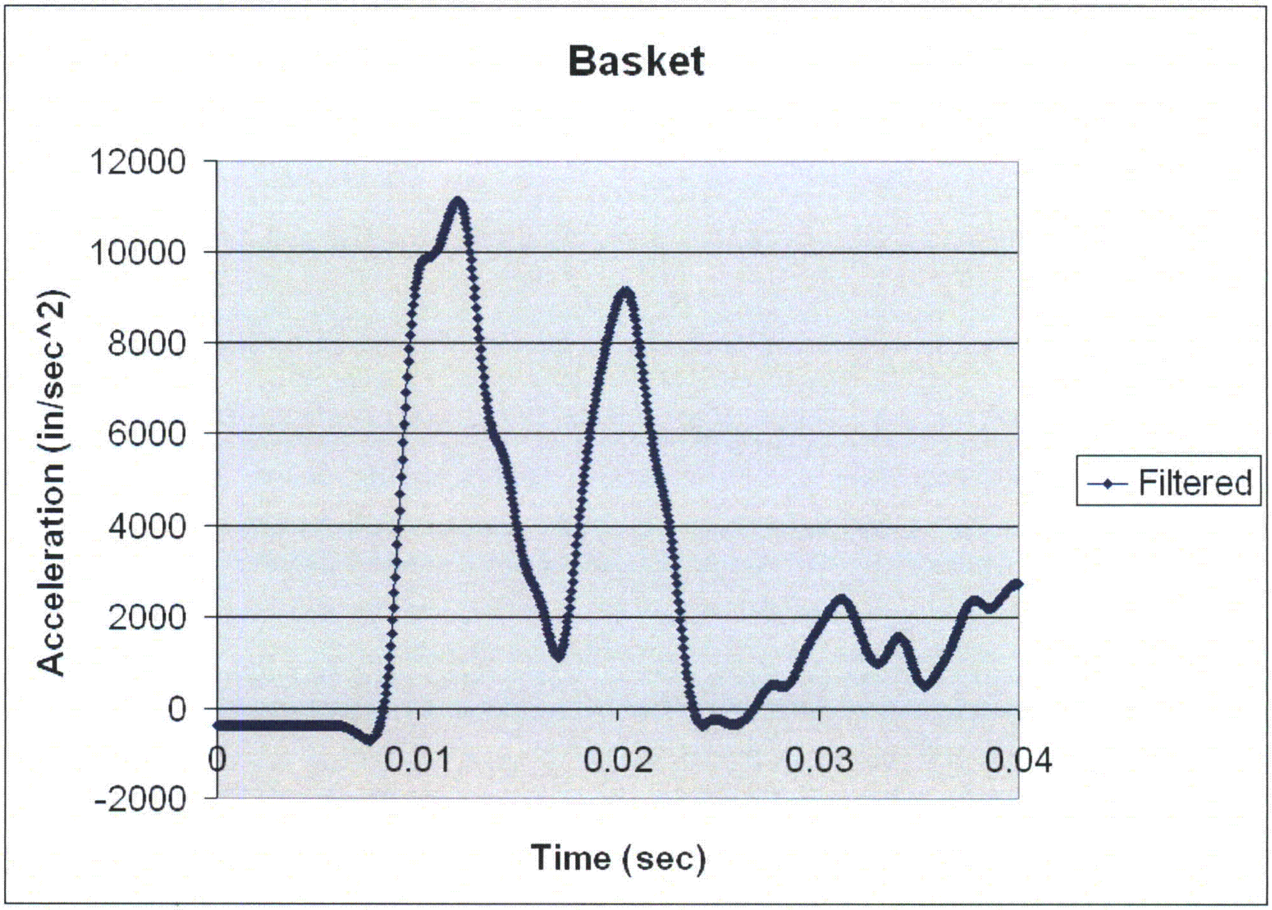


FIGURE A4A.10-6
CASK BASKET ACCELERATION TIME HISTORY (350HZ FILTER)

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.

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.

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.

Other TS Related Changes

The following enhancements/changes were identified during the preparation process of the RAI responses. Although they are not directly related to any RAI question, they are summarized here for completeness.

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 Bases 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 Bases SR 3.4.1.2, Verifying identity of the Fuel and inserts.

Added statement that fuel assembly and insert identity shall be independently verified.

ENCLOSURE 5

SAR, TS, AND TS BASES UPDATING INSTRUCTONS

13 pages follow

SAR, TS, and TS Bases SAR Updating Instructions

SAR Section A1.5 Pages

Remove Page		Insert Page
TN40HT-72-2 Revision 0		TN40HT-72-2 Revision 1
TN40HT-72-21 Sheet 1 of 7 Revision 1		TN40HT-72-21 Sheet 1 of 7 Revision 2
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TN40HT-72-21 Sheet 4 of 7 Revision 1		TN40HT-72-21 Sheet 4 of 7 Revision 2
TN40HT-72-21 Sheet 5 of 7 Revision 1		TN40HT-72-21 Sheet 5 of 7 Revision 2
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TN40HT-72-21 Sheet 7 of 7 Revision 1		TN40HT-72-21 Sheet 7 of 7 Revision 2

SAR Section A3 Pages

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A3.1-2 Revision TBD		A3.1-2 Revision B
A3.3-14 Revision TBD		A3.3-14 Revision B
A3.3-15 Revision TBD		A3.3-15 Revision B
A3.3-17 Revision TBD		A3.3-17 Revision B
A3.3-18 Revision TBD		A3.3-18 Revision B
A3.3-19 Revision TBD		A3.3-19 Revision B
A3.3-20 Revision TBD		A3.3-20 Revision B
A3.3-21 Revision TBD		A3.3-21 Revision B
A3.3-22 Revision TBD		A3.3-22 Revision B
A3.3-23 Revision TBD		A3.3-23 Revision B
A3.3-24 Revision TBD		A3.3-24 Revision B
A3.3-25 Revision TBD		A3.3-25 Revision B
A3.3-26 Revision TBD		A3.3-26 Revision B
A3.3-27 Revision TBD		A3.3-27 Revision B
A3.3-28 Revision TBD		A3.3-28 Revision B
A3.3-29 Revision TBD		A3.3-29 Revision B
A3.3-30 Revision TBD		A3.3-30 Revision B
A3.3-31 Revision TBD		A3.3-31 Revision B
A3.3-32 Revision TBD		A3.3-32 Revision B
A3.3-33 Revision TBD		A3.3-33 Revision B
A3.3-34 Revision TBD		A3.3-34 Revision B
A3.3-35 Revision TBD		A3.3-35 Revision B
A3.3-36 Revision TBD		A3.3-36 Revision B
A3.3-37 Revision TBD		A3.3-37 Revision B
A3.3-38 Revision TBD		A3.3-38 Revision B
A3.3-39 Revision TBD		A3.3-39 Revision B

A3.3-40 Revision TBD		A3.3-40 Revision B
A3.3-41 Revision TBD		A3.3-41 Revision B
A3.3-42 Revision TBD		A3.3-42 Revision B
A3.3-43 Revision TBD		A3.3-43 Revision B
A3.3-44 Revision TBD		A3.3-44 Revision B
A3.3-45 Revision TBD		A3.3-45 Revision B
A3.3-46 Revision TBD		A3.3-46 Revision B
A3.3-47 Revision TBD		A3.3-47 Revision B
A3.3-48 Revision TBD		A3.3-48 Revision B
A3.3-49 Revision TBD		A3.3-49 Revision B
A3.3-50 Revision TBD		A3.3-50 Revision B
A3.3-51 Revision TBD		A3.3-51 Revision B
A3.6-3 Revision TBD		A3.6-3 Revision B
Table A3.1-1 Revision TBD		Table A3.1-1 Revision B
Table A3.3-8, Page 6 of 7 Revision TBD		Table A3.3-8, Page 6 of 7 Revision B

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A4.2-6 Revision TBD		A4.2-6 Revision B
A4.2-12 Revision TBD		A4.2-12 Revision B
A4.2-19 Revision TBD		A4.2-19 Revision B
A4.2-20 Revision TBD		A4.2-20 Revision B
A4.2-21 Revision TBD		A4.2-21 Revision B
A4.2-22 Revision TBD		A4.2-22 Revision B
A4.2-23 Revision TBD		A4.2-23 Revision B
A4.2-24 Revision TBD		A4.2-24 Revision B
A4.2-25 Revision TBD		A4.2-25 Revision B
A4.2-26 Revision TBD		A4.2-26 Revision B
------------		A4.2-27 Revision B
------------		A4.2-28 Revision B
------------		A4.2-29 Revision B
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Table A4.2-2 Revision TBD		Table A4.2-2 Revision B
Table A4.2-26 Revision TBD		Table A4.2-26 Revision B
Table A4.2-27 Revision TBD		------------
Table A4.2-28 Revision TBD		------------
Table A4.2-29 Revision TBD		------------
Figure A4.2-9 Revision TBD		Figure A4.2-9 Revision B
Figure A4.2-10 Revision TBD		Figure A4.2-10 Revision B
Figure A4.2-11 Revision TBD		Figure A4.2-11 Revision B
Figure A4.2-12 Revision TBD		Figure A4.2-12 Revision B
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A4A.10-1 Revision TBD		A4A.10-1 Revision B
A4A.10-2 Revision TBD		A4A.10-2 Revision B
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------------		A4A.10-5 Revision B
------------		A4A.10-6 Revision B
------------		A4A.11-1 Revision B
------------		A4A.11-2 Revision B
------------		Figure A4A.10-1 Revision B
------------		Figure A4A.10-2 Revision B
------------		Figure A4A.10-3 Revision B
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SAR Section A4B Pages

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A4B.1-5 Revision TBD		A4B.1-5 Revision B
A4B.1-6 Revision TBD		A4B.1-6 Revision B
A4B.1-7 Revision TBD		A4B.1-7 Revision B
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Figure A4B.1-2 Revision TBD		Figure A4B.1-2 Revision B
Figure A4B.1-3 Revision TBD		Figure A4B.1-3 Revision B
------------		Figure A4B.1-21 Revision B
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A7.2-2 Revision TBD		A7.2-2 Revision B
A7.2-3 Revision TBD		A7.2-3 Revision B
A7.2-4 Revision TBD		A7.2-4 Revision B
A7.2-5 Revision TBD		A7.2-5 Revision B
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----------		A7.2-7 Revision B
----------		A7.2-8 Revision B
A7.4-2 Revision TBD		A7.4-2 Revision B
Table A7.2-1 Revision TBD		Table A7.2-1 Revision B
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A7A.4-1 Revision TBD		A7A.4-1 Revision B
A7A.4-2 Revision TBD		A7A.4-2 Revision B
A7A.4-3 Revision TBD		A7A.4-3 Revision B
----------		A7A.4-4 Revision B
A7A.7-2 Revision TBD		A7A.7-2 Revision B
A7A.7-3 Revision TBD		A7A.7-3 Revision B
A7A.8-1 Revision TBD		A7A.8-1 Revision B
A7A.8-2 Revision TBD		A7A.8-2 Revision B
A7A.8-3 Revision TBD		A7A.8-3 Revision B
A7A.8-6 Revision TBD		A7A.8-6 Revision B
A7A.8-8 Revision TBD		A7A.8-8 Revision B
A7A.8-9 Revision TBD		A7A.8-9 Revision B
A7A.8-11 Revision TBD		A7A.8-11 Revision B
----------		Table A7A.7-4. Revision B
Table A7A.8-1 Revision TBD		Table A7A.8-1 Revision B
Table A7A.8-3 Revision TBD		Table A7A.8-3 Revision B
Table A7A.8-4 Revision TBD		Table A7A.8-4 Revision B
Table A7A.8-5 Revision TBD		Table A7A.8-5 Revision B
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SAR Section A8 Pages

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A8.2-7 Revision TBD		A8.2-7 Revision B
A8.2-8 Revision TBD		A8.2-8 Revision B
A8.2-9 Revision TBD		A8.2-9 Revision B
A8.2-10 Revision TBD		----------
A8.2-11 Revision TBD		----------
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Figure A8.2-1 Revision TBD		----------

SAR Section A9 Pages

Remove Page		Insert Page
A9.7-1 Revision TBD		A9.7-1 Revision B
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B 3.1.5-3 Revision A		B 3.1.5-3 Revision B
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B 3.3.1-3 Revision A		B 3.3.1-3 Revision B
B 3.3.1-4 Revision A		B 3.3.1-4 Revision B
B 3.4.1-2 Revision A		B 3.4.1-2 Revision B
B 3.4.1-3 Revision A		B 3.4.1-3 Revision B

ENCLOSURE 7

PI ISFSI TECHNICAL SPECIFICATIONS PAGE CHANGES

7 pages follow

1.0 USE AND APPLICATION

1.1 Definitions

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

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor output as practicable to verify the operability of required alarm functions. The COT shall include adjustments, as necessary, of the required alarm setpoint so that the setpoint is within the required range and accuracy.
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;

1.1 Definitions (continued)

DAMAGED FUEL ASSEMBLY (continued)	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).
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a cask while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the cask and end when the cask is supported by the transporter.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the Independent Spent Fuel Storage Installation (ISFSI) while a cask containing one or more spent fuel assemblies is sitting on a storage pad within the ISFSI.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on a cask loaded with one or more spent fuel assemblies when it is being moved to or from the ISFSI. TRANSPORT OPERATIONS begin when the cask is first suspended from the transporter and end when the cask is at its destination and no longer supported by the transporter.

1.1 Definitions (continued)

UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a cask while fuel assemblies are being unloaded. UNLOADING OPERATIONS begin when the cask is no longer supported by the transporter and end when the last fuel assembly is removed from the cask.

2.0 FUNCTIONAL AND OPERATING LIMITS

2.1 Fuel Characteristics for Fuel Stored in a TN-40 or TN-40HT Cask

a. Fuel shall be unconsolidated assemblies;

b. Fuel shall be irradiated at the Prairie Island Nuclear Generating Plant Units 1 or 2;

c. Fuel shall be limited to fuel types:

   i. Westinghouse 14X14 Standard,
   ii. Exxon 14X14 Standard (includes high burnup standard),
   iii. Exxon 14X14 TOPROD, and
   iv. Westinghouse 14X14 OFA (including VANTAGE+);

d. Fuel may include burnable poison rod assemblies (BPRAs) provided:

   i. the BPRA has cooled for ≥ 18 years,
   ii. the cask average cumulative burnup of the fuel assembly(s) where the BPRA(s) resided during reactor operation shall be ≤ 30,000 MWd/MTU;

e. Fuel may include thimble plug devices (TPDs) provided:

   i. the TPD has cooled for a minimum of 16 years,
   ii. the cask average cumulative burnup of the fuel assembly(s) where the TPD(s) resided during reactor operation shall be ≤ 125,000 MWd/MTU;

f. The combined weight of a fuel assembly and any BPRA or TPD shall be < 1330 lbs;

g. The combined weight of all fuel assemblies, BPRAs, and TPDs stored in a single cask shall be < 52,000 lbs;

h. The number of assemblies stored shall be ≤ 40; and

i. The fuel shall not be a DAMAGED FUEL ASSEMBLY.

2.0 FUNCTIONAL AND OPERATING LIMITS (continued)

2.2 Additional Fuel Characteristics for Fuel Stored in a TN-40 Cask

a. The initial enrichment shall be ≤ 3.85 weight percent U-235;

b. The assembly average burnup shall be ≤ 45,000 MWd/MTU;

c. The cooling time prior to loading shall be ≥10 years; and

d. The maximum combined heat load of an assembly and any associated BPRA or TPD shall be < 675 Watts.

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 be ≥ 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:

2.0 FUNCTIONAL AND OPERATING LIMITS

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

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

2.4 Functional and Operating Limits Violations

If any Functional and Operating Limit of 2.1, 2.2, or 2.3 is violated, the following actions shall be completed.

2.4.1 The affected fuel assemblies shall be removed from the cask;

2.4.2 Within 24 hours, notify the NRC Operations Center; and

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

5.0 ADMINISTRATIVE CONTROLS

5.1 General

The Prairie Island ISFSI is located on the Prairie Island Nuclear Generating Plant site and will be managed and operated by Northern States Power Company, a Minnesota corporation (NSPM), staff. The administrative controls shall be in accordance with the requirements of the Prairie Island Nuclear Generating Plant Facility Operating Licenses (DPR-42 and –60) and associated Technical Specifications, as appropriate.

5.2 Environmental Monitoring Program

The licensee shall include the Prairie Island ISFSI in the environmental monitoring program for the Prairie Island Nuclear Generating Plant. An environmental monitoring program is required pursuant to 10 CFR 72.44(d)(2). This program shall include the quarterly determination of ISFSI radiation levels from two (2) thermoluminescent dosimeters on the fence at each side of the ISFSI (8 total).

The licensee shall include the ISFSI in the environmental monitoring report for the Prairie Island Nuclear Generating Plant, and a copy shall be sent to the Director, Office of Nuclear Material Safety and Safeguards

5.3 Annual Environmental Report

An annual report, as required by 10 CFR 72.44(d)(3), shall be submitted to the NRC Region III, Office, with a copy to the Director, Office of Nuclear Material Safety and Safeguards, within 60 days after January 1 of each year. This report should specify the quantity of each of the principal radionuclides released to the environment in liquid and in gaseous effluents during the previous year of operation and such other information as may be required by the Commission to estimate maximum potential radiation dose commitment to the public resulting from effluent release.

ENCLOSURE 8

PI ISFSI TECHNICAL SPECIFICATIONS BASES PAGE CHANGES

10 pages follow

B 2.0 FUNCTIONAL AND OPERATING LIMITS

BASES

BACKGROUND

To protect the integrity of the fuel cladding and ultimately the public from radioactive materials in effluents and direct radiation levels associated with cask operation, the TN-40 and TN-40HT storage cask design requires certain criteria and limits to be placed on the spent fuel parameters for the fuel to be stored in a cask. These criteria and parameter limits include fuel type, initial enrichment, maximum burnup, minimum cooling time, and fuel assembly physical condition (i.e., unconsolidated and not DAMAGED FUEL ASSEMBLY). To limit the associated radiological dose terms from other devices to be stored in casks, i.e., burnable poison rod assemblies (BPRA's) and thimble plug devices (TPD's), similar limitations are placed on BPRA's and TPD's. These criteria and the associated limits are placed on the respective input assumptions used in the thermal, structural, criticality, shielding, and confinement analyses performed for the TN-40 and TN-40HT casks.

APPLICABLE SAFETY ANALYSIS

The applicable safety analyses, as described in the SAR, are the thermal, structural, criticality, shielding, and confinement. The associated Technical Specification criteria and limits are applied to the input assumptions for the specific fuel parameters within these analyses. Within these SAR analyses fuel is considered "Design Bases Fuel" which bounds all specific fuel types to be considered for the TN-40 or TN-40HT. Therefore, the respective SAR analyses do not describe the maximum uranium content for each fuel type. The fuel geometry is determined by the fuel type designation (i.e. 14x14 std, 14x14 TOPROD, 14x14 OFA, etc.). 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.

BASES

ACTIONS (continued)

B.1

If a helium cask environment cannot be achieved and maintained, fuel clad temperatures may increase beyond the analyzed condition. Therefore, the cask will be required to be placed back into the spent fuel pool within 7 days and re-flooded. This time is sufficient time to return the cask to the spent fuel pool and re-flood the cask cavity. Once placed in the spent fuel pool, the fuel is provided adequate decay heat removal to maintain the loaded fuel within limits.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

This Surveillance is modified by a Note. The Note clarifies that meeting the Surveillance is not required, and thus there is not a failure to meet the LCO per SR 3.0.1 and SR 3.0.4 does not apply, prior to the specified Frequency.

Establishment of even a low pressure (i.e. a fraction of a mbar) helium environment satisfies the helium properties described in design basis thermal analyses because thermal conductivity of gases is not pressure dependent until a high vacuum is attained. Thereby, design basis heat removal requirements will be satisfied provided some helium as been introduced to, and maintained in, the cask cavity within the 34 hour vacuum drying time frame analyzed in Reference 3.

SR 3.1.2.2

This Surveillance is modified by a Note. The Note clarifies that meeting the Surveillance is not required, and thus there is not a failure to meet the LCO per SR 3.0.1 and SR 3.0.4 does not apply, prior to the specified Frequency.

BASES

ACTIONS (continued)

B.1

The 30 day Completion Time of Required Action B.1 is based on engineering judgment that any credible seal leak within the 30 day period would not result in a significant loss of helium inventory that would affect the heat removal capability of the cask. In the event of a significant leak, the cask environment would not be reduced to less than one atmosphere of helium because there is no mechanism to exchange the helium in the cask with external air. Based on engineering judgment, this 30 day Completion Time is sufficient to disconnect the test equipment, vent the cask, and return it to the spent fuel pool. Once placed in the spent fuel pool, the fuel is provided adequate decay heat removal to maintain the loaded fuel within limits.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

This Surveillance is modified by a Note. The Note clarifies that meeting the Surveillance is not required, and thus there is not a failure to meet the LCO per SR 3.0.1 and SR 3.0.4 does not apply, prior to the specified Frequency.

A primary design consideration of the cask is that it adequately contain radioactive material and retain an inert environment. The specified helium leak rate for this Surveillance demonstrates that an adequate confinement barrier has been established and that the cask is within design assumptions. The determination of the leak rate shall be done in accordance with ANSI N14.5 (Reference 2). The minimum sensitivity of the leak rate test is 5×10^{-6} atm-cc/sec and the test includes the overpressure system up to the isolation valve.

Measuring the helium leak rate must be performed successfully on each cask prior to placing it in storage. Once the helium atmosphere is established by SR 3.1.2.1, there is enough conduction to maintain

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.3.1 (continued) the loaded fuel within its temperature limits, and to prevent thermal expansion from damaging the basket. Therefore, no time limit is required for this Surveillance, other than completion prior to Transport Operations.
REFERENCES	1. SAR Section A8.2. 2. American National Standards Institute, "National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials", ANSI N14.5-1997, New York, Oct. 1987.

BASES

ACTIONS (continued)

A.1

If the cask interseal pressure is below the limit, an appropriate assessment and evaluation is to be performed to determine the cause of the low pressure condition. The 7-day period is sufficient time to perform an assessment of the condition and make necessary repairs to the overpressure system and reestablish a pressure above the limit. Reestablishing the pressure above the limit prevents leakage of radioactive material from the cask cavity.

B.1

If it is determined that there is a leakage path in the cask seals or overpressure system, a repair is to be performed in a timely manner. If the interseal pressure has been reestablished to 30 psig or above, no leakage of radioactive material from the cask cavity can occur. The 30-day Completion Time of Required Action B.1 provides ample time to implement necessary repairs or for the return of the cask to the spent fuel pool and to be re-flooded. Once placed in the spent fuel pool, the fuel is provided adequate decay heat removal to maintain the loaded fuel within limits. The allowed completion times are bounded by the 45 day exposure duration for off-normal conditions in Reference 3.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

This Surveillance is modified by a Note. This Note clarifies that performing the Surveillance is not required, and thus SR 3.0.4 does not apply, until 24 hours after first completion of SR 3.1.5.2. This Note is necessary to allow entry into STORAGE OPERATIONS and subsequent installation of the necessary monitoring equipment on the ISFSI pad to allow for performing the Surveillance during the STORAGE OPERATIONS.

BASES (continued)

REFERENCES	1.	SAR Section 8.2.
	2.	SAR Section A8.2.
	3.	SAR Section A7A.8

BASES

ACTIONS (continued)	A.2 If the dissolved boron concentration in the spent fuel pool and therefore, the cask cavity, is not within the limit, all fuel assemblies must be removed from the cask. Removal of fuel from the cask places the cask in a condition where this LCO is no longer applicable. The 24-hour Completion Time takes into consideration the time necessary to unload a fully loaded cask.
SURVEILLANCE REQUIREMENTS	SR 3.3.1.1 This SR specifically applies to LOADING OPERATIONS. The boron concentration of the spent fuel pool water is determined prior to commencing cask loading using chemical analysis of two samples analyzed by different individuals (per the requirements of License Condition 15G) to reduce the risk that a single error could lead to not meeting the LCO. The requirement to verify the boron concentration within 4 hours prior to commencing LOADING OPERATIONS ensures that the water added to the cask is within the limit. The Frequency is based on the operating experience that boron concentration changes occur very slowly. SR 3.3.1.2 This Surveillance is modified by a Note. The Note clarifies that meeting the Surveillance is not required, and thus there is not a failure to meet the LCO per SR 3.0.1 and SR 3.0.4 does not apply, prior to the specified Frequency.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

This SR specifically applies to UNLOADING OPERATIONS. The boron concentration is analyzed as described above in SR 3.3.1.1. The requirement to verify the boron concentration within 4 hours prior to flooding the cask for UNLOADING OPERATIONS ensures that the water added to the cask cavity is within the limit. The Frequency is based on operating experience the boron concentration changes very slowly.

REFERENCES

1. SAR, Section A3.3.

BASES (continued)

ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each cask. This Note is acceptable because the fuel loading into one cask is independent of the fuel loaded in subsequent casks or adjacent casks. The Required Actions for each Condition provide appropriate compensatory actions for each cask not meeting the LCO. Subsequent casks that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If a fuel assembly, previously placed in a cask, is found to not meet the specified functional and operating limits, the fuel assembly is to be immediately removed from the cask. The immediate Completion Time reflects the importance of maintaining the protection and integrity of the fuel clad barrier as well as the public from radioactive materials in effluents and direct radiation levels associated with cask operation by only storing fuel in accordance with cask design requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This Surveillance is modified by a Note. The Note clarifies that performing the Surveillance is not required, and thus SR 3.0.4 does not apply, prior to the specified Frequency.

This SR applies prior to inserting the fuel into the cask. The spent fuel assembly compliance with the Functional and Operating Limits is to be demonstrated by administrative verification. This verification applies to fuel assemblies as well as BPRA's or TPD's. Per the requirements of License Condition 15F, satisfying the Functional and Operating Limits shall be independently verified by an individual other than the original individual making the selections. The Frequency is selected to ensure only fuel meeting cask design requirements is inserted into a cask.

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.1.2 This Surveillance is modified by a Note. The Note clarifies that performing the Surveillance is not required, and thus SR 3.0.4 does not apply, prior to the specified Frequency. The spent fuel assembly identity is to be verified once prior to inserting in a cask and once again prior to final closure of the cask. The fuel assembly and insert identity shall be independently verified. This verification applies to fuel assemblies as well as BPRA's or TPD's. The Frequency is selected to ensure only fuel meeting cask design requirements are inserted into a cask.
REFERENCES	None.

ENCLOSURE 10

NON-PROPRIETARY PAGES RETYPED

150 pages follow (not including section dividers)

WITHHELD PURSUANT TO 10 CFR 2.390

THIS DRAWING IS MADE FROM DRAWING TN40HT-72-2

REVISION	DESCRIPTION	DATE
1	REVISED PER RAI ROUND 1	5/22/2008
0	INITIAL ISSUE	5/3/2007

TRANSNUCLEAR
An AREVA Company

SAFETY ANALYSIS REPORT
TN-40HT HIGH BURNUP DRY STORAGE CASK ASSEMBLY

TN40HT-72NP-2 | NONE | 1 OF 1 | 1

TN40HT-72NP-21 1 OF 7

PARTS LIST (24)

ITEM	QTY	DRAWING NO. PART NUMBER	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATION	QUALITY CATEGORY	CODE JURISDICTION
1	1	TN40HT-72-21	BASKET ASSEMBLY			
2	40		FUEL COMAPARTMENT 3/16 STK.	SA240, TYPE 304	SR	NG
3	2		BOTTOM PLATE 7/16 STK.	SA240, TYPE 304 OR SA479, TYPE 304	SR	NG
3A	2		BOTTOM PLATE 5/16 STK.	SA240, TYPE 304 OR SA479, TYPE 304	SR	NG
4	2		BOTTOM PLATE 7/16 STK.	SA240, TYPE 304 OR SA479, TYPE 304	SR	NG
5	3		BOTTOM PLATE 7/16 STK.	SA240, TYPE 304 OR SA479, TYPE 304	SR	NG
6	N/A		NOT USED			
7	20		BASKET PLT. SUPPORT BAR 0°/180° 7/16 STK	SA240, TYPE 304	SR	NG
8	20		BASKET PLT. SUPPORT BAR 0°/180° 7/16 STK.	SA240, TYPE 304	SR	NG
9	30		BASKET PLT. SUPPORT BAR 0°/180° 7/16 STK.	SA240, TYPE 304	SR	NG
10	20		BASKET PLT. SUPPORT BAR 90°/270° 5/16 STK.	SA240, TYPE 304	SR	NG
11	N/A		NOT USED			
12	40		BASKET PLT. SUPPORT BAR 90°/270° 7/16 STK.	SA240, TYPE 304	SR	NG
13	20		BASKET PLT SUPPORT BAR 90°/270° 7/16 STK	SA240, TYPE 304	SR	NG
14	20		ALUMINUM PLATE 90°/270° 5/16 STK.	B209, TYPE 1100	SR	NOTE 23
15	40		ALUMINUM PLATE 90° / 270°	B209, TYPE 1100 (15)	SR	NOTE 23
15A	A/R		POISON PLATE	(15)(17)	SR	NOTE 23
16	N/A		NOT USED			
17	20		ALUMINUM PLATE 90° / 270°	B209, TYPE 1100 (15)	SR	NOTE 23
17A	A/R		POISON PLATE	(15)(17)	SR	NOTE 23
18	20		ALUMINUM PLATE 0° / 180°	B209, TYPE 1100 (15)	SR	NOTE 23
18A	A/R		POISON PLATE	(15)(17)	SR	NOTE 23
19	20		ALUMINUM PLATE 0° / 180°	B209, TYPE 1100 (15)	SR	NOTE 23
19A	A/R		POISON PLATE	(15)(17)	SR	NOTE 23
20	30		ALUMINUM PLATE 0° / 180°	B209, TYPE 1100 (15)	SR	NOTE 23
20A	A/R		POISON PLATE	(15)(17)	SR	NOTE 23
21	2	TN40HT-72-22	R180 TRANSITION RAIL ASSEMBLY		SR	NG
22	2	TN40HT-72-22	R90 TRANSITION RAIL ABOVE AZIM. 90° / 270°		SR	NG
22A	2	TN40HT-72-22	R90 TRANSITION RAIL BELOW AZIM. 90° / 270°		SR	NG
23	2	TN40HT-72-22	R45 TRANSITION RAIL ABOVE AZIM. 90° / 270°		SR	NG
23A	2	TN40HT-72-22	R45 TRANSITION RAIL BELOW AZIM. 90° / 270°		SR	NG
24	280		3/4-10 UNC THREADED STUDS	A-479 TYPE 304	AQ	NG
25	288		LOCK WASHER	SST (25)	NQR	NOTE 23 (1)
26	296		FLAT WASHER	SST	NQR	NOTE 23 (1)
27	288		3/4-10 UNC NUTS	A-194 GR B8	AQ	NG (1)
28	20		BASKET PLT. SUPPORT BAR 90°/270° 5/16 STK.	SA-240 TYPE 304	AQ	NG
29	A/R		BASKET RAIL SHIMS INTERNAL	B209, TYPE 6061	NQR	NOTE 23 (16)
30	2		FUEL COMPARTMENT EXTERNAL SHIM	B209, TYPE 6061 (16)	NQR	NOTE 23
31	20		ALUMINUM PLATE 0°/180° 5/16 STK.	B209, TYPE 1100	SR	NOTE 23
32	22		1/4-28 UNC FLAT COUNTERSUNK HEAD CAP SCREW	A-479 TYPE 304	NQR	NOTE 23

NOTES:

(1) ALTERNATE MATERIAL FOR ATTACHMENT HARDWARE MAY BE USED WITH TRANSNUCLEAR APPROVAL.

2. WELD SYMBOLS ARE PER ANSI/AWS 2.4-98. WELD SIZES ARE MINIMUM. ALTERNATE WELDS OF EQUIVALENT STRENGTH MAY BE USED WITH TRANSNUCLEAR APPROVAL.

(3) 3.50 BOTTOM PLATES (ITEMS 3-5), SHALL BE ORIENTED IN ONE DIRECTION ONLY (0° TO 180°).

(4)

PROPRIETARY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

(5)

(6) A ±.25 MISMATCH BETWEEN DRAIN SLOTS IS ACCEPTABLE. AFTER ASSEMBLY, THE MINIMUM OPENING SHALL BE .62 X .62.

(7) PROPRIETARY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

(8) PRIOR TO BASKET ASSEMBLY, THE FUEL COMPARTMENT SHALL HAVE A MINIMUM OPENING OF 8.05.

(9)

PROPRIETARY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

(10)

(11) THE BASKET RAIL MOUNTING STUDS SHALL BE WELDED TO THE SST SUPPORT BARS AT ALL 11 ELEVATIONS.

(12) RAIL ATTACHMENT TORQUE VALUE IS 30 FT-LBS. STUDS SHALL BE LUBRICATED WITH LOCTITE N-5000 ANTI-SEIZE OR EQUIVALENT.

(13)

PROPRIETARY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

(14)

(15) THE TOTAL COMBINED THICKNESS OF THE ALUMINUM PLATE AND POISON PLATE SHALL BE .435 MINIMUM. THE MAXIMUM TOTAL THICKNESS MAY BE DETERMINED BY THE FABRICATOR BASED ON MEETING THE OVERALL BASKET DIMENSIONS BUT SHALL NOT BE THICKER THAN THE SUPPORT BAR.

ENRICHED BORON ALUMINUM ALLOY OR METAL MATRIX COMPOSITE

BORAL

B-10 CONTENT mg/cm²	
MATERIAL	MINIMUM
TYPE I	37.5
TYPE II	45.0

(16) PROPRIETARY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

(17) POISON PLATE MAY BE FABRICATED IN TWO PIECES TO MAKE UP THE REQUIRED HEIGHT, WITH TRANSNUCLEAR APPROVAL.

18. NOT USED.

(19) CHAMFER BOTTOM OUTSIDE CORNER OF FUEL COMPARTMENT AS REQUIRED TO CLEAR ANY FILLET WELD/RADIUS AT THE CASK CAVITY BOTTOM. IN ADDITION, EACH FUEL COMPARTMENT MAY BE CHAMFERED 1.00 MAX X .10 MAX ALL AROUND BOTTOM END TO AID IN BASKET ASSEMBLY. ALL DETAILS PER TRANSNUCLEAR APPROVAL.

(20) NOT USED.

(21) THE FINISHED LENGTH OF THE RAIL ASSEMBLIES (ITEMS 21 THRU 23A) SHALL NOT EXTEND BEYOND THE TOP EDGE OF THE BASKET ASSEMBLY WITH BOTH ASSEMBLIES RESTING ON THE CASK BOTTOM. THE RAIL MAY BE FLUSH WITH THE BASKET TOP EDGE TO 1/4" BELOW FLUSH.

(22) FABRICATED LENGTH TO BE DETERMINED BY FABRICATOR.

(23) NON-CODE.

(24) QUALITY DESIGNATIONS:
SR - SAFETY RELATED
AQ - AUGMENTED QUALITY
NQA- NON QA RELATED

25. NOT USED.

26. STANDARD TOLERANCES FOR PARTS AND DETAILS UNLESS OTHERWISE SPECIFIED: .XX ±.03 .XXX ±.010 ∠ ± .5°

(27) RAIL SHALL HAVE A CUTOUT TO ACCOMODATE THE DRAIN TUBE CLAMP (DRAWING TN40HT-72-4).

THIS DRAWING IS MADE FROM DRAWING TN40HT-72-21

REVISION	DESCRIPTION	DATE
2	REVISED PER RAI ROUND 1	05/22/09
1	REVISED PER DCR TN40HT-005	10/10/07
0	INITIAL ISSUE	5/3/2007

ALL DIMENSIONS APPLY AFTER WELDING AND FINAL MACHINING UNLESS OTHERWISE SPECIFIED.

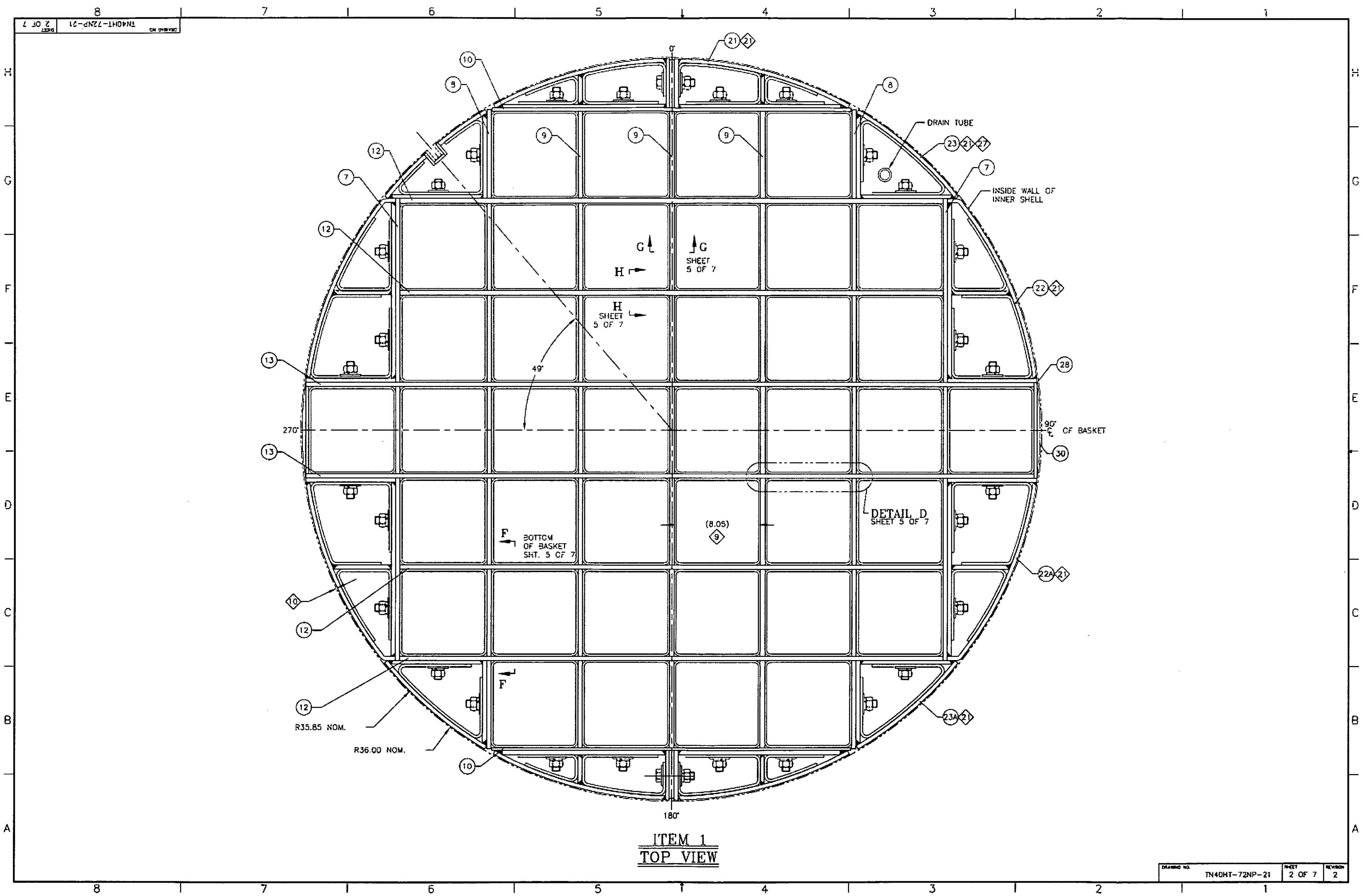
DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M

INTERPRETATION OF WELD SYMBOLS PER ANSI / AWS 2.4

TRANSNUCLEAR
AN AREVA COMPANY

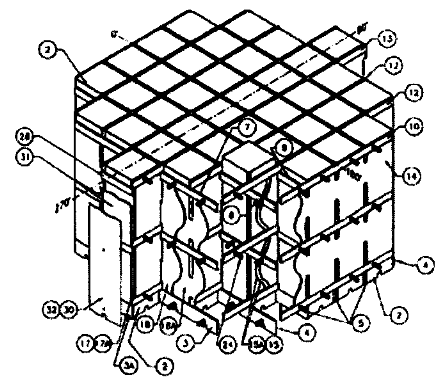
SAFETY ANALYSIS REPORT
TN-40HT
BASKET ASSEMBLY AND DETAILS

DRAWING NO.	SCALE	SHEET	REVISION
TN40HT-72NP-21	NONE	1 OF 7	2



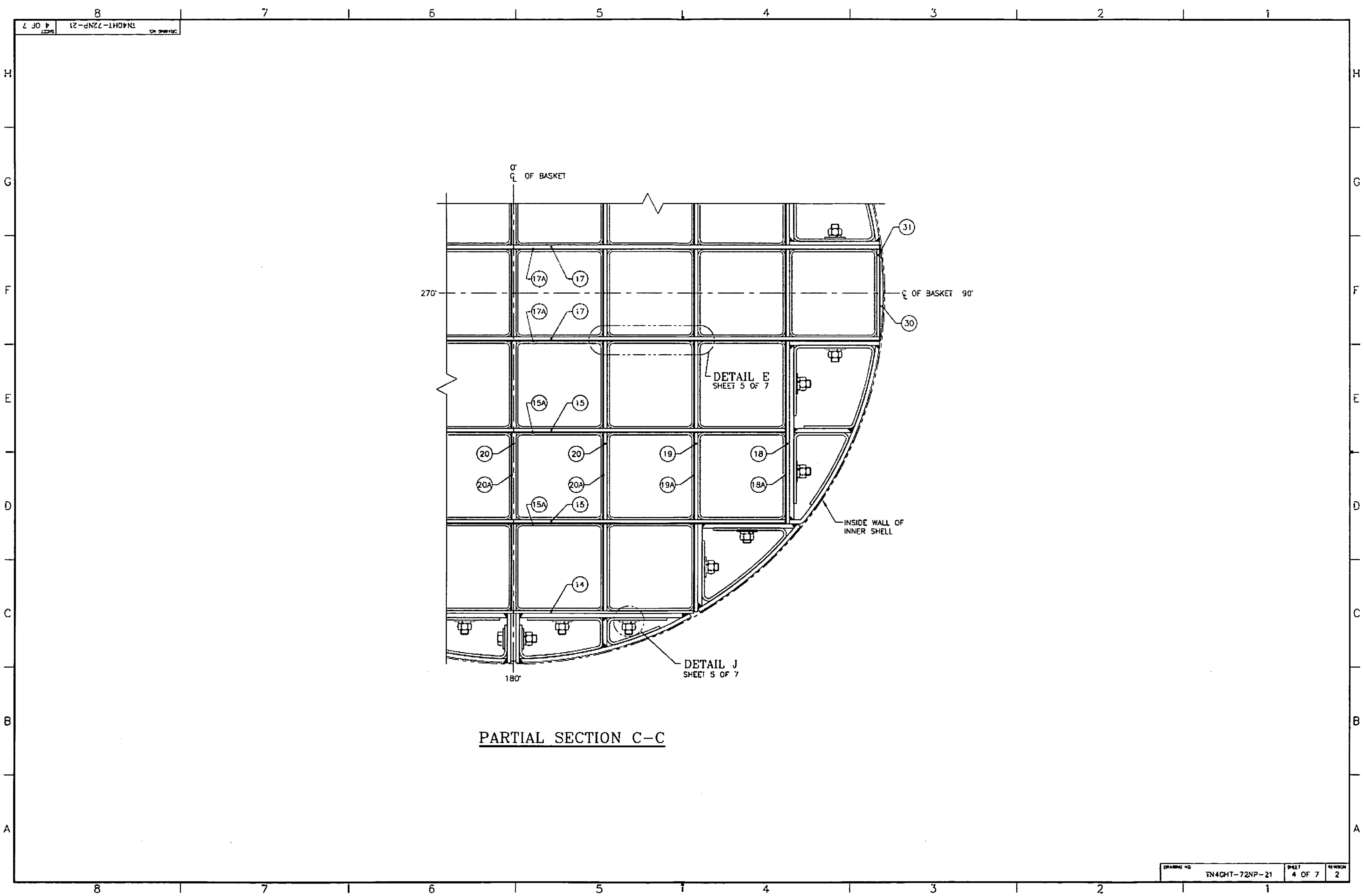
ITEM 1
TOP VIEW

WITHHELD PURSUANT TO 10 CFR 2.390

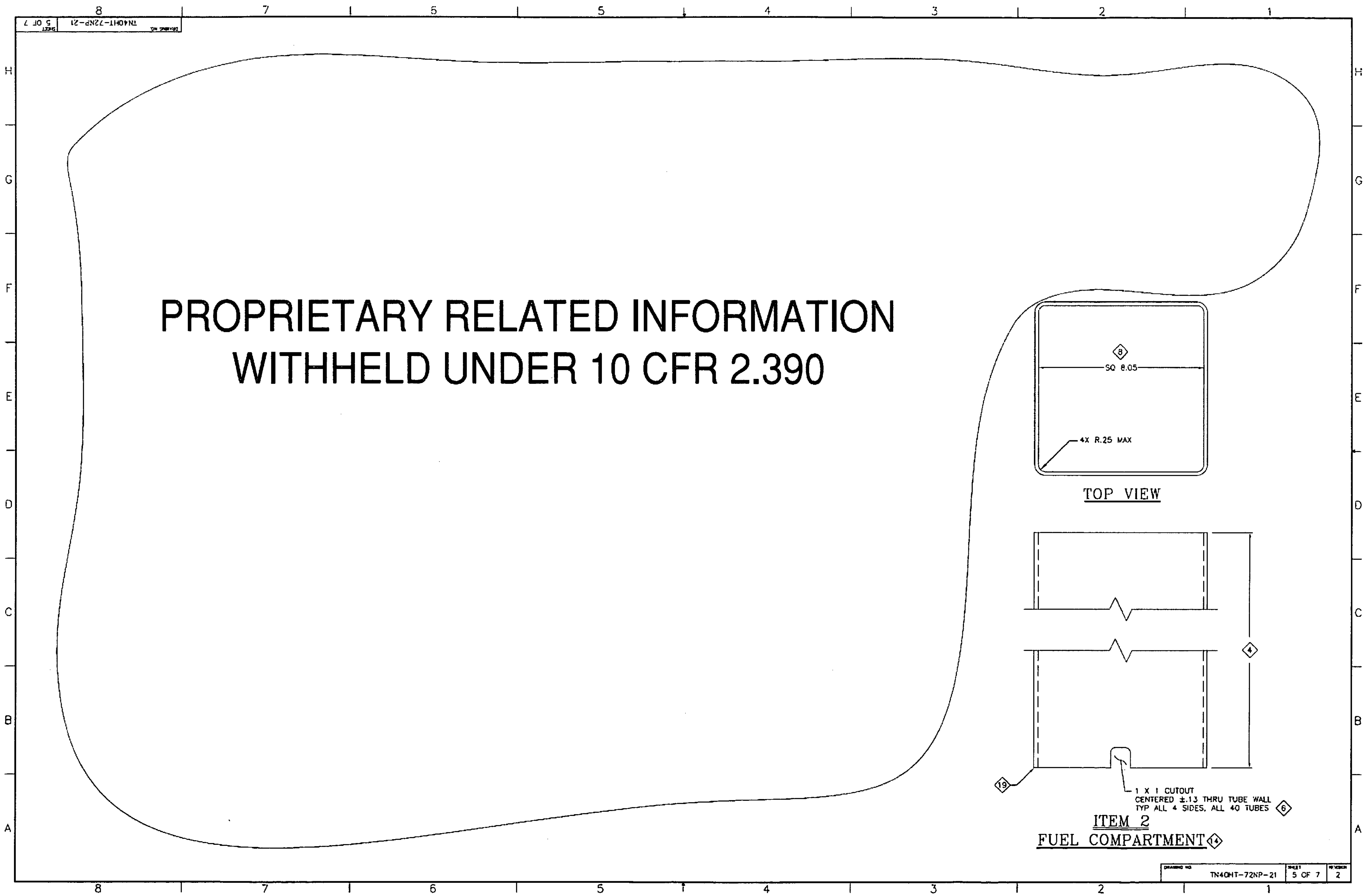


PARTIAL ISOMETRIC VIEW
TO SHOW PLATE ORIENTATION TYPICAL FOR ALL SECTIONS NOT TO SCALE
BASKET RAILS NOT SHOWN
SOME COMPONENTS REMOVED FOR CLARITY

BASKET ELEVATION
VIEW FROM 270°
SOME COMPONENTS (RAILS, SHIMS) REMOVED FOR CLARITY



PARTIAL SECTION C–C


PROPRIETARY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390
SQ 8.05
4X R.25 MAX
TOP VIEW
1 X 1 CUTOUT
CENTERED ±.13 THRU TUBE WALL
TYP ALL 4 SIDES, ALL 40 TUBES
ITEM 2
FUEL COMPARTMENT
TN40HT-72NP-21
5 OF 7
2

PROPRIETARY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
TN40HT-72NP-21	6 OF 7	2

PROPRIETARY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

TN40HT-72NP-21 7 OF 7 2

Assemblies containing fuel inserts (TPAs and BPRAs) may be stored in the TN-40HT cask. Reconstituted assemblies, (natural uranium dioxide replacement rods, Zirconium inert rods, or stainless steel rods replacing fuel rods), may also be stored in the cask. The decay heat of a reconstituted assembly with stainless steel rods is bounded by an intact assembly. However, the irradiated stainless steel rods do increase the gamma source term for a period of time after irradiation. This period is shorter than the 12 year minimum cooling time required and thus no additional cooling time is required for these reconstituted assemblies.

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.

A3.1.2 GENERAL OPERATING FUNCTIONS

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

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

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

Boral™	Thermal conductivity of Core		Thermal conductivity of Poison Plate
Temperature (°F)	(W/cm-K) (Reference 33)	(Btu/hr-in-°F)	(Btu/hr-in-°F)
100	0.859	4.14	4.14
500	0.768	3.70	3.70

Based on data from Reference 33, the average density for Boral™ plate is calculated as follows:

$D_a = 2.713(g/cm^3) \times t_a(cm)$; t_a = cladding thickness

$D_c = 2.481(g/cm^3) \times t_c(cm)$; t_c = core thickness

$D_t = (2 \times D_a + D_c)$

Poison plate density = D_t / t_t; t_t= Boral™ plate thickness

For a core thickness of 0.1" for a 0.125" thick Boral™ plate, the plate density is 0.091 lbm/in^3. A value of 0.0896 lbm/in^3 is conservatively used for poison plates in the analysis.

Specific heat for Boral™ plate can be calculated as follows:

$$Cp_t = \frac{2Cp_a \times D_a + Cp_c \times D_c}{D_t}$$ (Reference 33)

Boral™	Specific heat of Aluminum Cladding (Cp_a)		Specific heat of Core (Cp_c)		Specific heat of Poison Plate (Cp_t)
Temperature (°F)	(kJ/kg-K)	(Btu/lbm-°F)	(kJ/kg-K)	(Btu/lbm-°F)	(Btu/lbm-°F)
100	0.919	0.22	0.936	0.22	0.22
500	1.12	0.27	1.38	0.33	0.32*

*0.33 Btu/lbm-°F is used in the model. Due to small differences between these values the thermal evaluation remains unaffected.

A3.3.2.2.3.2 DETERMINATION OF HELIUM THERMAL PROPERTIES

The thermal properties for helium are calculated based on the following polynomial function from Reference 27.

$k = \sum C_i T_i$ for conductivity in (W/m-K) and T in (K)

For 300 < T < 500 K		for 500< T < 1050 K	
c0	-7.761491E-03	c0	-9.0656E-02
c1	8.66192033E-04	c1	9.37593087E-04
c2	-1.5559338E-06	c2	-9.13347535E-07
c3	1.40150565E-09	c3	5.55037072E-10
c4	0.0E+00	c4	-1.26457196E-13

A3.3.2.2.3.3 DETERMINATION OF AIR THERMAL PROPERTIES

The thermal properties for air are calculated based on the following polynomial function from Reference 27.

$$k = \sum C_i \, T_i \quad \text{for conductivity in (W/m-K) and T in (K)}$$

For 250 < T < 1050 K	
c0	-2.2765010E-03
c1	1.2598485E-04
c2	-1.4815235E-07
c3	1.7355064E-10
c4	-1.0666570E-13
c5	2.4766304E-17

A3.3.2.2.3.4 DETERMINATION OF CONCRETE AND SOIL THERMAL PROPERTIES

The thermal conductivity of normal, saturated concrete varies from 1.2 to 2.0 Btu/ft-hr-°F at temperatures ranging from 50 to 150 °F (Reference 30). The conductivity of concrete decreases rapidly with the rise in temperature and assumes, at 750 °C (1382 °F) a conductivity value equal approximately to 50 percent of that of normal temperature (Reference 30). For the thermal analyses a thermal conductivity of 1.15 Btu/hr-ft-°F (0.0958 Btu/hr-in-°F) is used for concrete at 70 °F. This conductivity is reduced by half to a value of 0.0479 Btu/hr-in-°F at 1382 °F. A thermal conductivity of 0.3 W/m-K (0.0144 Btu/hr-in-°F) is considered for soil (Reference 31).

Since the concrete pad is not included in the transient runs such as the fire accident and the vacuum drying cases, the density and specific heat of concrete and soil are not provided.

A3.3.2.2.3.5 EMISSIVITIES AND ABSORPTIVITIES

All outer surfaces of TN-40HT cask and all inner and outer surfaces of the protective cover are painted white. Reference 32 gives an emissivity between 0.92 and 0.96, and a solar absorptivity between 0.09 and 0.23 for white paints. To account for dust and dirt and to bound the problem, the thermal analysis uses an emissivity of 0.9 and a solar absorptivity of 0.3 for white painted surfaces.

Emissivity of concrete is between 0.9 and 0.94 (References 31 and 32). An emissivity of 0.90 is used for concrete surfaces. For conservatism a solar absorptivity of 1.0 is used for concrete surface to bound the effect of insolation.

The emissivity of the cask outer surface is set to 0.8 to be consistent with the requirements in Reference 38 during the accident fire. It is assumed that the cask surface is covered with soot after the fire. The solar absorptivity of soot is 0.95 (Reference 32). To bound the problem, the thermal analysis uses a solar absorptivity of 1.0 and an emissivity of 0.9 for cask surfaces after the fire.

Temperature (°C)	c_p (cal/g-°C)	Temperature (°F)	c_p (Btu/lbm-°F)
0	0.056	32	0.056
100	0.063	212	0.063
200	0.0675	392	0.068
400	0.0722	752	0.072
1200	0.079	2192	0.079

The density of fuel pellets (UO_2) is 10.96 g/cc = 0.396 lbm/in^3.

The thermal conductivities shown above represent values for un-irradiated UO_2 pellets. A study performed by Transnuclear (TN) and provided to the NRC in Reference 37 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 a temperature of 700°F.

The sensitivity runs in the TN study showed that the fuel cladding temperature changes by approximately 1°F when using irradiated UO_2 conductivity. Since a cladding temperature change of 1°F is negligible, the results of the study show that the fuel cladding temperature is not sensitive to the conductivity of UO_2. Therefore, use of un-irradiated UO_2 fuel pellet conductivity from NUREG/CR-0200 (Reference 14) is reasonable for irradiated UO_2.

A3.3.2.2.3.6.2.2 FUEL CLADDING, ZIRCALOY-4 / ZIRLO

Table B-2.I of Reference 41 lists measured and calculated values of thermal conductivity for zircaloy at various temperatures. The measured values used in this calculation are listed below.

Temperature (K)	k (W/m-K)	Temperature (°F)	k (Btu/hr-in-°F)
373.2	13.6	212	0.655
473.2	14.3	392	0.689
573.2	15.2	572	0.732
673.2	16.4	752	0.790
773.2	18.0	932	0.867
873.2	20.1	1112	0.968

Table B-1.1 of Reference 41 lists specific heat values for zircaloy as a function of temperature.

Temperature (K)	c_p (J/kg-K)	Temperature (°F)	c_p (Btu/lbm-°F)
300	281	80	0.067
400	302	260	0.072
640	331	692	0.079
1090	375	1502	0.090

The density of zircaloy is 6.56 g/cm^3 = 0.237 lbm/in^3, as defined in Reference 41.

Table B-3.11 of Reference 41 lists the measured emissivity values for fuel cladding. For ease of calculation a temperature independent emissivity of 0.8 is set for Zircaloy-4 in this calculation.

Note that Reference 42 states that ZIRLO and Zircaloy-4 alloys are very similar in terms of their thermal characteristics. Therefore zircaloy properties above are adequate for modeling ZIRLO.

A3.3.2.2.3.6.2.3 EMISSIVITY OF FUEL COMPARTMENTS, STAINLESS STEEL PLATES

An emissivity of 0.3 for the stainless steel plates is used for the fuel compartments in calculating the transverse effective fuel conductivity. This value is conservative relative to the values provided in Reference 6.

A3.3.2.2.3.6.3 EFFECTIVE FUEL CONDUCTIVITY

A3.3.2.2.3.6.3.1 TRANSVERSE EFFECTIVE CONDUCTIVITY

The purpose of the effective conductivity in the transverse direction of a fuel assembly is to relate the temperature drop of a homogeneous heat generating square to the temperature drop across an actual assembly cross section for a given heat load. This relationship is established by the following equation obtained from Reference 43:

$$k_{eff} = \frac{q}{4L_a(T_c - T_o)}(0.29468) = \frac{q_{react}}{(T_c - T_o)}(0.29468)$$

Where:

k_{eff} = Effective thermal conductivity (Btu/hr-in-°F)

q = Assembly heat generation (Btu/hr)

q_{react} = Reaction solution retrieved from the quarter-symmetric 2D model (Btu/hr)

$q = 4 \times q_{react} \times L_a$

L_a = Assembly active length (in)

T_c = Maximum temperature (°F)

T_o = Compartment Wall temperature (°F)

A two dimensional, quarter-symmetric finite element model of a 14x14 fuel assembly is developed using the ANSYS computer code (Reference 36). All components are modeled using 2D PLANE55 thermal solid elements. This two-dimensional model simulates the heat transfer by radiation and conduction. No convection is considered within the fuel assembly for conservatism. Radiation between the fuel rods and the compartment walls is simulated using the /AUX12 processor in ANSYS. For this purpose, LINK32 elements are placed on the exteriors of the fuel rods for creation of the radiation super-element. All LINK32 elements are unselected prior to solution of the thermal problem.

The geometry of this model is shown in Figure A3.3-17.

A fuel assembly decay heat load of 0.80 kW is used for heat generation. An active length of 144" is assumed for the model. Several computational runs were made for the model using isothermal boundary temperatures ranging from 100 to 1000 °F. The isothermal boundary conditions are applied on the outermost nodes of the model, which represent the compartment walls. In determining the temperature dependent effective conductivities of the fuel assembly an average temperature, equal to $(T_o + T_c)/2$, is used for the fuel temperature. A typical temperature distribution for the fuel assembly model is shown in Figure A3.3-18.

The transverse effective conductivity is calculated with helium as backfill gas for all conditions except for vacuum drying. For vacuum drying conditions, the conductivity of helium is replaced by conductivity of air.

A3.3.2.2.3.6.3.2 AXIAL EFFECTIVE CONDUCTIVITY

The backfill gas, fuel pellets, and fuel cladding behave like resistors in parallel. However, due to the small conductivity of the fill gas and the axial gaps between fuel pellets, credit is only taken for the Zircaloy-4 in fuel cladding in the determination of the axial effective conductivities.

$$k_{eff,axl} = \frac{cladding\ area}{4a^2} \times k_{Zr} \qquad \text{(Btu/hr-in-°F)}$$

a = half of compartment width = 8.05"/2 = 4.025"

A3.3.2.2.3.6.4 EFFECTIVE FUEL DENSITY AND SPECIFIC HEAT

Volume average density and weight average specific heat are calculated to determine the effective density and specific heat for the fuel assembly. The equations to determine the effective density and specific heat are shown below.

$$\rho_{eff} = \frac{\sum \rho_i V_i}{V_{assembly}} = \frac{\rho_{UO_2} \; V_{UO_2} + \rho_{Zr} \; V_{Zr}}{4a^2 L_a}$$

$$C_{P,eff} = \frac{\sum m_i C_{Pi}}{\sum m_i} = \frac{M_{UO_2} C_{P,UO_2} + M_{Zr} C_{P,Zr}}{M_{UO_2} + M_{Zr}}$$

A3.3.2.2.3.6.5 CONCLUSION

The transverse effective conductivity values for fuel assemblies in helium are listed in Table A3.3-9. The effective transverse conductivity used for fuel assemblies in thermal analyses for normal/off-normal (when in helium) and accident conditions is lower than the value calculated in this section. The applied values for transverse fuel conductivity are shown in Table A3.3-8 and are compared to the calculated value in Figure A3.3-19. Note that the transverse effective conductivity values for fuel assemblies in air are also shown on Table A3.3-8 as are other calculated effective properties for a homogenized fuel assembly.

A3.3.2.2.4 HEAT TRANSFER COEFFICIENTS

A3.3.2.2.4.1 TOTAL HEAT TRANSFER COEFFICIENT TO AMBIENT (NORMAL/OFF-NORMAL)

The outer surfaces of the cask dissipate heat to the ambient via free convection and radiation. Total heat transfer coefficient is defined as:

$$H_t = h_r + h_c$$

Where,

h_r = radiation heat transfer coefficient

h_c = free convection heat transfer coefficient

The radiation heat transfer coefficient, h_r, is given by the equation:

$$h_r = \varepsilon F_{12} \left[\frac{\sigma \left(T_1^{\,2} - T_2^{\,2}\right)}{T_1 - T_2} \right] \quad \text{(Btu/hr-ft}^2\text{-°F)}$$

Where,

ε = surface emissivity

F_{12} = view factor from cask surface to ambient

$\sigma = 0.1714 \times 10^{-8}$ Btu/hr-ft^2-°R^4

T_1 = cask surface temperature, °R

T_2 = ambient temperature, °R

Since the TN-40HT casks can be modeled as being stored in a $2 \times \infty$ array with a nominal pitch of 18 ft. as shown in Figure A3.3-20, the radiation view factor from the radial cask surfaces to the environment is less than 1.0. A view factor of 0.8138 is calculated for in Section 3.3.2.2.1 between the TN-40 cask and the ambient for this same configuration. This value is multiplied by surface emissivity to calculate radiation heat transfer from cask radial surfaces.

Cask gray body exchange factor $= F_{12} \times \varepsilon = 0.8138 \times 0.9 = 0.7324$

For conservatism, a value of 0.72 is used in the determination of the total heat transfer coefficient that is applied in the detailed TN-40HT cask model described in Section A3.3.2.2.1.1 for calculating of radiation heat transfer.

A3.3.2.2.4.1.1 STORAGE ARRAY RADIANT HEAT TRANSFER EVALUATION

The view factor between the cask surface and the concrete pad and service road surrounding the concrete pad were not considered explicitly in the calculation of the view factor for the TN-40 in Section 3.3.2.2.1.

The surface temperature of the concrete pad and service roads maybe higher than ambient temperature due to solar radiation and thermal radiation from the casks, but it is significantly lower than the cask surface temperature. Therefore, the radiation exchange between the casks and the concrete pad/service road is significant, but it maybe lower than the radiation exchange between the casks and the ambient.

A three dimensional model of TN-40HT casks in a storage array is developed using ANSYS (Reference 36) to investigate the effects of the cask view factor on the thermal performance. The model includes 18 casks and considers a cask pitch of 18 ft. The storage pad width is extended three times the cask pitch (72 ft.) to minimize any uncertainty regarding the radiation exchange between the casks and the surroundings. This model is shown in Figure A3.3-21.

Effective conductivities for cask shells and cask body in axial and radial directions are calculated using the detailed model of TN-40HT cask described in Section A3.3.2.2.1.1.1. The TN-40HT cask is divided into five sections as shown in Figure A3.3-22 to calculate the effective conductivities. The methodologies and results for effective conductivities of cask sections are described in below. These effective conductivities are used in the storage array model to reduce the number of elements and create manageable input and output files.

The dimensions of the TN-40HT cask in the storage array model are identical to those used in Section A3.3.2.2.1.1.1. The decay heat load is applied as a uniform heat flux on the cask inner surface over the length of basket (160 in.). This heat flux is calculated as follows.

Decay heat = 32 kW = 109,193.6 Btu/hr

ID_{cask} = 72"

$Height_{basket}$ = 160"

Decay heat flux = Decay heat /(π ID_{cask} × $Height_{basket}$) = 3.017 Btu/hr-in^2

The solar heat flux on the cask outer surface and the fixed temperatures at 10 ft. below the cask bottom plate are identical to those described in Section A3.3.2.2.1.1.3. Free convection boundary conditions are considered at the cask outer surface using correlations described in Section A3.3.2.2.4.3. Radiation exchange between cask and surroundings is simulated using radiosity methodology in ANSYS (Reference 36). An ambient temperature of 100 °F is considered for this model.

The resultant temperature distributions are shown in Figure A3.3-23 for cask outer surfaces, in Figure A3.3-24 for the storage pad surface temperature, and in Figure A3.3-25 for the cask inner shell.

As seen in Figure A3.3-21, the storage array model considers complete casks without symmetry planes. The detail cask model described in Section A3.3.2.2.1.1.1 considers a 90 degree, quarter symmetric segment of the cask. In order to compare the temperature profiles of these two models, average temperatures at the hottest cross sections are retrieved form each model.

The average and maximum temperatures for storage array model and TN-40HT detailed cask model are compared in Table A3.3-10.

As seen in Table A3.3-10, the average temperatures for the cask inner shell in the array storage and the cask detail models are within 1 °F of each other. Therefore, the temperatures for the basket and its contents would not be affected significantly if the view factor from the cask to surrounding storage pads were included in the thermal model.

The average cask surface temperature in the detailed cask model is 242 °F while the average cask surface temperature in the storage array model is 260 °F. This shows a temperature increase of 18 °F due to the cask view factors.

The calculated average resin temperature at the hottest cross section of the detailed model is 267 °F. The addition of the storage pad to the cask view factor calculation increases the average resin temperature at the hottest cross section by at most 18 °F to 285 °F. This temperature remains below the allowable limit of 300 °F for the radial resin.

Since the basket temperatures calculated in the TN-40HT detailed model remain unaffected, these temperatures can be used for calculation of the thermal stress, thermal expansion, and cask cavity pressure for the structural evaluation.

The view factors from the emitting cask (shown in Figure A3.3-20) to other casks and to the storage pad are calculated from the storage array model. These view factors are compared in Table A3.3-11 to the values calculated in Section 3.3.2.2.1 for information purposes only.

Effective Conductivity for Cask Sections

To calculate the axial and the radial effective conductivities of the cask sections shown in Figure A3.3-22, corresponding nodes and elements for each section are selected from the detailed finite element model of the TN-40HT cask.

To calculate the axial effective conductivity of each cask section, constant temperature boundary conditions are applied at the top and bottom of that section. The value of reaction solution for the colder surface is retrieved from the model using PRNDL command (Reference 36) after solution was completed. Since the model is quarter symmetric, the amount of heat leaving the colder surface is four times the reaction solution resulted from PRNLD command. The axial effective conductivity is calculated using the following equation:

$$k_{eff,axl} = \frac{Q \times L}{A \times \Delta T}$$

Q = Amount of heat leaving the colder face of the cask section = Qreaction × 4

L = Length of the cask section

A = Surface area of the colder face (upper face) of the cask section

= $\pi\,(r_2^2 - r_1^2)$

r_1 = Cask cavity radius = 36" for cask sections 1 through 4, 0 for cask section 5

r_2 = Outer radius of the cask section = 44.75" for cask sections 1, 2, and 4

= 50.5" for cask section 3

= 36" for cask section 5

$\Delta T = (T_2 - T_1)$ = Temperature difference between lower and upper faces of the section (°F)

T_1 = Constant temperature applied on the upper face of the model (°F)

T_2 = Constant temperature applied on the lower face of the model (°F)

To calculate the radial effective conductivity of the cask sections, constant temperature boundary conditions are applied on the outermost and innermost nodes of the sections. The value of reaction solution on the colder surface is then retrieved from the model using PRNDL command and multiplied by four to give the amount of heat leaving the colder surface. The radial effective conductivity is calculated using the following equation with the same terms defined for axial effective conductivity:

$$k_{eff,rad} = \frac{Q \times \ln(r_2/r_1)}{2\pi\, L\, \Delta T}$$

For Section 5, the lowest conductivity between the cask lid and cask top shield (SA203, Gr. E, Table 3.3-8) is used for radial conductivity.

In determining the temperature dependent effective conductivities an average temperature, equal to $(T_2 + T_1)/2$, is used. The resulting effective conductivities are listed in Table 3.3-12.

A3.3.2.2.4.2 TOTAL HEAT TRANSFER COEFFICIENT TO AMBIENT FOR FIRE

The radiation and forced convection from the fire toward the cask surface are combined together as a total heat transfer coefficient. Total heat transfer coefficient is defined as:

$$H_{t,fire} = h_{r,fire} + h_f$$

Where,

$h_{r,fire}$ = radiation heat transfer coefficient from fire

h_f = forced convection heat transfer coefficient

A forced convection value of 4.5 Btu/hr-ft^2-°F (0.03125 Btu/hr-in^2-°F) is considered during the burning time from Reference 39.

The radiation heat transfer coefficient during the burning period of the hypothetical fire accident, $h_{r,fire}$, is given by the following equation:

$$h_r = \varepsilon_s\, F_{sf} \left[\frac{\sigma(\varepsilon_f\, T_f^{\,4} - T_s^{\,4})}{(T_f - T_s)} \right] \quad \text{(Btu/hr-ft}^2\text{-°F)}$$

Where,

ε_s= surface emissivity = 0.8 (Reference 38)

ε_f = fire emissivity = 0.9 (Reference 38)

F_{sf} = view factor from surface to fire = 1

$\sigma = 0.1714 \times 10^{-8}$ Btu/hr-ft^2-°R^4

T_f = fire flame temperature, 1475°F = 1935°R (Reference 38)

T_s = cask surface temperature, °R

A3.3.2.2.4.3 FREE CONVECTION COEFFICIENTS

The free convection coefficients are calculated based on the shape and position of the convective surface using correlations from Reference 27. The convection correlations are described in the following sections.

A3.3.2.2.4.3.1 VERTICAL CYLINDER

The following equations from Reference 27 are used to calculate the free convection coefficients for vertical cylindrical surfaces.

$$h_c = \frac{Nu\, k}{L}$$

With L = height of the vertical cylinder

D= diameter of vertical cylinder

k = air conductivity

$$Ra = Gr\,\Pr \quad ; \quad Gr = \frac{g\,\beta\,(T_w - T_\infty)L^3}{\nu^2}$$

$Nu^T_{Plate} = \overline{C}_l\, Ra^{1/4}$, $\quad \overline{C}_l = 0.515$ for gases (Reference 27)

$$Nu_{l,Plate} = \frac{2.0}{\ln(1 + 2/Nu^T_{Plate})}$$

$$\xi = \frac{1.8\,L/D}{Nu^T_{Plate}}$$

$Nu_l = \dfrac{\xi}{\ln(1+\xi)} Nu^T_{Plate}$ Nusselt number for fully laminar heat transfer

$$C^V_t = \frac{0.13\Pr^{0.22}}{(1 + 0.61\Pr^{0.81})^{0.42}}$$

$$f = 1 + 0.078\left(\frac{T_w}{T_\infty} - 1\right)$$

$Nu_t = C^v_T\, f\, Ra^{1/3} / \left(1 + 1.4\times10^9 \Pr/Ra\right)$ Nusselt number for fully turbulent heat transfer

$Nu = \left[(Nu_l)^m + (Nu_t)^m\right]^{1/3}$ with m=6

The correlations to calculate the total heat transfer coefficient are incorporated in the ANSYS model via a macro. Air properties are taken from Reference 27 and listed in Table A3.3-8.

A3.3.2.2.4.3.2 HORIZONTAL FLAT SURFACES FACING DOWNWARDS

The following equations from Reference 27 are used to calculate the free convection coefficients for horizontal flat surfaces facing downwards.

$$h_c = \frac{Nu\, k}{L}$$

With L = A/P

A=surface area of heated surface

P= perimeter of the heated surface

k = air conductivity

$$Ra = Gr\,\mathrm{Pr} \quad ; \quad Gr = \frac{g\,\beta\,(T_w - T_\infty)L^3}{\nu^2}$$

$$Nu_l = \frac{0.527\,Ra^{1/5}}{[1 + (1.9/\mathrm{Pr})^{9/10})^{2/9}}$$

$$Nu = Nu_l$$

The above correlations are incorporated in ANSYS model via a macro. Air properties are taken from Reference 27 and listed in Table 3.3-8.

A3.3.2.2.4.3.3 HORIZONTAL FLAT PLATE FACING UPWARDS

The following equations from Reference 27 are used to calculate the free convection coefficients for horizontal flat surfaces facing upwards.

$$h_c = \frac{Nu\, k}{L}$$

With L = A/P

A=surface area of heated surface

P= perimeter of the heated surface

k = air conductivity

$$Ra = Gr\,\mathrm{Pr} \quad ; \quad Gr = \frac{g\,\beta\,(T_w - T_\infty)L^3}{\nu^2}$$

$$Nu^T = 0.835\,\overline{C}_l\,Ra^{1/4}$$

$\overline{C}_l = 0.515$ for gases (Reference 27)

$Nu_l = \dfrac{1.4}{\ln(1 + 1.4/Nu^T)}$ Nusselt number for fully laminar heat transfer

$C_t^H \approx 0.14$ for Pr < 100 (Reference 27)

$Nu_t = C_t^H \, Ra^{1/3}$ Nusselt number for fully turbulent heat transfer

$$Nu = \left[(Nu_l)^m + (Nu_t)^m\right]^{1/m} \qquad with \quad m = 10 \quad for \quad Ra > 1$$

The above correlations are incorporated in ANSYS model via a macro. Air properties are taken from Reference 27 and listed in Table 3.3-8.

A3.3.2.2.5 THERMAL EVLUATION OF LOADING AND UNLOADING

Fuel loading and unloading operations occur in the fuel handling building. During loading operation fuel assemblies are submerged in pool water permitting heat dissipation. After fuel loading is complete, the cask is removed from the pool, drained, sealed, dried, and backfilled with helium.

A3.3.2.2.5.1 VACUUM DRYING

The vacuum drying operation evaluated is the heatup of the cask before helium is introduced into the cask cavity. The time for this operation ($t_{loading}$) is defined as the interval from the start of water being drained out of the cask cavity to the beginning of helium backfilling.

The duration of vacuum drying operation being evaluated is limited by the amount of time required to reach the maximum fuel cladding temperature of 752 °F (400 °C) allowed by ISG-11 (Reference 25).

To determine the time-temperature histories of the cask components, a transient analysis is performed using the cross-section model of the cask described in Section A3.3.2.2.1.2. Although cask cavity is full of water at $t_{loading}=0$, the following changes are considered to occur immediately at the start of the transient run and remain unchanged.

- Effective conductivity values in a vacuum are considered for elements representing homogenized fuel assemblies.
- Air conductivity is given to the elements representing the gas within the cask cavity.

The conductivity of air or helium is independent of the pressure for the conditions considered during the loading and vacuum drying operations. All other material properties of the cask cross-section model remain unchanged. The effective fuel conductivity values used for vacuum conditions are listed in Table A3.3-8.

The decay heat is applied as heat generation load on the elements representing the homogenized fuel assemblies with a peaking factor of 1.1. The applied value for the heat generation is:

Heat generation rate = $\dot{q}'' = \dfrac{q}{a^2 \, L_a} \times PF = 0.3218$ Btu/hr-in^3

where,

q = Decay heat load per assembly = 2,730 Btu/hr (0.80 kW)
a = Width of the modeled fuel assembly = 8.05 in.
L_a =Active fuel length = 144 in.
PF = maximum peaking factor = 1.1 (Reference 26)

Adiabatic boundary conditions are applied on the top and bottom faces of the cross-section model for conservatism.

An average, initial temperature of 215 °F is assumed for the cask at the start of water draining. This temperature is higher than boiling point of water.

It is assumed that the cask resides in the pool for 2 hours after the water draining starts. Maximum pool temperature is 150 °F. Conservatively, a constant temperature of 215 °F is applied on the cask surface during this period.

After leaving the pool, the cask dissipates heat to ambient inside of the fuel handling building. The free convection and radiation are combined together to calculate the total heat transfer coefficient from the cask outer surface to the ambient. An ambient temperature of 120 °F is considered conservatively for the fuel handing building.

The volumetric average temperatures of the basket, rails, inner shells, basket support bars and basket aluminum plates are retrieved from the thermal model to calculate the thermal expansion. A zero hot gap at thermal equilibrium is assumed between the rails and the cask inner shell in retrieving the average temperatures. These temperatures are listed in Table A3.3-13.

The time-temperature history of the maximum fuel cladding temperature is illustrated in Figure A3.3-26 for loading operations with 32 kW decay heat load. The time period for this operation was defined as the time from the beginning of cask draining to when helium backfilling begins. A time limit of 34 hours was chosen for this operation to ensure that the fuel cladding temperature limit of 752 °F was not exceeded.

The temperature distribution for cask cross-section model at $t_{loading}$ = 34 hours is shown in Figure A3.3-27.

The temperature limits for fuel cladding, radial neutron shielding material, and seals are not exceeded at 34 hours as shown in Table A3.3-14.

Once helium is introduced, subsequent evacuation and backfilling with helium will not change the conductivity of the cavity gas, and therefore no repeated cycling of the fuel cladding temperature occurs.

A3.3.2.2.5.2 REFLOODING

For unloading operations, the cask will be slowly filled with borated water to gradually cool the fuel in the cask.

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.

A3.3.2.2.6 INTERNAL CASK PRESSURE DETERMINATION

A3.3.2.2.6.1 MAXIMUM INTERNAL PRESSURE

The following methodology is used to determine the maximum pressures within the TN-40HT cask cavity for normal, off-normal, and accident conditions:

- Average cavity gas temperatures are derived from the TN-40HT cask thermal models.
- The amount of helium present within the cask cavity after the initial backfilling is determined via the ideal gas law.
- The total amount of free gas within the fuel assemblies, including both fill and fission gases, are calculated based on NUREG 1536 (Reference 35) guidelines.
- The amount of released gas from the fuel rods into the cask cavity is determined based on the maximum fraction of the ruptured fuel rods considered in NUREG 1536 (Reference 35).
- The amount of helium backfill gas is added to the amount of released gases to make the total amount of gases in the cask cavity.
- Finally, the maximum cask internal pressures are determined via the ideal gas law.

To bound the maximum internal pressure, the maximum daily average temperature of 100 °F is considered for both normal and off-normal conditions.

As it is assumed in NUREG 1536 (Reference 35), the maximum fractions of the fuel rods (f_B) that can rupture and release their free gases to the cask cavity for normal, off-normal, and accident conditions are 1, 10, and 100%, respectively.

A3.3.2.2.6.1.1 AVERAGE GAS TEMPERATURE

To determine the average cavity gas temperature, volume average temperatures of the elements representing the helium gaps ($T_{avg,void}$) and the homogenized fuel assemblies ($T_{avg,fuel}$) are retrieved from the thermal models using the ETABLE commands in ANSYS (Reference 36). Although the average temperature of the homogenized fuel elements includes the fuel rods and the helium gas between them, this average temperature is conservatively used as the average gas temperature within the fuel compartments.

The volume of helium gaps in the model is (V_{void}) is retrieved from the model using ETABLE commands (Reference 36) and is equal to 28,272 in^3.

The approximate volume of the gas in the fuel compartments ($V_{gas,comp}$) is determined as follows (note this is only an approximation and not intended to represent the physical configuration of the fuel, e.g. actual fuel pin length, open guide tubes, and fuel assembly hardware).

$V_{gas,comp} = N_c \times$ (Volume of one fuel compartment – Volume of fuel rods in one assembly)
N_c = Number of fuel compartment in finite element model (1/4 of cask is modeled) = 10
Volume of one fuel compartment = W x W x H
Volume of fuel rods in one assembly = N x $\pi/4$ OD^2 x H

- W = compartment width = 8.05"
- H = compartment height = 160"
- N = number of fuel rods and tubes = 196
- OD = the smallest outer diameter of fuel rods and tubes = 0.40 in.

Volume of one fuel compartment = 10,368 in^3
Volume of fuel rods in one assembly = 3,941 in^3
$V_{gas,comp}$ = 64270 in^3

The average gas temperature in the cask cavity is calculated as follows.

$$\bar{T}_{Cavity} = \frac{T_{avg,fuel} \times V_{gas,comp} + T_{avg,void} \times V_{void}}{(V_{gas,comp} + V_{void})}$$

The results are summarized below.

Operating Condition	$\bar{T}$ Cavity (°F)
Normal (Off-Normal) Storage Conditions	456
Fire Accident	592
Buried Cask Accident, 75 hrs after occurrence	835
Buried Cask Accident, 95.75 hrs after occurrence	929

A3.3.2.2.6.1.2 AMOUNT OF INITIAL HELIUM BACKFILL

A maximum cavity pressure of 1.43 atm abs (21 psia) is considered for the initial backfill pressure of helium. The amount of the helium backfill gas at this moment is calculated based on the ideal gas law.

The cavity gas temperature for the cask located in the fuel handling building is calculated considering an ambient temperature of 70 °F, a view factor of 1.0, and no insolance. The full-length cask model described in Section A3.3.2.2.1.1.1 is used for this purpose with steady state conditions. The average cavity gas temperature is retrieved from the model using the methodology described in Section A3.3.2.2.6.1.1. The retrieved average cavity gas temperature when cask is in the handling building is 426 °F (886 °R).

For the maximum gas backfill, it is assumed that helium does not instantaneously reach 426 °F, but is conservatively assumed to be the average of the ambient (70 °F) and the steady state cavity temperature of 426 °F.

A displacement of 480 in^3 is used for each BPRA.

From the backfill pressure and the backfill gas temperature, the amount of helium backfill gas is calculated as follows.

n_{back} = (PV)/(RT) = 0.580 lb-moles without BPRAs

= 0.549 lb-moles with BPRAs

P = maximum initial backfill pressure = 1.43 atm = 21 psia

V = cask cavity free volume without BPRAs (loaded) = 362,440 in^3 = 209.7 ft^3

= cask cavity free volume with BPRAs = cavity volume – BPRA volume

= $(362{,}440 - 40*480)/12^3$ = 198.6 ft^3

T = initial backfill temperature = 0.5 (70+426) = 248 °F = 708 °R

R = universal gas constant = 10.730 psia-ft^3/lb-moles-°R (Reference 29)

A3.3.2.2.6.1.3 FREE GAS WITHIN FUEL ASSEMBLIES

As indicated in Section A7.2, Table A7.2-1, the maximum volume of free gas per assembly is 0.226 m^3 at standard temperature and pressure (0 °C and 760 mmHg). Total amount of free gases within the fuel assemblies is:

$$n_{fuel} = N \times \frac{P_{Std} \cdot V_{free}}{R \cdot T_{Std}} = 40 \times \frac{760 \times 0.226 \times 10^3}{62.361 \times 273.15} = 403.3 \text{ g-mole} = 0.889 \text{ lb-mole}$$

With

N = number of fuel assemblies in the cask = 40

P_{Std} = standard pressure = 760 mmHg

V_{free} = maximum volume of free gas per assembly = 0.226 m^3

T_{Std} = standard temperature = 0 °C = 273.15 K

R = universal gas constant = 62.361 (mmHg-lit/g-mole-K) (Reference 29)

A bounding amount of 2.0E-4 lb-mole is considered for free gas in each BPRA rod. The maximum amount of free gases from the BPRA rods is:

$n_{BPRA} = N \times n_{Rod} \times (2.0E-4\ lb-mole/rod) = 40 \times 16 \times 2.0E\text{-}4 = 0.128$ lb-mole

With

N = number of fuel assemblies in the cask = 40

n_{Rod} = maximum number of BPRA rods in one assembly = 16

Total amount of free gases within fuel assemblies is:

$n_{free} = n_{fuel} = 0.889$ lb-mole without BPRAs

$= n_{fuel} + n_{BPRA} = 1.017$ lb-mole with BPRAs

A3.3.2.2.6.1.4 TOTAL AMOUNT OF GASES WITHIN CASK CAVITY

The total amount of gases within the cask cavity is equal to the amount of the initial helium backfill gas plus any free gases that are released to the cask cavity from ruptured fuel and BPRA rods.

Total amount of gases in the cask cavity is:

$n_{total} = n_{back} + f_B\ (n_{free})$

n_{back} = total amount of backfill gas

n_{free} = total amount of free gases

f_B = fraction of the ruptured fuel rods from NUREG-1536 (Reference 35)

A3.3.2.2.6.1.5 MAXIMUM CASK INTERNAL PRESSURE

The maximum cask internal pressure (P_{cavity}) is determined via the ideal gas law:

$P_{cavity} = (n_{total}\, R\, \bar{T}_{cavity})/V$

V = cask cavity free volume without BPRAs = 209.7 ft^3

= cask cavity free volume with BPRAs = 198.6 ft^3

R = universal gas constant = 10.73 (psia-ft^3/lb-mol-°R)

The results are summarized in Table A3.3-15 and Table A3.3-16.

The internal cask cavity pressures are at or below the design limit of 100 psig.

A3.3.2.2.6.2 INTERNAL PRESSURE AT END OF SERVICE LIFE

A minimum helium backfill pressure of 19.5 psia was determined on the basis that a minimum of 1 atm pressure must exist on the coldest day at the end of life.

The full-length cask model was run with steady state conditions in the handling building to determine the average cavity gas temperature after completion of the helium backfilling. An ambient temperature of 70 °F is considered for this run. The average gas cavity temperature of 426 °F (886 °R) was retrieved from the model using the methodology described in Section A3.3.2.2.6.1.1.

The determination of the end of life cavity pressure was based on the average gas backfill temperature of 426°F (886°R) at the time of backfill and an average gas temperature of 216°F (676°R) after 25 years of storage an external ambient temperature of -40°F.

The initial pressure of 19.5psia assures that at the end of 25 years, on the coldest day (–40 °F ambient), the internal pressure of the cask is:

$$P_{cavity} = 19.5 \text{ psia} \times (676°R/886°R) = 14.9 \text{ psi}$$

Therefore, the internal pressure of the cask is above the 1 atm minimum.

A3.3.2.2.7 RADIAL HOT GAP BETWEEN THE BASKET RAILS AND THE CASK INNER SHELL

A nominal diametrical cold gap of 0.30 in. is considered between the basket and the cask cavity wall for the TN-40HT cask.

A radial, hot gap of 0.13" at thermal equilibrium is assumed in the ANSYS model for normal storage conditions. To verify this assumption, the hot dimensions of the cask inner diameter and basket outer diameter are calculated at thermal equilibrium as follows.

The outer diameter of the hot basket is:

$$OD_{B,hot} = OD_B + [L_{SS,B} \times \alpha_{SS} (T_{avg,B} - T_{ref}) + L_{Al} \times \alpha_{Al} (T_{avg,Al} - T_{ref})]$$

Where:

$OD_{B,hot}$ = Hot outer diameter of the basket

OD_B = Cold outer diameter of the basket = 72" – 0.30" = 71.70"

$L_{SS,B}$ = Length of basket at 90-270 direction = O_{DB} – 2×0.46" = 70.78"

L_{Al} = Length of aluminum shim = 2×0.46 = 0.92"

α_{SS} = Stainless steel axial coefficient of thermal expansion

= 9.66E-6 in/in-°F @ 479 °F (Reference 28)

α_{Al} = Aluminum coefficient of thermal expansion
= 13.43E-6 in/in-°F @ 358 °F (Reference 28)

$T_{avg,B}$ = Average basket temperature at the hottest cross section = 479 °F (Table A3.3-4)

$T_{avg,Al}$ = Average shim temperature at the hottest cross section = 358°F (Table A3.3-4)

T_{ref} = reference temperature for stainless steel and aluminum alloys = 70°F (Reference 28)

The inner diameter of the hot cask is:

$ID_{c,hot} = ID_c\,[1 + \alpha_{LS}\,(T_{avg,C} - T_{ref})]$

Where:

$ID_{C,hot}$ = Hot inner diameter of cask cavity

ID_C = Cold inner diameter cask cavity = 72"

α_{LS} = Coefficient of thermal expansion for low alloy steel
= 6.91E-6 in/in-°F @ 302 °F (Reference 28)

$T_{avg,C}$ = Average cask inner shell and gamma shield temperature at hottest x-section = 302 °F (Table A3.3-4)

T_{ref} = reference temperature for low alloy steel = 70°F (Reference 28)

The hot gap between the basket outer diameter and cask inner diameter is:

$G_{hot} = ID_{C,hot} - OD_{B,hot}$ =0.132" (diametrical)

Radial hot gap = 0.066"

The assumed radial hot gap of 0.13" is larger than the above calculated hot gap. This assumption is therefore conservative.

A3.3.2.2.8 HEAT GENERATION RATE AS A FUNCTION OF SPENT FUEL PARAMETERS

A3.3.2.2.8.1 INTRODUCTION

The spent fuel loading into the TN-40HT Cask is based on a uniform loading of 800 watts per fuel assembly. The objective of this section is to describe a mathematical function that determines the heat generation rate for high burnup fuel as a function of initial enrichment, assembly burnup and cooling time. This attachment also provides a sample fuel qualification table for heat generation rate, based on the mathematical function. This function applies only to 14x14 high burnup fuel assemblies.

A3.3.2.2.8.2 CALCULATIONAL METHODOLOGY AND INPUT MODELS

The SAS2H and the ORIGEN-S modules of SCALE4.4 (Reference 14) computer code, with the 44 Group ENDF-V cross section library, are used to determine the thermal source terms.

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A3.3.2.2.8.3 MATHEMATICAL FUNCTION TO DETERMINE HEAT GENERATION

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The functional form is expressed below with decay heat (DH) in watts as:

$$DH = F1 * \text{Exp}(\{G*[1-(12.0/X3)]\}*[(X3/X1)^{H}]*[(X2/X1)^{I}])$$

Where:

$$F1 = A + B*X1 + C*X2 + D*X1^2 + E*X1*X2 + F*X2^2$$

and,

F1	Intermediate Function, basically the Thermal source at 12 year cooling
X1	Assembly Burnup in GWd/MTU (45 to 60)
X2	Initial Enrichment in wt % U235 (3.4 – 5.0)
X3	Cooling Time in Years (12 min)

A	18.76
B	11.27

C	6.506
D	0.163
E	-1.826
F	6.617
G	-0.309
H	0.431
I	-0.374

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A3.3.2.2.8.4 DECAY HEAT FOR LOW BURNUP ASSEMBLIES

For assemblies with burnups less than 44 Gwd/Mtu, the SASH2 analyses shows that decay heat load is less than 800 watts with a minimum cooling time of 12 years.

A3.3.2.2.8.5 HEAT GENERATION FOR INSERTS

The decay heat load for a thimble plug device with a minimum of 16 years cooling was determined to be 0.42 watts. This value was based on a maximum cumulative host assembly(s) burnup of 125,000 Mwd/Mtu.

The decay heat load for a burnable poison rod assembly with a minimum of 18 years cooling was determined to be 0.90 watts. This value was based on a maximum cumulative host assembly(s) burnup of 30,000 Mwd/Mtu.

If the fuel assembly to be stored contains an insert, these heat loads are added to the fuel assembly heat load to determine a combined heat load for comparison to the 800 watt criterion.

A3.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

A3.3.3.1 EQUIPMENT

The design criteria for the TN-40HT casks are described in Section A3.4 and summarized in Table A3.4-1.

A3.3.3.2 INSTRUMENTATION

Due to the totally passive and inherently safe nature of the storage system, safety-related instrumentation is not necessary. Instrumentation to monitor cask pressure is furnished. Appropriate capabilities to check and recalibrate these monitors are also provided. The pressure monitoring system is further described in Section A3.3.2.1.

A3.3.4 NUCLEAR CRITICALITY SAFETY

A3.3.4.1 CONTROL METHODS FOR PREVENTION OF CRITICALITY

The design criterion for criticality is that an upper subcritical limit (USL) of 0.95 minus benchmarking bias and modeling bias will be maintained for all postulated arrangements of fuel within the cask. The fuel assemblies are assumed to stay within their basket compartment based on the cask and basket geometry.

The control methods used to prevent criticality are:

1. Incorporation of neutron absorbing material (boron) in the basket material.

2. Loading of the irradiated fuel assemblies in the fuel pool containing at least 2450 ppm boron.

3. Prevention of fresh water entering the loaded cask.

The basket has been designed to assure an ample margin of safety against criticality under the conditions of fresh fuel in a cask flooded with borated pool water. The method of criticality control is in keeping with the requirements of 10 CFR72.124.

The TN-40HT cask system's criticality safety is ensured by fixed neutron absorbers in the basket, soluble boron in the pool. The cask basket uses a Borated-Aluminum alloy, Aluminum/B4C metal matrix composite or Boral® as the fixed neutron poison material. The collective term B-Al refers to the Borated-Aluminum alloy and Aluminum/B4C metal matrix composite materials where the analysis uses a 90% credit for the neutron poison. A credit of 75% is taken for the presence of neutron poison for Boral® plates.

A single fixed poison loading is utilized in the TN-40HT basket. Table A3.3-17 lists the minimum B10 poison loading required for the various poison materials and the corresponding poison content modeled in the analyses.

The minimum soluble boron concentration in the pool credited in the analysis is 2450 ppm.

A3.3.4.1.1 DISCUSSION AND RESULTS

Figure A3.3-28 shows the cross section of the TN-40HT cask. The TN-40HT cask stainless steel basket consists of an "egg-crate" plate design. The fuel assemblies are housed in 40 stainless steel fuel compartment tubes. The basket structure, including the fuel compartment tubes, is held together with stainless steel insert bars (note that the input files in Appendix A3A refer to these bars as plates) and the poison and aluminum plates that form the "egg-crate" structure. The fuel compartment tube structure is connected to the perimeter rail assemblies forming the cylindrical outer geometry required to be housed within the cask. The poison/aluminum plates are located between the fuel compartment tubes.

The analysis presented herein is performed for a TN-40HT Cask during normal, off-normal and accident loading conditions. This analysis also bounds all conditions of storage, including loading and transfer. The cask consists of an inner steel shell, a steel gamma shield shell and a hydrogenous neutron shield.

Table A3.3-18 lists the fuel assemblies considered as authorized contents of the TN-40HT cask system. The criticality analysis begins by determining the most reactive assembly type for the Westinghouse 14x14 fuel assembly (WE 14x14) class identified in Table A3.3-18. Then the most reactive configuration for the basket (including rail configuration) and fuel assembly position is determined. Next, criticality calculations are performed using the maximum allowable initial fuel enrichment for the TN-40HT cask shown in Table A3.3-17. These calculations determine k_{eff} with the CSAS25 control module of SCALE-4.4 (Reference 14) including all uncertainties to assure criticality safety under all credible conditions.

The Control Components (CCs) are also authorized for storage in the TN-40HT casks. The authorized CCs are Burnable Poison Rod Assemblies (BPRAs) and Thimble Plug Devices (TPDs).

The results of the evaluation demonstrate that the maximum expected k_{eff}, including all applicable biases and uncertainties, is less than the Upper Subcritical Limit (USL) determined from a statistical analysis of benchmark criticality experiments. The statistical analysis procedure includes a confidence band with an administrative safety margin of 0.05.

A3.3.4.1.2 PACKAGE FUEL LOADING

The TN-40HT Cask is capable of transferring and storing a maximum 40 intact WE 14x14 class PWR fuel assemblies. The reactivity of a cask loaded with less than 40 PWR fuel assemblies is lower than that calculated here since the more absorbing borated water replaces the fuel in the empty locations. Reconstituted fuel assemblies, where the fuel pins are replaced by lower enriched fuel pins or non-fuel pins that displace the same amount of borated water, would also lower the reactivity of the cask.

Table A3.3-19 lists the fuel parameters for the PWR fuel assemblies considered in this evaluation. Reload fuel from other manufacturers with the same parameters are also considered as authorized contents.

For the WE 14x14 class assemblies, CCs are also included as authorized contents. The only change to the package fuel loading to evaluate the addition of these CCs is replacing the borated water in the water holes conservatively with $^{11}B_4C$. Since these CCs displace borated moderator in the assembly guide tubes, an evaluation is performed to determine the potential impact of storage of CCs that extend into the active fuel region on the system reactivity. For CCs such as BPRAs no credit is taken for the cladding and absorbers; rather the CCs are modeled as $^{11}B_4C$ in the entire tube of the respective design. Thus, the highly borated moderator in the tube is conservatively modeled as $^{11}B_4C$. The inclusion of more Boron-11 and carbon enhances neutron scattering causing the neutron population in the fuel assembly to be slightly increased which increases reactivity. Therefore, these calculations bound any CC design that is compatible with the WE 14x14 class assemblies. CCs that do not extend into the active fuel region of the assembly (e.g., TPD) do not have any effect on the reactivity of the system as evaluated because only the active fuel region is modeled in this evaluation with periodic boundary conditions making the model infinite in the axial direction. The dimensions of the fuel assemblies reported in Table A3.3-19 remain unchanged for the CC cases. The models that include CCs only differ in that the region inside the guide tubes are modeled as $^{11}B_4C$ instead of moderator.

A3.3.4.1.3 MODEL SPECIFICATION

The following subsections describe the physical models and materials of the TN-40HT cask used for input to the CSAS25 module of SCALE-4.4 (Reference 14) to perform the criticality evaluation. The reactivity of cask under storage conditions is bounded by the analysis with zero (or near zero) internal moderator density case during loading and transfer.

A3.3.4.1.3.1 DESCRIPTION OF CALCULATIONAL MODEL

The TN-40HT cask is explicitly modeled using the appropriate geometry options in KENO V.a of the CSAS25 module in SCALE-4.4. Several models are developed to evaluate the fabrication tolerances of the basket/cask, fuel assembly locations, fuel assembly design and storage of fuel with CCs like BPRAs and TPDs.

The criticality evaluation is performed using an "egg-crate" section length of 14.49 inches in the basket. The actual "egg-crate" length is 15.0 inches in the active fuel region of the assembly. This represents more reactive design than the actual basket because of the shorter "egg-crate" section length. Utilizing a shorter section length in the calculational model ensures that the model is conservative since the amount of poison per unit length is minimized. The key basket dimensions utilized in the calculation are shown in Table A3.3-20.

The fixed poison modeled in the calculation is based on a poison plate thickness of 0.075 inches for initial sensitivity calculations and 0.125 inches for the final design basis calculations. The important parameter is the minimum B-10 areal density; therefore the modeled thickness of the poison plate does not affect the results of the calculation.

The basic calculational KENO model, as discussed above, is a 14.49-inch axial section and full-radial cross section of the basket within the cask with periodic boundary conditions at the axial boundaries (top and bottom) and reflective boundary conditions at the radial boundaries (sides). This axial section essentially models one building block of the egg crate basket structure. Periodic boundary conditions ensure that the resulting KENO model is essentially infinite in the axial direction. The model does not explicitly include the solid neutron shield (polyester resin); however the infinite array of casks without the neutron shield does contain unborated water between the casks. For the purpose of storage, the TN-40HT cask configuration is not expected to encounter any regions containing unborated water once the fuel assemblies are loaded. Therefore, this hypothetical configuration that models an infinite array of casks in close reflection is conservative.

The fuel assemblies within the basket are modeled as arrays of fuel pins and guide/instrument tubes. Spacer grids and sub-components like oversleeves (when present) are not modeled since their effect on reactivity is insignificant. The fuel compartment tubes surround each fuel assembly that is in-turn bounded by the basket plates consisting of 0.4375" aluminum/poison plates. These plates are arranged to represent an egg-crate design with the 0.3625"- Aluminum and a 0.075"-poison plate. The thermal expansion and egg-crate slot gaps are not modeled assuming plate continuity, thus replacing the more absorbing borated water (internal moderator) with basket material (aluminum/poison). KENO model plots in 2D for the various views of the basket compartment are shown in Figure A3.3-29 through Figure A3.3-35.

There are a total of 13 poison plates in the TN-40HT basket. They are located at all the faces where at least five fuel assemblies are lined up. Thus, all the interior 30 fuel assemblies are surrounded by poison plates on all four faces and the outer 10 fuel assemblies do not have poison plates on the radially outward face. The fuel assembly and poison plate positions (and the aluminum plate positions) in the KENO model of the basket is shown in Figure A3.3-36. Even though the poison and aluminum plates have been shown as discrete plates around the fuel compartment, they are all continuous running from one end of the basket to the other.

The basket structure is connected to the cask shell by perimeter rail assemblies. The rail material is a combination of aluminum and SS304. The rails provide a structural function as well as provide a heat conduction path from the basket to the cask shell. Due to limitations in the geometry options available in KENO, it is not possible to exactly model the rails. However, bounding evaluations are performed to determine the effect of rail material / geometry modeling on the reactivity of the system.

A list of all the geometry units used in the basic KENO model is shown in Table A3.3-21. Figure A3.3-37 shows the various radial "cylinders" utilized in the KENO model surrounding the fuel assemblies. Basically, this shows the cask details.

The first model developed uses nominal dimensions for the fuel compartments, fuel compartment thickness, poison plate thickness and the fuel assemblies centered in the fuel compartment. The rails are modeled simply using horizontal and vertical aluminum plates around the periphery of the basket. The cavity between the fuel compartments and the cask inner shell is modeled with internal moderator.

This basic KENO model is used to determine the most reactive fuel assembly design, the most reactive assembly-to-assembly pitch, and to determine the most reactive cask configuration accounting for manufacturing tolerances. The second model is of the most reactive configuration identified above. This model is used to determine the maximum allowable initial enrichment for the TN-40HT cask. Plots of the various KENO models utilized in these calculations are shown in Figure A3.3-38 through Figure A3.3-41.

A3.3.4.1.3.2 PACKAGE REGIONAL DENSITIES

The Oak Ridge National Laboratory (ORNL) SCALE code package (Reference 14) contains a standard material data library for common elements, compounds, and mixtures. All the materials used for the TN-40HT cask analysis are available in this data library. Table A3.3-22 provides a complete list of all the relevant materials used for the criticality evaluation. The material density for the B10 in the poison plates includes a 10% reduction for B-Al poison and a 25% reduction for Boral® poison.

A3.3.4.1.4 CRITICALITY CALCULATIONS

This section describes the analysis methodology utilized for the criticality analysis. The analyses are performed with the CSAS25 module of the SCALE system. A series of calculations are performed to determine the relative reactivity of the various fuel assembly designs to determine the most reactive assembly type without CCs. The most reactive intact fuel design, as demonstrated by the analyses, is the WE 14x14 Standard assembly. The most reactive credible configuration is an infinite array of flooded casks, each containing 40 fuel assemblies, with minimum fuel compartment tube ID, maximum fuel compartment tube thickness, minimum stainless steel plate thickness, and minimum assembly-to-assembly pitch.

Finally, using the most reactive credible configuration determined above, k_{eff} is determined for the maximum initial enrichment for the WE 14x14 assembly class at a soluble boron concentration of 2450 ppm.

A3.3.4.1.4.1 CALCULATIONAL METHOD

A3.3.4.1.4.1.1 COMPUTER CODES

The CSAS25 control module of SCALE-4.4 (Reference 14) is used to calculate the effective multiplication factor (k_{eff}) of the fuel in the TN-40HT Cask. The CSAS25 control module allows simplified data input to the functional modules BONAMI-S, NITAWL-II, and KENO V.a. These modules process the required cross sections and calculate the k_{eff} of the system. BONAMI-S performs resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections. NITAWL-II applies a Nordheim resonance self-shielding correction to nuclides having resonance parameters. Finally, KENO V.a calculates the k_{eff} of a three-dimensional system. A sufficiently large number of neutron histories are run so that the standard deviation is below 0.0015 for all calculations.

A3.3.4.1.4.1.2 PHYSICAL AND NUCLEAR DATA

The physical and nuclear data required for the criticality analysis include the fuel assembly data and cross-section data as described below.

Table A3.3-19 provides the pertinent data for criticality analysis for each fuel assembly evaluated in the TN-40HT cask. The criticality analysis used the 44-group cross-section library built into the SCALE system. ORNL used ENDF/B-V data to develop this broad-group library specifically for criticality analysis of a wide variety of thermal systems.

A3.3.4.1.4.1.3 BASES AND ASSUMPTIONS

The analytical results reported in Section A4 demonstrate that the cask containment boundary, basket structure and fuel cladding do not experience any significant distortion under hypothetical accident conditions. Therefore, for both normal and hypothetical accident conditions the cask geometry is identical except for the neutron shield and neutron shield jacket (outer skin). As discussed above, the neutron shield and neutron shield jacket (outer skin) are removed and the interstitial space modeled as water.

The TN-40HT cask is modeled with KENO V.a using the available geometry input. This option allows a model to be constructed that uses regular geometric shapes to define the material boundaries. The following conservative assumptions are also incorporated into the criticality calculations for intact fuel:

1. No credit is taken for the presence of burnable poisons such as Gadolinia, Erbia or any other absorber in the fuel.

2. CCs that extend into the active fuel region, such as BPRAs are conservatively assumed to exhibit neutronic properties of $^{11}B_4C$.

3. Unirradiated fuel – no credit taken for fissile depletion due to burnup or fission product poisoning.

4. For intact fuel, the lattice average fuel enrichment is modeled as uniform everywhere throughout the assembly. Natural Uranium blankets and axial or radial enrichment zones are modeled as enriched uranium at the lattice average enrichment.

5. All fuel rods are filled with full density fresh water in the pellet/cladding gap.

6. Only a 14.49-inch section of the basket (actual is 15.0-inches) with fuel assemblies is explicitly modeled with periodic axial boundary conditions, therefore the model is effectively infinitely long and the actual poison height for each section is conservatively modeled about 0.5 inches shorter.

7. The neutron shield material is modeled using water and is not expected to result in any significant change in the system reactivity since it is located in a relatively unimportant region for criticality.

8. Only 90% credit is taken for the B10 in the B-Al poison plates and 75% credit for Boral® in the KENO models.

9. The fuel rods are modeled assuming a stack density of 96.5% theoretical density with no allowance for dishing or chamfer. This assumption conservatively increases the total fuel content in the model.

10. All calculations are carried out at a temperature of 20 °C (293 K).

11. All basket stainless steel materials are modeled as SS304. The cask steel materials are modeled as SS304 and carbon-steel. The small differences in the composition of the various stainless steels have no effect on results of the calculation.

12. All zirconium based materials in the fuel (including ZIRLO) are modeled as Zircalloy-4. The small differences in the composition of the various clad / guide tube materials have no effect on the results of the calculation.

13. The thermal expansion and egg-crate gaps are replaced with the basket material wherever present. This modeling assumption results in replacing the soluble boron moderator in the gap regions with basket material (aluminum/poison), thereby, decreasing the neutron absorption around the fuel.

14. The transition rails between the basket and the cask shell are modeled as solid aluminum with cuboid "holes" of borated water in the aluminum.

A3.3.4.1.4.1.4 DETERMINATION OF K_{EFF}

The Monte Carlo calculations performed with CSAS25 (KENO V.a) use a flat neutron starting distribution. The total number of histories traced for each calculation is approximately 800,000. This number of histories is sufficient to achieve source convergence and produce standard deviations of less than 0.0015 in Δk_{eff}. The maximum k_{eff} for the calculation is determined with the following formula:

$$k_{eff} = k_{KENO} + 2\sigma_{KENO}$$

A3.3.4.1.4.2 FUEL LOADING OPTIMIZATION

A. Determination of the Most Reactive Fuel Type

All fuel assemblies listed in Table A3.3-19 are evaluated to determine the most reactive fuel assembly type with initial enrichments of 4.50 wt. % U-235. The fuel types are analyzed with fresh water in the fuel pellet cladding annulus, a soluble boron concentration of 2400 ppm and a fixed borated aluminum poison loading of 18.7 mg $B10/cm^2$. The parameters utilized for these sensitivity evaluations are not critical since it is intended to determine the most reactive configuration that estimates a "relative" reactivity and not "absolute" reactivity of the system. Nominal basket dimensions are utilized in the KENO model.

These evaluations are carried out for two fuel assembly positions within the fuel compartment – "centered" and "inward." The centered position is when the fuel assemblies are centered within the fuel compartment while the inward position is when the fuel assemblies are positions closest relative to the center of the basket. The inward position results in the fuel assemblies being clustered closer together, thereby resulting in an increase in the reactivity. The internal moderator density (IMD) is varied from 80% to 100% of full density to determine any sensitivity of the fuel assembly design reactivity to moderator density.

In all other respects, the model is the same as the basic model described above. A simple representation of the peripheral rails as shown in Figure A3.3-38 is utilized for this evaluation.

The Cask model for this evaluation differs from the actual design in the following ways:

- The neutron shield of the cask is conservatively replaced with water between the casks.
- The stainless steel and aluminum basket rails, which provide support to the fuel compartment tube grid, are modeled as aluminum plates located in the periphery.
- The "egg-crate" section length is modeled as 14.49" high (12.67" basket section + 1.75" steel insert bar + 0.07" gap). The actual design for the TN-40HT has an "egg-crate" section length of 15.0" (13.18" basket section + 1.75" steel insert bar + 0.07" gap).

The results of this evaluation are provided in Table A3.3-23. The most reactive design is the Westinghouse 14x14 Standard fuel assembly. The results also indicate that the inward position of the fuel assemblies results in a more reactive configuration and that the relative reactivity ranking of the fuel assemblies remains unchanged with IMD variations. The case corresponding to the highest k_{eff} in this evaluation is highlighted in Table A3.3-23.

B. Determination of the Most Reactive Configuration

The fuel loading configuration of the cask affects the reactivity of the package. Several series of analyses determined the most reactive configuration for the TN-40HT cask.

For this analysis, the most reactive fuel type is used to determine the most reactive configuration. The cask is modeled with the WESTD assembly, over a 14.49-inch axial section with periodic axial boundary conditions and reflective radial boundary conditions. This represents and infinite array in the x-y direction of the cask that is infinite in length, which is conservative for criticality analysis. The starting model is identical to the model used above.

The cask model for this evaluation differs from the actual design in the following ways:

- The neutron shield of the cask is conservatively replaced with water between the casks.
- The stainless steel and aluminum basket rails, which provide support to the fuel compartment tube grid, are modeled as aluminum plates located in the periphery. Further, an evaluation is performed to determine an acceptable and conservative representation of the rails in the basket periphery.
- The "egg-crate" section length is modeled as 14.49" high (12.67" basket section + 1.75" steel insert bar + 0.07" gap). The actual design for the TN-40HT has an "egg-crate" section length of 15.0" (13.18" basket section + 1.75" steel insert bar + 0.07" gap).

Each evaluation is performed with the fuel assemblies in the inward position at various IMD values to determine the optimum moderator density where the reactivity is maximized.

The first set of analyses evaluates the effect of fuel compartment tube internal width on the system reactivity. The model starts with the nominal poison plate thickness modeled as above. For this evaluation the fuel compartment tube internal width is varied from 8.00 to 8.10 inches square. The results of the evaluation are given in Table A3.3-24. The results show that the most reactive configuration is with the minimum fuel compartment tube size. The balance of this evaluation uses the minimum fuel compartment tube width because it represents the most reactive configuration. An example input file for the most reactive fuel evaluation is included in Appendix A3A.

The second set of analyses evaluates the effect of fuel compartment tube and the stainless steel tie bar thicknesses on reactivity. The model starts with the most reactive configuration basket model determined above. The compartment tube thickness is varied from 0.1775" to 0.2325". The stainless steel tie bar thickness is varied from 0.4275" to 0.4925". Varying the thickness of the tie bar also serves the purpose of varying the aluminum plate thickness as the thickness of the poison plate is maintained constant. The results in Table A3.3-25 show that the system reactivity is not very sensitive to the compartment and stainless steel bar thicknesses.
However, in order to obtain a design basis configuration for criticality analyses, a few of the more reactive combinations are evaluated for variations with other parameters. Therefore, the next sets of analyses are performed with bounding configurations that are highlighted in Table A3.3-25. Note that there are four bounding configurations and they are statistically similar.

The third set of analyses evaluates the effect of poison plate thickness and modeling on the system reactivity. Three of the bounding configurations identified above are utilized in this evaluation. For this evaluation Boral® is utilized as the poison material with a core thickness of 0.050" and a total (core + clad) thickness of 0.075". The effect of poison plate thickness (and hence the absorber thickness) variation on the system reactivity is shown to be statistically insignificant based on the results in Table A3.3-26. These results also indicate that poison plates of higher thicknesses can be used as long as the minimum absorber loading is maintained. These results also demonstrate that there is no effect on the reactivity due to the aluminum panels when Boral® poison is used. These results further indicate that effect of a change in the thickness of the egg-crate plates (poison and aluminum) is statistically insignificant provided the total thickness remains constant.

The fourth set of analyses evaluates the effect of rail structure modeling on the system reactivity. Due to the limitations in the geometry capabilities of the CSAS25 code, it is not possible to exactly model the rail structure. However, due to relatively low importance of the rail structure modeling to the criticality of the system, an exact model is not essential. In addition to the simplistic rail model utilized so far in these evaluations, three additional variations of the rails are also evaluated. The first variation is based on a modeling the rails as internal moderator (*rail1* option). The second variation is based on *rail1* option with cuboids of aluminum placed within the periphery using "hole" modeling (*rail2* option). The third variation is different from the *rail2* option where the periphery is modeled with solid aluminum with cuboids of internal moderator (*rail3* option where the aluminum and internal moderator from *rail2* are interchanged). The modeling of the cuboids as "holes" at the periphery (*rail2* and *rail3* options) is shown in Figure A3.3-40.

Note that the models – *rail1*, *rail2* and *rail3* represent a reduction in the volume fraction of internal moderator in the basket periphery. The results of the evaluation are given in Table A3.3-27. These results indicate that the *rail3* option results in a bounding configuration. The results also indicate that the reactivity of the basket increases with a reduction in the volume fraction of the internal moderator at the basket periphery. It is clear based on a comparison of the rail configurations from Figure A3.3-28 and Figure A3.3-40 that the *rail3* option conservatively models a lower volume fraction of internal moderator. Therefore, modeling the basket periphery with the *rail3* option is appropriate and conservative. All the four bounding configurations identified above are utilized in this evaluation. These calculations also result in the determination of the most reactive configuration for the four bounding configurations evaluated.

Based on these evaluations the most reactive cask configuration is for:

- Fuel assemblies pushed toward the center of the basket (inward arrangement),
- Minimum fuel compartment tube internal width,
- Maximum fuel compartment tube wall thickness,
- Nominal poison plate thickness,
- Minimum stainless steel bar thickness and
- Basket periphery modeled using the *rail3* option.

C. Maximum Initial Enrichment for the TN-40HT Cask

The analysis performed in this section is performed using the most reactive configuration as determined in Section B above. The internal moderator density is varied to determine the peak reactivity for the specific configuration. The maximum initial enrichment (5.00 wt. % U-235) is also shown in Table A3.3-17.

The cask model for this evaluation differs from the actual design in the following ways:

- The neutron shield of the cask is conservatively replaced with water between the casks.
- The stainless steel and aluminum basket rails, which provide support to the fuel compartment tube grid, are modeled conservatively using the *Rail3* option.
- The worst case material conditions, as determined in the previous Section above, are modeled,
- The "egg-crate" section length is modeled as 14.49" high (12.67" basket section + 1.75" steel insert bar + 0.07" gap). The actual design for the TN-40HT has an "egg-crate" section length of 15.0" (13.18" basket section + 1.75" steel insert bar + 0.07" gap).

A fixed poison loading of 33.7 mg B-10/cm^2 is utilized in the criticality calculations as described in Table A3.3-17. The soluble boron concentration utilized for these calculations is 2450 ppm. A maximum initial enrichment of 5.0 wt. % U-235 is utilized in these calculations. An example input file is included in Appendix A3A.

The most reactive WE 14x14 class assembly is the WE 14x14 Standard fuel assembly as demonstrated in Table A3.3-23. The results for the WE 14x14 class assembly calculations without CCs are listed in Table A3.3-28.

The results for the WE 14x14 class assembly calculations with CCs are listed in Table A3.3-29. The results demonstrate that the no reduction in the initial enrichment is required due to the presence of CCs.

The maximum calculated k_{eff} corresponds to the configuration with an initial enrichment of 5.0 wt. % U235 with 2450 ppm borated water with CCs. The maximum calculated k_{eff} is 0.9357 ± 0.0008 or 0.9373 which is below the USL (0.9419) calculated in Section A3.3.4.3.2. The maximum calculated dry k_{eff} (normal condition for storage) is 0.5787 ± 0.0004 or 0.5795.

A3.3.4.2 ERROR CONTINGENCY CRITERIA

Provision for error contingency is built into the criterion used in Section A3.3.4.1 above. The criterion, used in conjunction with the KENO-Va and NITAWL codes, is common practice for licensing submittals. Because conservative assumptions are made in modeling, it is not necessary to introduce additional contingency for error.

A3.3.4.3 VERIFICATION ANALYSIS – BENCHMARKING

The computer codes described in Section A3.3.4.1.4.1.1 were used to benchmark 121 experiments. The results of these benchmarks were used to determine the Upper Subcritical Limit (USL-1).

The benchmark problems used to perform this verification are representative of benchmark arrays of commercial light water reactor (LWR) fuels with the following characteristics:

(1) water moderation,

(2) boron neutron absorbers,

(3) unirradiated light water reactor type fuel (no fission products or "burnup credit") near room temperature (vs. reactor operating temperature),

(4) close reflection, and

(5) uranium oxide.

The 121 uranium oxide experiments were chosen to model a wide range of uranium enrichments, fuel pin pitches, assembly separation, soluble boron concentration and control elements in order to test the codes ability to accurately calculate k_{eff}. These experiments are discussed in detail in NUREG/CR-6361 (Reference 18).

A3.3.4.3.1 BENCHMARK EXPERIMENTS AND APPLICABILITY

A summary of all of the pertinent parameters for each experiment is included in Table A3.3-30 along with the results of each run. The best correlation is observed for fuel assembly separation distance with a correlation of 0.66. All other parameters show much lower correlation ratios indicating no real correlation. All parameters were evaluated for trends and to determine the most conservative USL.

The USL is calculated in accordance to NUREG/CR-6361 (Reference 18). USL Method 1 (USL-1) applies a statistical calculation of the bias and its uncertainty plus an administrative margin (0.05) to the linear fit of results of the experimental benchmark data. The basis for the administrative margin is from NUREG/CR-5661 (Reference 20). Results from the USL evaluation are presented in Table A3.3-31.

The criticality evaluation used the same cross section set, fuel materials and similar material/geometry options that were used in the 121 benchmark calculations as shown in Table A3.3-30.

The modeling techniques and the applicable parameters listed in Table A3.3-32 for the actual criticality evaluations fall within the range of those addressed by the benchmarks in Table A3.3-30.

A3.3.4.3.2 RESULTS OF THE BENCHMARK CALCULATIONS

The results from the comparisons of physical parameters of each of the fuel assembly types to the applicable USL value are presented in Table A3.3-32. The minimum value of the USL is determined to be 0.9419 based on comparisons to the limiting assembly parameters as shown in Table A3.3-32.

A3.3.5 RADIOLOGICAL PROTECTION

Provisions for radiological protection by confinement barriers and systems are described in Section A3.3.2.1.

A3.3.5.1 ACCESS CONTROL

The ISFSI does not require the continuous presence of operators or maintenance personnel. In addition, it is located within a fenced-in area shared only with the Equipment Storage Building and Security Building which will be used for storage of cask handling and security related equipment and will not be continuously manned. Access to the fenced-in area is limited to personnel needed during operations at the ISFSI. Activities will include periodic inspections of these facilities, emplacement of storage casks, and security checks. These activities will be defined and controlled by the Radiation Protection and Security procedures manuals covering the ISFSI.

A3.3.5.2 SHIELDING

The storage casks provide sufficient radiation shielding to allow handling of the loaded casks with as low as reasonably achievable (ALARA) doses to the operators and to comply with the radiation limits in 10 CFR72. For specific dose estimates, see Section A7.

A3.3.5.3 RADIOLOGICAL ALARM SYSTEMS

There are no credible events which result in releases of radioactive products or unacceptable increases in direct radiation. In addition, the releases postulated as the result of the hypothetical accidents described in Section A8 are of a very small magnitude. Therefore, radiological alarm systems are not necessary. However, as described in Section A3.3.2.1, nonsafety-grade pressure monitors are provided. Procedures to be followed when these alarms are activated will be specified in the ISFSI operating procedures.

A3.3.6 FIRE AND EXPLOSION PROTECTION

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

As indicated in Section 2.2, overpressures of 2.25 psi can be conservatively postulated to occur at the ISFSI as a result of accidents involving explosive materials which are stored or transported near the site. This impact is less than that postulated to result from the tornado wind loading and missile impact analysis, as described in Section A3.2.1, and is well within the design basis of the cask.

A3.3.7 MATERIAL HANDLING AND STORAGE

A3.3.7.1 SPENT FUEL HANDLING AND STORAGE

The handling of spent fuel within the Prairie Island Nuclear Generating Plant will be conducted in accordance with existing fuel handling procedures. Only fuel that is not a DAMAGED FUEL ASSEMBLY will be considered for storage in the TN-40HT casks.

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

a. has visible deformation of the rods in the spent nuclear fuel assembly. Note: This is not referring to the uniform bowing that occurs in the reactor. This refers to bowing that significantly opens up the lattice spacing;
b. has individual fuel rods missing from the assembly. Note: The assembly is not a DAMAGED FUEL ASSEMBLY if a dummy rod that displaces a volume

equal to, or greater than, the original fuel rod, is placed in the empty rod location;

c. has missing, displaced, or damaged structural components such that radiological and/or criticality safety is adversely affected (e.g., significantly changed rod pitch);
d. has missing, displaced, or damaged structural components such that the assembly cannot be handled by normal means (i.e., crane and grapple);
e. has reactor operating records (or other records) indicating that the spent nuclear fuel assembly contains cladding breaches; or
f. is no longer in the form of an intact fuel bundle (e.g., consists of, or contains, debris such as loose fuel pellets or rod segments).

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

A3.3.7.2 RADIOACTIVE WASTE TREATMENT

The ISFSI will not generate radioactive waste. However, cask loading and decontamination operations, while in the Auxiliary Building, may generate small amounts of waste. This waste is disposed of in accordance with the radioactive waste handling procedures described in Section A6, and is part of the 10 CFR50 licensed activities. Waste storage facilities are neither required nor provided for the ISFSI.

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TABLE A3.1-1
PRAIRIE ISLAND FUEL ASSEMBLY DESIGN CHARACTERISTICS

Fuel Designations	Exxon/ANF (ANP) (14x14) Standard	Exxon/ANF (ANP) High Burnup (14x14)	Exxon/ANF (ANP) TOPROD (14x14)	Westinghouse Standard (14x14)	Westinghouse OFA (Including Vantage+) (14x14)
Max Length (in)	161.3	161.3	161.3	161.3	161.3
Max Width (in)	7.763	7.763	7.763	7.763	7.763
Maximum No. of Fuel Rods	179	179	179	179	179
Nominal Fuel Rod OD (in)	0.4240	0.4260	0.4170	0.4220	0.4000
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4/ZIRLO
Guide Tube #	16	16	16	16	16
Instrument Tube #	1	1	1	1	1
Maximum MTU/assembly(1)	0.370	0.370	0.370	0.410	0.360

Note:

(1) The maximum MTU/assembly is calculated based on the theoretical density. The calculated value is higher than the actual value.

TABLE A3.3-8
MATERIAL THERMAL PROPERTIES
(PAGE 6 OF 7)

Aluminum Alloy 6063 (Radial Neutron Shield Boxes)

(The conductivity, diffusivity and density values are from Reference 28.)

Al96063	Thermal conductivity		Thermal Diffusivity	Specific heat capacity *
Temperature (°F)	(Btu/hr-ft-°F)	(Btu/hr-in-°F)	(ft^2/hr)	(Btu/lbm-°F)
70	120.8	10.067	3.340	0.214
100	120.3	10.025	3.299	0.215
150	119.7	9.975	3.232	0.219
200	119.0	9.917	3.177	0.221
250	118.5	9.875	3.133	0.223
300	118.1	9.842	3.088	0.226
350	118.0	9.833	3.040	0.229
400	117.6	9.800	3.000	0.231
ρ = 0.098 lbm/in^3				

Neutron Absorber (Poison) Plates

Boral™	Thermal conductivity of Poison Plate	Specific heat of Poison Plate (Cp_t)
Temperature (°F)	(Btu/hr-in-°F)	(Btu/lbm-°F)
100	4.14	0.22
500	3.70	0.33

Solid Neutron Shield Resin (Borated Polyester – also used for Polypropylene)

Thermal conductivity	Density	Specific heat capacity
(Btu/hr-in-°F)	(lbm/in^3)	(Btu/lbm-°F)
0.0083	0.057	0.311

* Thermal diffusivity, $\alpha = \frac{k}{\rho \, c_P}$, is used to calculate the specific heat with a density of 0.098 lbm/in^3 for aluminum alloys.

A4.2.3.3.3 BASKET

The basket is designed, fabricated and inspected in accordance with the ASME Code Subsection NG (Reference 1) to the maximum practical extent. Alternatives to the ASME Code are discussed in Section A3.5.

The stress limits for the basket are summarized in Table A4.2-4. The basket fuel compartment wall thickness is established to meet heat transfer, nuclear criticality, and structural requirements. The basket structure must provide sufficient rigidity to maintain a subcritical configuration under the applied loads. The 304 stainless steel members in the TN-40HT basket are the primary structural components. The aluminum plates are the primary heat conductors and neutron poison plates provide the necessary criticality control.

The fusion welds between the stainless steel support bars and the stainless steel fuel compartments shall be qualified by testing. The 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-1342 (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.

A4.2.3.4 EVALUATION

The stress calculations performed on the cask and basket are presented in Appendices A4A and A4B respectively. The off-normal loads are bound by normal loads and compared with normal load allowables. Finite element models of the cask body and basket have been developed, and detailed computer analyses have been performed using the ANSYS computer program (Reference 3). The stress analysis of the lid bolts is based on the methodology of NUREG/CR-6007 (Reference 4). Other components such as trunnions are analyzed using conventional textbook methods. Table A4.2-1 lists the specific individual load cases analyzed for each major cask component. The SAR sections where these analyses are described and the tables listing the stress results, where applicable, are also indicated.

Section A3.2 categorizes the loads for the cask body as indicated in Tables A3.2-5 through A3.2-8 into Normal (Level A) and Hypothetical Accident (Level D) Service Loadings. Table A3.2-9 and Table A3.2-10 lists the load combinations to be evaluated. Table A4.2-5 and Table A4.2-6 summarize the combination of the cask body individuals loads evaluated for normal conditions and hypothetical accident conditions respectively. Table A4.2-7 summarizes the basket load combinations. Each combination is a set of loads that are assumed to occur simultaneously.

Key dimensions for the TN-40HT cask body are shown in Figure A4.2-1.

A4.2.3.4.1 CONTAINMENT VESSEL

The evaluation of the containment vessel stresses are summarized along with the evaluation of the gamma shield stress discussed in Section A4.2.3.4.2 below.

A4.2.3.6.3 LUBRICANTS AND CLEANING AGENTS

Neolube, Loctite N-5000, or equivalent may be used to coat the threads and bolt shoulders of the TN-40HT cask closure bolts. Loctite N-5000 or equivalent may be used to coat the contact areas of the top and bottom trunnions prior to lifting operations. The lubricant shall be removed prior to immersing the cask into the spent fuel pool unless the lubricant has been approved for compatibility with the spent fuel pool water.

The cask and basket are cleaned at the fabricator in accordance with approved procedures. The cleaning agents and lubricants have no significant effect on the cask materials and their safety related functions.

A4.2.3.6.4 HYDROGEN GENERATION

Prairie Island's report to the NRC (Reference 7) in response to NRC Bulletin 96-04 demonstrates that galvanic reactions in hydrogen generation are insignificant for the TN-40 cask. This report is also applicable for the TN-40HT cask.

A4.2.3.6.5 EFFECT OF GALVANIC REACTIONS ON THE PERFORMANCE OF THE CASK

There are no significant reactions that could reduce the overall integrity of the cask or its contents during storage. The period of immersion in pool water is too short, and any oxidizing gases remaining after vacuum drying and helium backfill is too small to cause corrosion that could have significant effect on the fuel cladding, neutron absorber integrity, or the basket and cask structural performance.

There are no reactions that would cause binding of the mechanical surfaces or the fuel to basket compartment boxes due to galvanic or chemical reactions.

The stainless steel, aluminum, neutron absorber and thermal spray are negligibly affected by the short term exposure to borated water during loading. The three acceptable neutron absorber materials, Boral®, borated aluminum, and metal matrix composites, are all aluminum-based, with the addition of boron in the inert form of boron carbide, aluminum diboride, or titanium diboride. The corrosion behavior of these materials is bounded by Boral® because of its porous core.

While formation of blisters in Boral® during vacuum drying and heating has been reported, this has not been associated with displacement of the Boral® core material containing the boron carbide and therefore has no effect on the Boral® criticality safety design function (Reference 8). Furthermore, in the TN-40HT cask, the Boral® is captured between the structural basket components, including 3/16 inch thick walls of the fuel compartments, to provide it with added mechanical support and durability.

The outer aluminum lid seals may experience some combination of crevice and galvanic corrosion if they are exposed to water for an extended period. However, this would affect only the outer (non-containment) seal, and the overpressure monitoring system would detect any significant leakage.

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A4.2.3.8.3 MATERIAL PROPERTIES OF FUEL CLADDING

The fuel cladding is evaluated based on the mechanical properties obtained from Reference 17 which provides expressions to calculate the modulus of elasticity and yield strength for both Zircaloy-2 (BWR cladding) and Zircaloy-4 (PWR cladding). These expressions were derived from correlations of experimental results of several different investigations. Assumptions used include the following:

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Temperature is a significant factor in derivation of Zircaloy properties. These properties are calculated over a range of temperatures for Zircaloy-4. An example calculation is carried out below for Zircaloy-4 (PWR cladding) at 750°F.

Modulus of Elasticity (Page 4 of Reference 17)

$$E = \frac{1.088 \times 10^{11} - 5.475 \times 10^{7} \cdot T + K_1 + K_2}{K_3}$$

Where:

E = elastic modulus, Pa

T = temperature, K
= 672.052K (750°F)

$K_1 = \left(6.61 \times 10^{11} + 5.912 \times 10^{8} \cdot T\right)\Delta$
= 1.270×10^9

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Where:

CW = cold work, unitless ratio of areas (valid between 0 and 0.75)

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$K_3 = 0.88 + 0.12\exp\left(-\Phi/10^{25}\right)$

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Φ = fast neutron fluence, n/m^2

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Substituting these values into the expression for E above:

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Yield Stress (Equation 3, Page 3 of Reference 17)

$$\sigma_y = \left[\frac{K}{E^n}\left(\frac{\varepsilon}{10^{-3}}\right)^m\right]^{\frac{1}{1-n}}$$

Strength coefficient

$$K = K(T)\cdot(1+K(CW)+K(\Phi))/K(Zry)$$

Where:

$$K(T) = 1.17628\times10^{9} + 4.54859\times10^{5}T - 3.28185\times10^{3}T^{2} + 1.72752\cdot T^{3}$$
$$= 5.24\times10^{8}$$

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$K(\Phi) = 0.731995$ for $\Phi > 7.5 \times 10^{25}$ n/m^2

$K(Zry) = 1.0$ for Zircaloy-4

Substituting these values into the expression for K above:

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Strain Hardening Exponent

$$n = n(T)\cdot n(\Phi)/n(Zry)$$

Where:

n = strain hardening exponent

$$n(T) = -9.490\times10^{-2} + 1.165\times10^{-3}T - 1.992\times10^{-6}T^{2} + 9.588\times10^{-10}T^{3}$$
$$419.4 < T < 1099.0772K$$

$n(\Phi) = 1.608953$ $\Phi > 7.5\times10^{25} n/m^{2}$

$n(Zry) = 1.0$ for Zircaloy-4

Substituting these values into the expression for K above:

$$n = 0.07938\cdot1.608953/1.0 = 0.1277$$

Strain Rate

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Strain Rate Exponent

$m = 0.015 \qquad T < 750K$
$m =$ strain rate exponent

The values calculated above can then be inserted into the following expression for yield stress, σ_y:

$$\sigma_y = \left[\frac{K}{E^n}\left(\frac{\varepsilon}{10^{-3}}\right)^m\right]^{\frac{1}{1-n}}$$

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Calculated values for Modulus of Elasticity and Yield Stress
The expressions above are used to calculate the modulus of elasticity E and the yield stress σ_y over a range of temperatures. The result for Zircaloy-4 (PWR) is presented in Table A4.2-25, Figure A4.2-6 and Figure A4.2-7. Note that the figures included calculated data points not summarized in the Table.

ZIRLO vs Zircaloy-4
Reference 22 states that ZIRLO and Zircaloy-4 alloys are very similar in terms stress/strain characteristics. Therefore Zircaloy-4 properties above are adequate for modeling ZIRLO cladding.

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A4.2.3.8.6 CONCLUSION

From the above results, for an accident condition bottom end drop of a fuel assembly inside a TN-40HT cask, the maximum total strain remains in the elastic range. Since plastic deformation does not occur in this case, it can be concluded that the fuel assembly cladding will not fail in the event of an 18 inch bottom end drop accident.

A4.2.3.9 THERMAL STRESS OF FUEL CLADDING DUE TO UNLOADING OPERATIONS

To evaluate the effects of the thermal loads on the fuel cladding during unloading operations, the following assumptions are made:

- A conservative high maximum fuel cladding temperature of 700 °F and quench water temperature of 50 °F are used.
- The Fuel rod is assumed to be simply supported at both ends.
- The outer surface temperatures of the fuel cladding are conservatively assumed as shown in Figure A4.2-13. 50 °F (water), 212 °F (steam), and 700 °F (cladding) temperature occurs at three equal heights.
- The fuel cladding thickness and cladding outside diameter are reduced by 0.00270 inch to account for oxidation.

A4.2.3.9.1 FINITE ELEMENT MODEL

The finite element model is shown in Figure A4.2-14. ANSYS (Reference 3) finite element Plane 55 and Plane 42 (Axisymmetric) are used for thermal and structural analysis respectively. The fuel rod with the thinnest cladding (WE14 x 14 STD) is modeled, as this will result in the largest temperature gradient across the cladding (temperatures are kept constant at the inner and outer surfaces). The cladding thickness is 0.0216 inches and the rod outer diameter is 0.4166 inches. A tube length of 2 inches is considered for the analysis such that maximum stresses are not affected by the boundary conditions.

A4.2.3.9.2 MATERIAL PROPERTIES

The following material properties are used for the thermal and structural analysis:

Material Properties for Thermal Analysis

Temp °F	Conductivity Btu/hr-in- °F
212	0.655
392	0.689
572	0.732
752	0.790

Material Properties for Structural Analysis

Temp °F	E (psi)	α in/in- °F	ν	S_y (psi) at 750 °F
300	12.2×10^6	3.73×10^{-6}	0.404	126,102
400	11.7×10^6			116,272
500	11.2×10^6			108,921
600	10.7×10^6			102,512
700	10.2×10^6			95,793
750	9.93×10^6			92,000

A4.2.3.9.3 THERMAL ANALYSIS

Steady state thermal analysis was conducted using the surface nodal temperatures as shown in Figure A4.2-13. The inside surface nodal temperatures are all assumed to be 700 °F , and the outside surface temperatures to conservatively represent the quench water temperature. The temperature distribution resulting from this analysis is shown in Figure A4.2-15.

A4.2.3.9.4 THERMAL STRESS ANALYSIS AND RESULTS

A thermal stress analysis using the same model was conducted using the nodal temperatures obtained from the thermal analysis. The resulting nodal stress intensity distribution is shown in Figure A4.2-16. The maximum nodal stress intensity in the fuel cladding is 24.0 ksi. This stress is less than the yield strength of Zircaloy, which is 92 ksi at 750 °F.

A4.2.4 INSTRUMENTATION SYSTEM DESCRIPTION

No safety related instrumentation is required for the TN-40HT casks due to the passive nature of the ISFSI design.

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TABLE A4.2-2
CONTAINMENT VESSEL STRESS LIMITS

Classification	Stress Intensity Limit
Normal (Level A) Conditions(1)	
P_m	S_m
P_l	$1.5\ S_m$
$(P_m \text{ or } P_l) + P_b$	$1.5\ S_m$
Shear Stress	$0.6\ S_m$
Bearing Stress	S_y
$(P_m \text{ or } P_l) + P_b + Q$	$3\ S_m$
$(P_m \text{ or } P_l) + P_b + Q + F$	S_a
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	$\frac{\sigma_{tb}^2}{F_{tb}^2} + \frac{\tau_{yb}^2}{F_{yb}^2} \leq 1.0$
Hypothetical Accident (Level D)(2)	
P_m	Smaller of $2.4\ S_m$ or $0.7\ S_u$
P_l	Smaller of $3.6\ S_m$ or S_u
$(P_m \text{ or } P_l) + P_b$	Smaller of $3.6\ S_m$ or S_u
Shear Stress	$0.42\ S_u$
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	$\frac{\sigma_{tb}^2}{F_{tb}^2} + \frac{\tau_{yb}^2}{F_{yb}^2} \leq 1.0$

Notes:
1. Classifications and Stress Intensity Limits are as defined in ASME B&PV Code, Section III, Subsection NB.

2. Stress intensity limits are in accordance with ASME B&PV Code, Section III, Appendix F.

3. Bolt allowables are from Reference 4

**PROPRIETARY - TRADE SECRET INFORMATION
WITHHELD PURSUANT TO 10 CFR 2.390**

**PROPRIETARY - TRADE SECRET INFORMATION
WITHHELD PURSUANT TO 10 CFR 2.390**

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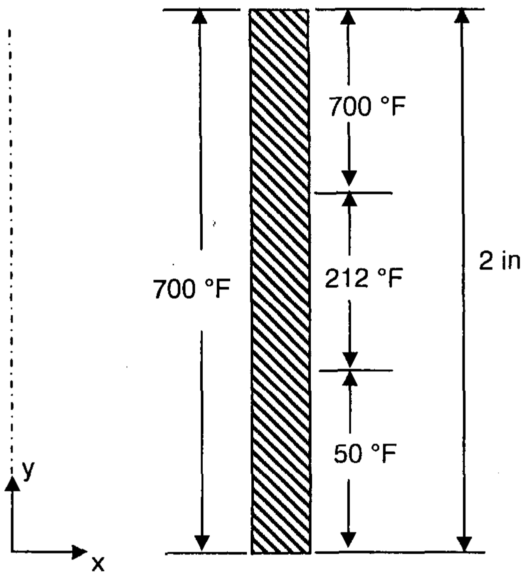

FIGURE A4.2-13
TN40HT FUEL CLADDING OUTER SURFACE TEMPERATURES

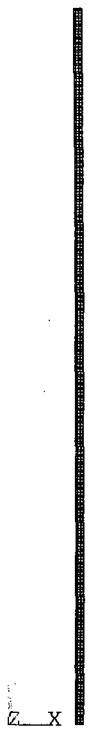

FIGURE A4.2-14
TN40HT FUEL CLADDING FINITE ELEMENT MODEL

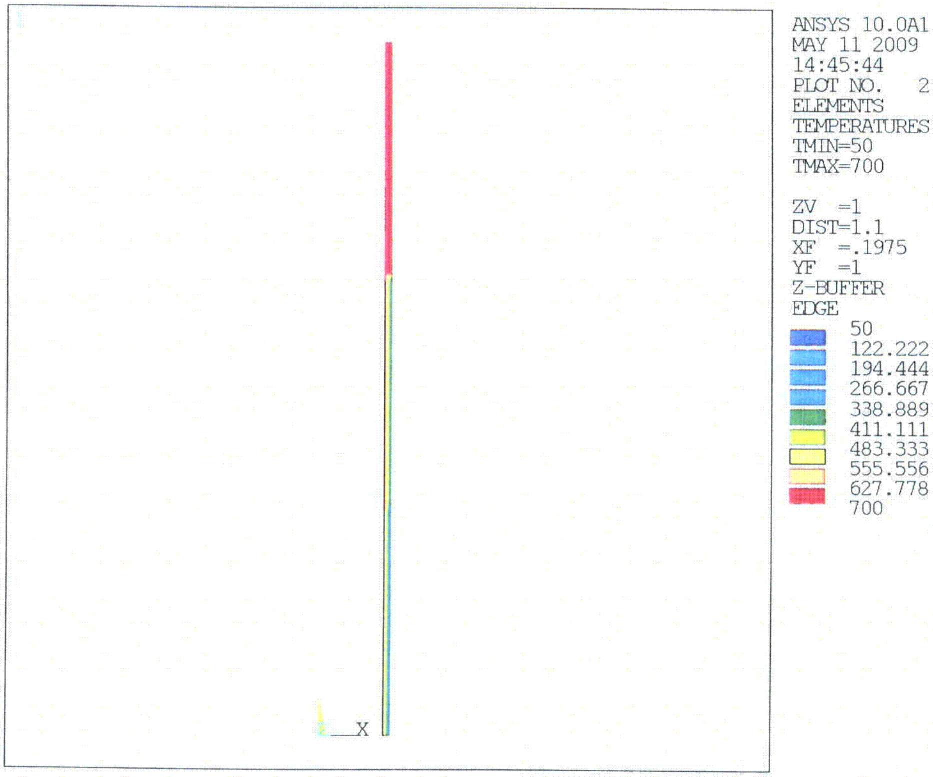

FIGURE A4.2-15
TN40HT FUEL CLADDING TEMPERATURE DISTRIBUTION

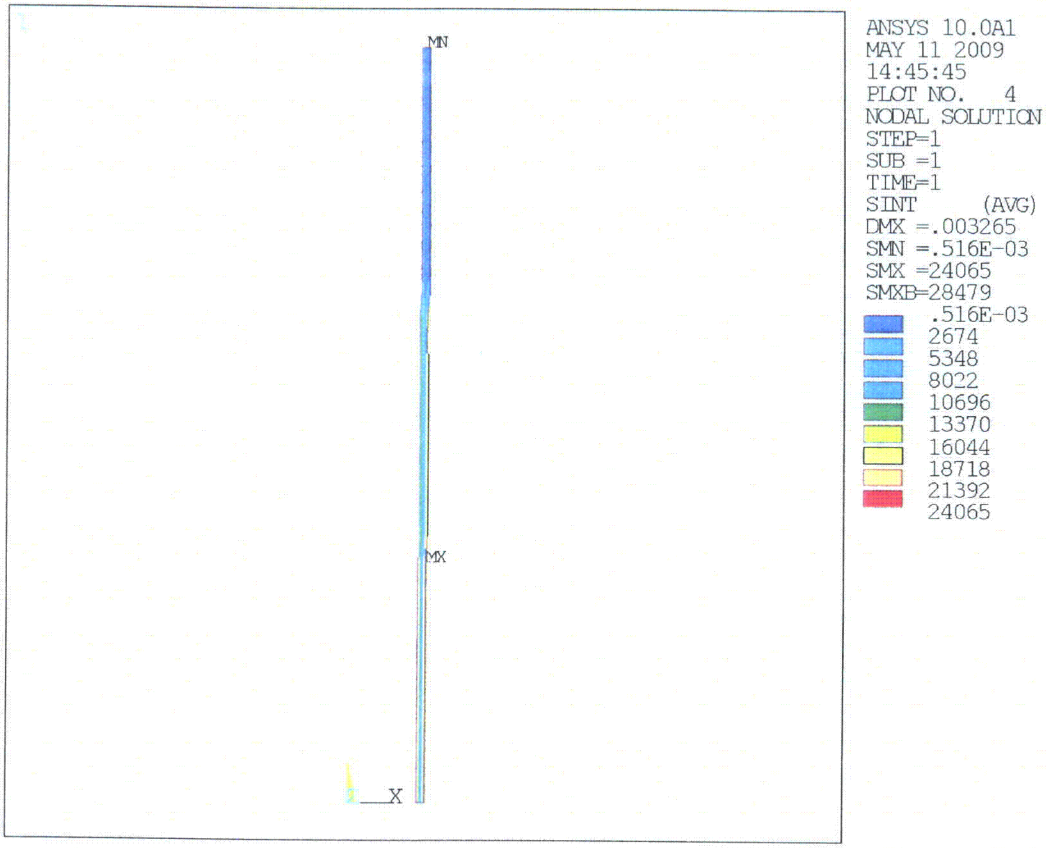

FIGURE A4.2-16
TN40HT FUEL CLADDING STRESS INTENSITY

A4A.10 TN-40HT STORAGE CASK END DROP ANALYSIS

The purpose of this section is to determine the rigid body accelerations for the TN-40HT Cask during a vertical drop height of 18 inches on concrete.

The rigid body transfer cask accelerations were predicted numerically by the LS-DYNA 3D explicit nonlinear dynamic analysis finite element solver, Version 9.71s (Reference 18). The methodology used in performing this analysis is based on work conducted at the Lawrence Livermore National Laboratory (LLNL), where an analysis methodology was developed and validated through comparisons with test data (Reference 19 and Reference 20). The analysis methodology was benchmarked in Reference 25.

The results of these analyses are used as input to the detailed analyses for the cask body, internal basket and fuel assemblies.

A4A.10.1 FINITE ELEMENT MODEL DESCRIPTION

The ANSYS finite element model of the TN-40HT Cask developed for the cask stress analysis (Appendix A4A.3) was simplified for use in the dynamic impact analysis. The TN-40HT Cask model consists of the cask body, simplified basket structure, concrete pad and soil. Each of these components was modeled using 3D 8-node brick elements. Fully integrated selectively-reduced solid elements were used for all elements to reduce the risk of hourglassing problems.

The finite element model was developed with ANSYS and transferred to LS-DYNA. Modifications were made to the LS-DYNA input file to add the material definitions, non-reflecting boundaries and equation of state into LS-DYNA. Features of the cask, such as the trunnions and neutron shield were neglected in terms of stiffness but their weight was lumped into the density of the cask.

The fuel and basket were modeled as a solid cylinder inside the cask walls with elastic material properties approximately equivalent to that of the structure as a whole.

The geometry of the cask finite element model including the cask internals, concrete and base soil is shown in Figure A4A.10-1 and Figure A4A.10-2.

Only ½ of the cask, internals, concrete and soil were modeled, because the entire arrangement is symmetric about the *x-y* plane. The concrete modeled was 16'-8" long, 6'-8" wide, and 3' thick, and the soil modeled was 66'-8" long, 18'-9" wide, and 39'-2" deep.

A4A.10.2 MATERIAL PROPERTIES

The material properties required to perform the analysis include modulus of elasticity, E, Poison's Ratio, v, and material density (ρ) for the cask body, basket, concrete, and soil. The concrete pad requires a more detailed material model since all of the significant nonlinear deformations occur in the concrete. Material properties used for the concrete and soil were based on those developed at Lawrence Livermore National Labs (Reference 19 and Reference 20).

All material properties were taken at room temperature. This is considered conservative because the cask loaded with spent fuel will typically reach temperatures higher than room temperature, and the lower modulus of elasticity at higher temperatures tends to soften the impact and consequently lower the computed g-loads.

TN-40HT Cask Material

The cask material properties were the same at those used in Appendix A4A.3. All cask materials were modeled as elastic.

Cask Component	Elastic Modulus (psi)	Density (lb-sec^2/in^4)	Poisson's Ratio
Lid Outer Plate	$27.8X10^6$	$8.230x10^{-4}$	0.3
Shield Plate	$29.0X10^6$	$8.230x10^{-4}$	0.3
Shell Flange	$27.8X10^6$	$7.324x10^{-4}$	0.3
Shell	$29.0X10^6$	$9.394x10^{-4}$	0.3
Bottom Plate	$29.0X10^6$	$7.324x10^{-4}$	0.3
Inner Liner	$27.8X10^6$	$7.324X10^{-4}$	0.3

Fuel and Basket Material

The basket structure material properties were the same as those used in Reference 20 except for density. The density of the basket was adjusted to calibrate the overall weight of the cask and basket assembly. The basket was modeled as elastic.

$E = 2.8X10^6$ psi
$v = 0.3$
$\rho = 3.215X10^{-4}$ lb sec^2/in^4

Total modeled weight of the cask and basket is 121,174 lbs since it is a half model. Therefore the total modeled weight is 242,348 lbs. Total actual weight of the cask and basket is 242,400 lbs.

Concrete Material

The concrete was modeled using material law 16 in LS-DYNA, which was developed specifically for granular type materials. The concrete data used in the analysis was originally designed by LLNL for the Shippingport Station Decommissioning Project in 1988. This model was also used in the LLNL (Reference 19) cask drop analysis. Material constants were implemented into Material Model 16, Mode II.B in LS-DYNA. The material represents 4,200 psi compressive strength concrete. A summary of the input used in the analysis is as follows.

$\rho = 2.09675\times10^{-4}$ lb sec^2 / in^4
$v = 0.22$
$a_0 = 1606$
$a_1 = 0.418$
$a_2 = 8.35\times10^{-5}$ psi^{-1}
$b_1 = 0$
$a_{0f} = 0.0$ psi
$a_{1f} = 0.385$

Effective Plastic Strain versus Scale Factor for Concrete Material

Effective Plastic Strain	Scale Factor, v
0	0
0.00094	0.289
0.00296	0.465
0.00837	0.629
0.01317	0.774
0.0234	0.893
0.04034	1.0
1.0	1.0

The maximum principal stress tensile failure cutoff was set at 870 psi. Strain rate effects were neglected in the analysis. Dilger (Reference 21) suggests that the major impact of strain rate effects is in the softening part of the stress-strain curve. Since the purpose of these analyses is primarily to predict the peak accelerations, the strain rate effects on the material behavior may be neglected.

The pressure-volume behavior of the concrete was modeled with the following tabulated pressure versus volumetric strain rate relationship using the equation of state feature in LS-DYNA.

Tabulated Pressure versus Volumetric Strain Rate for the Concrete Material

Volumetric Strain, ε	Pressure (psi)
0	0
-0.006	4,600
-0.075	5,400
-0.01	6,200
-0.012	6,600
-0.02	7,800
-0.038	10,000
-0.06	12,600
-0.0755	15,000
-0.097	18,700

An unloading bulk modulus of 700,000 psi was assumed to be constant at any volumetric strain, as was assumed in Reference 19.

One percent deformation was assumed in the concrete pad to account for the pad reinforcement.

The material properties used for the reinforcing bar are as follows.

$E = 30 \times 10^6$ psi
$v = 0.3$
$S_y = 30,000$ psi
Tangent Modulus, $E_T = 30 \times 10^4$ psi

Soil Material

The Lawrence Livermore National Labs report (Reference 20) and Brookhaven National Laboratory report (Reference 23) indicates that the stiffness of the soil has little impact on the peak accelerations predicted in the cask. Thus the same soil model was assumed as that used in the Livermore report. The soil material properties assumed for the analysis are:

$E = 6,000$ psi

$\nu = 0.45$
$\rho = 2.0368 \times 10^{-4}$ lb-sec^2 / in^4

A4A.10.3 BOUNDARY CONDITIONS

Only ½ of the cask was modeled with symmetry boundary conditions used to simulate the full structure. Non-reflecting boundaries were applied to the bottom and sides of the modeled soil not aligned with the plane of symmetry (bottom, left side, right side, and back) to prevent artificial stress waves from reflecting back into the model. Both dilatation and shear waves were damped as described in the LS-DYNA *BOUNDARY command.

An automatic surface to surface (contact_automatic_single_surface) contact definition was applied between all parts except the soil. The contact definition has a 0.5 penalty stiffness scale factor to prevent excessive contact stiffness leading to unrealistic part accelerations. A surface to surface (contact_surface_to_surface) contact definition was applied between the concrete and the soil with soft contact option 2. Soft contact option 2 was necessary between the soil and concrete as the materials have very different material stiffness. A conservatively low coefficient of friction (static and kinetic) of 0.25 was applied between all contact surfaces. It is conservative to use a low value for the coefficient of friction because less energy is absorbed due to friction resulting in greater impact acceleration forces.

A4A.10.4 INITIAL CONDITIONS AND LOADING

The analysis begins with a 1" gap between the cask and concrete to allow for at least 5 ms of zero acceleration other than gravity. An initial velocity was applied to all parts of the cask model. The initial velocity was computed by equating potential and kinetic energies. Due to the initial 1" gap and gravitational acceleration, initial velocities were computed 1" shorter than the drop heights.

V = potential energy = mgh
T = kinetic energy = $\frac{1}{2}mv^2$

For an 18" Drop:
$mgh = \frac{1}{2}mv^2$
$\Rightarrow v = \sqrt{2gh} = \sqrt{2(386.4)(18-1)} =$ 114.62 in./sec.

A gravitational acceleration of 386.4 in/sec^2 was applied to the cask and basket model.

A4A.10.5 RESULTS OF LS-DYNA ANALYSES

The resulting rigid body acceleration time histories were computed by LS-DYNA. The rigid body accelerations were computed for the bottom plate, circumferential shell, and basket representation. The parts can be seen in Figure A4A.10-3.

The peak filtered accelerations and corresponding time history plot for different parts of the TN-40HT cask 18" end drop are listed below. All results were filtered with a 4th order low pass butterworth filter with a 350Hz cutoff frequency.

Results Summary

Part	Peak Acceleration (g)	Time History Figure Number
Shell	41.5	A4A.10-4
Bottom Plate	44.1	A4A.10-5
Basket Representation	28.8	A4A.10-6

Based on the Results shown in the above table, the maximum acceleration in the TN-40HT cask during the 18 inch accident condition end drop event is 44.1g and occurs in the bottom plate. Also from this table, the highest acceleration in the basket and fuel is 28.8g. However, since the basket and fuel were not modeled explicitly, the maximum acceleration (28.8g) must be multiplied by the appropriate dynamic load factor (DLF). The maximum DLF for a triangular load is 1.52 (Reference 24). This results in a maximum loading of 43.8g.

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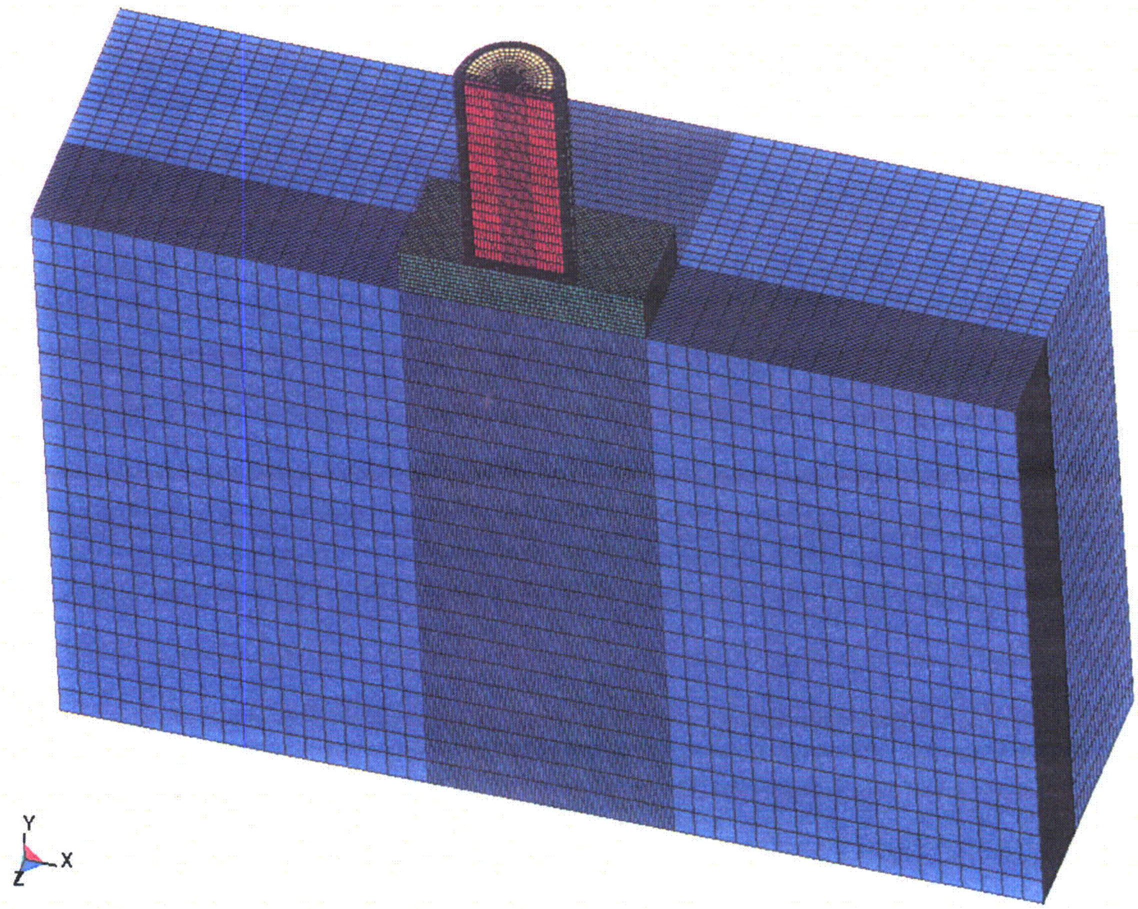

FIGURE A4A.10-1
OVERVIEW OF FINITE ELEMENT MODEL

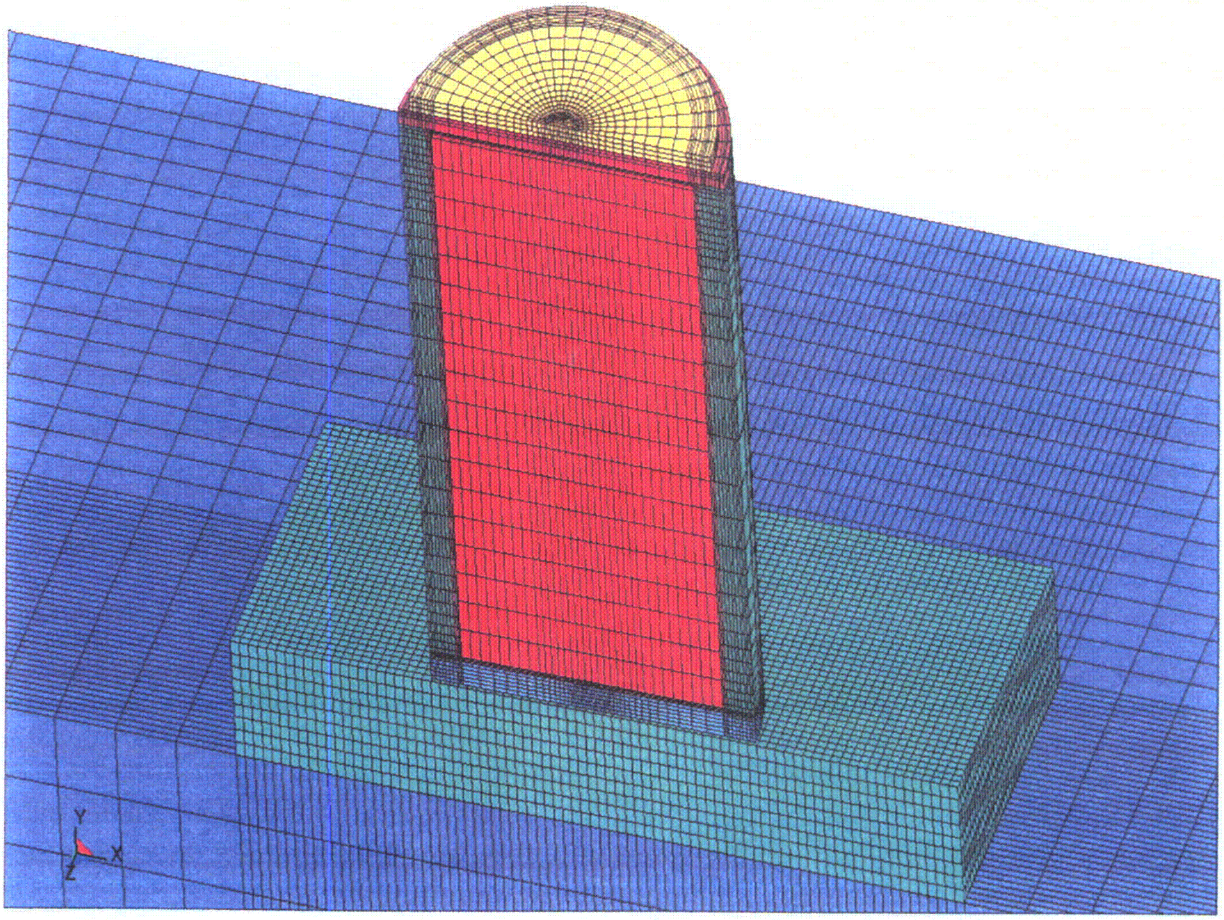

FIGURE A4A.10-2
OVERVIEW OF TN-40HT CASK FINITE ELEMENT MODEL

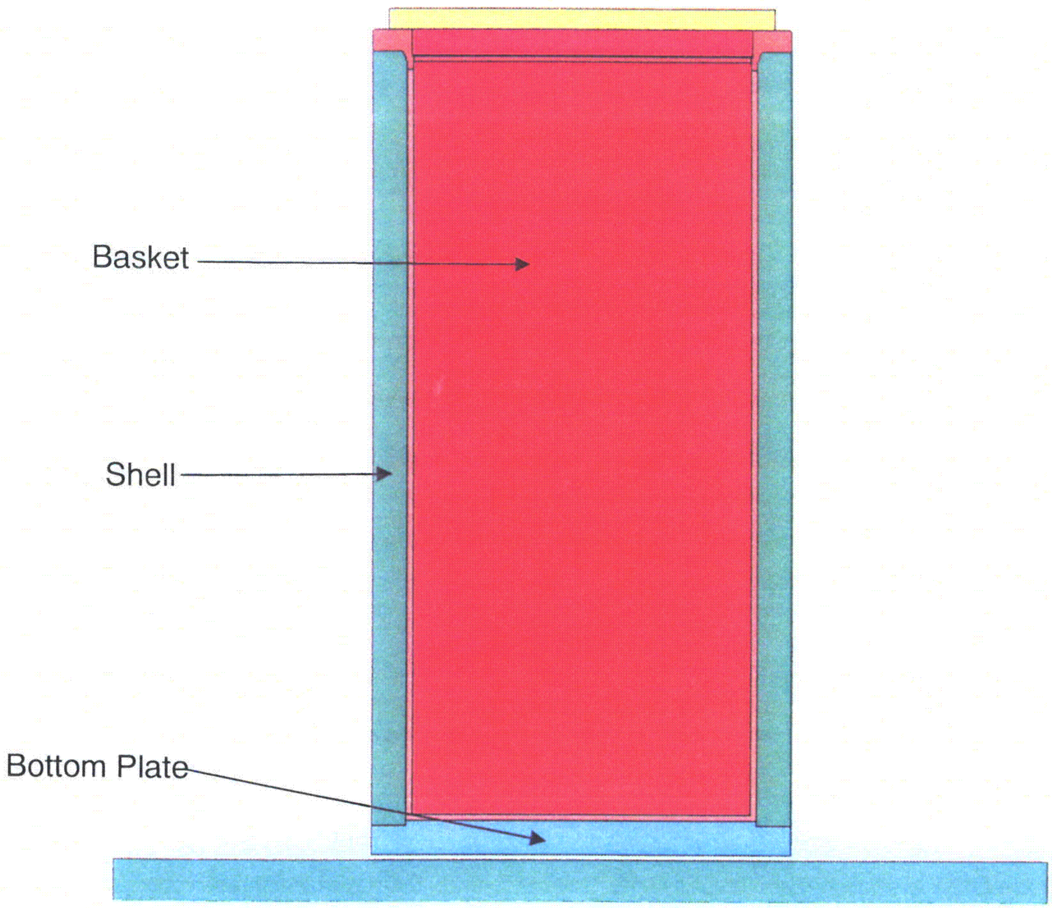

FIGURE A4A.10-3
PARTS ANALYZED FOR ACCELERATION TIME HISTORY

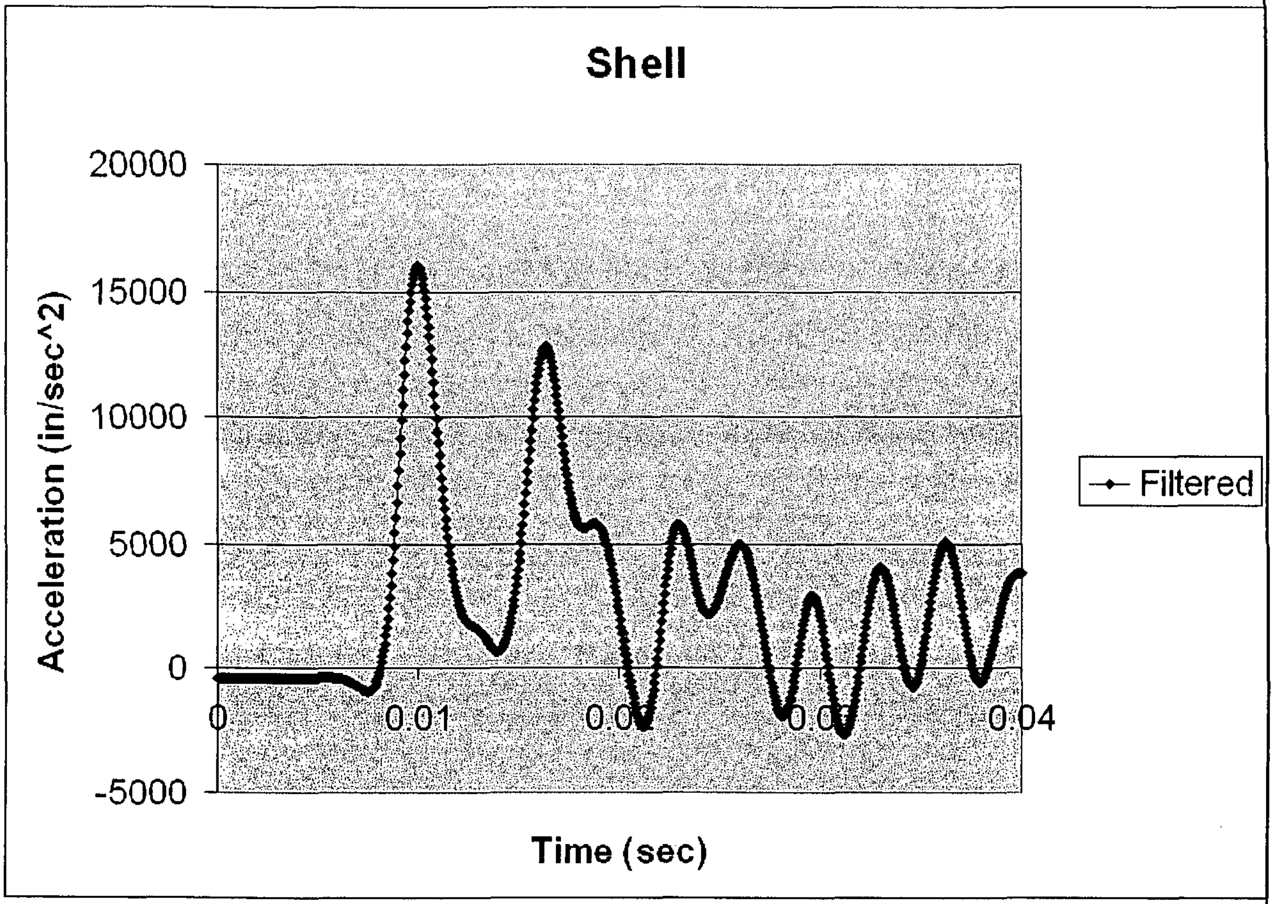

FIGURE A4A.10-4
CASK SHELL ACCELERATION TIME HISTORY (350HZ FILTER)

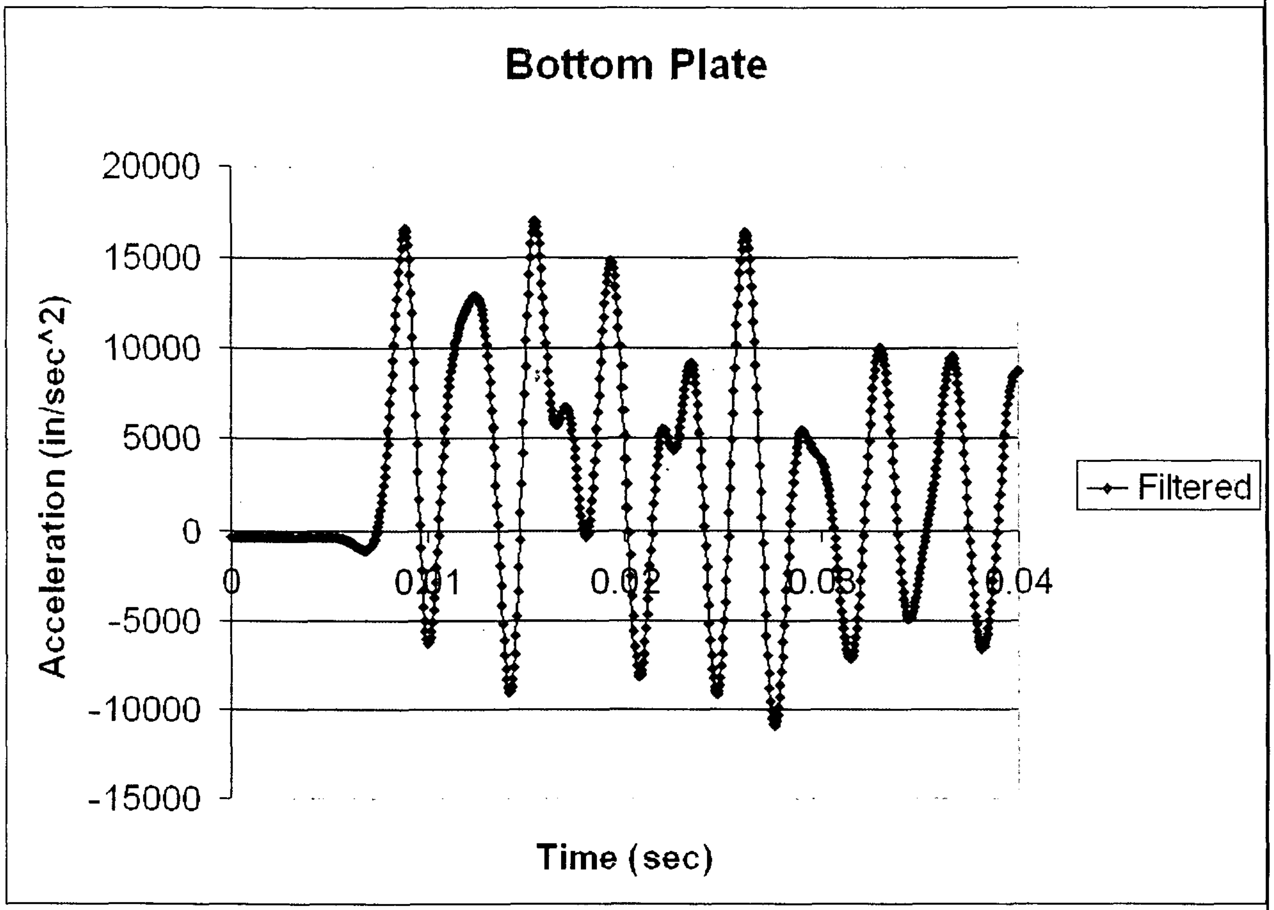

FIGURE A4A.10-5
CASK BOTTOM PLATE ACCELERATION TIME HISTORY (350HZ FILTER)

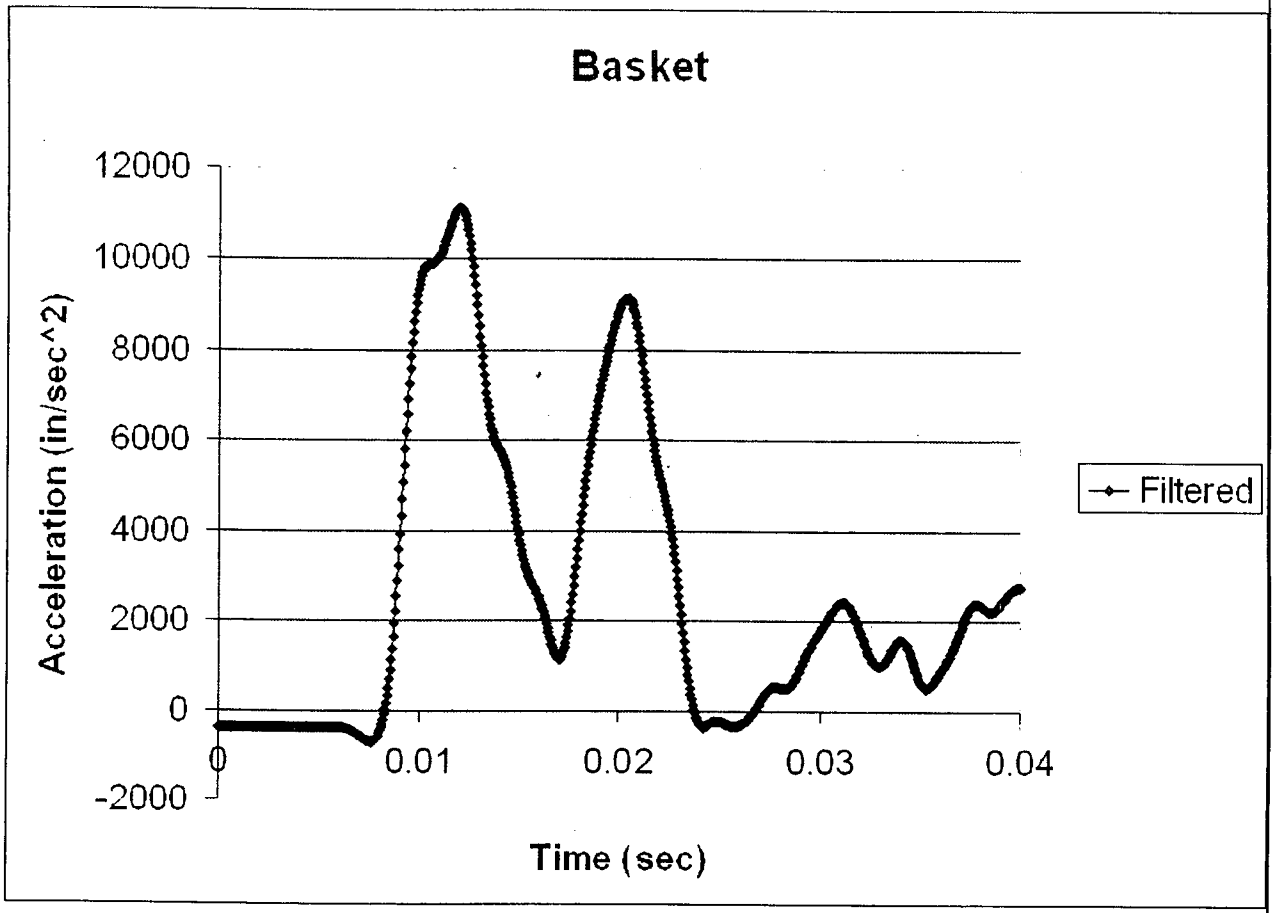

FIGURE A4A.10-6
CASK BASKET ACCELERATION TIME HISTORY (350HZ FILTER)

A4B.1.5 TN-40HT FUEL BASKET STRESS ANALYSIS

A4B.1.5.1 APPROACH

Bounding inertial loads of 3g vertical plus 3g lateral for the normal conditions and 50g vertical bottom end drop for accident conditions are applied to the basket. These inertial loads bound all applicable basket loads described in Section A3. 0°, 30°, 45°, 60° and 90° azimuth orientations are analyzed in order to bound all possible lateral loads (Figure A4B.1-2). Nonlinear (gap element) elastic analyses of the basket structure were performed using ANSYS (Reference 1).

A4B.1.5.2 BASKET ANALYSIS FOR VERTICAL AND LATERAL INERTIAL LOADS

A4B.1.5.2.1 FINITE ELEMENT MODEL DESCRIPTION

A three-dimensional finite element model of the basket is constructed using shell elements. The overall finite element model of the fuel basket is shown in Figure A4B.1-3. The fuel compartments and transition rails are included in the model. For conservatism, the strengths of aluminum plates and poison plates in the basket are neglected by excluding them from the finite element model. However, their weights are accounted for by increasing the structural steel plate material densities to 0.39 lbs/in^3.

Because of the large number of plates in the basket and large size of the basket, certain modeling approximations were necessary. In view of continuous support of fuel compartment tubes by the transition rails along the entire basket length during storage condition lateral loads, only a 15.0 inch long axial section of the basket and transition rail is modeled. At the two cut faces of the model, symmetry boundary conditions are applied.

The fusion welds, connecting the fuel compartments and plates, are modeled with 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. For the transition rails and cask interface the gap is varied in the circumferential direction such that it is zero at the point of contact, which depends on the orientation analyzed, and maximum 180 degrees from the point of contact.

The boundary conditions and interfaces for a typical fuel compartment are shown in Figure A4B.1-21 and Figure A4B.1-22.

A4B.1.5.2.2 Material Properties and Allowable Stresses

The stainless steel fuel compartment and transition rails are constructed from SA-240, Type 304 stainless steel. Table A4B.1-2 lists the material properties used in all analyses of the TN-40HT basket. Table A4B.1-3 and Table A4B.1-4 summarize the allowable stress for normal and accident conditions, respectively. Note that the transition rail allowable stresses are based on the allowable stress for surface PT weld (0.65 x S).

A4B.1.5.2.3 Vertical and Lateral Inertial Loads

The basket structure is analyzed for 0°, 30°, 45°, 60° and 90° azimuth lateral loads. Due to the basket structure symmetry, these orientations are assumed to envelop all other possible loading orientations.

A uniform fuel weight distribution is assumed over 144 inches, which is the active fuel length. A 15.0 inch section of the basket assembly is modeled. The weight of the aluminum plates and poison plates are accounted for by increasing the density of the steel plates. The aluminum plate stiffness and poison plate stiffness are conservatively neglected in the analysis.

The basket temperature is taken as 650 °F, uniform. The rail temperature is taken as 500 °F, uniform. These temperatures are conservatively taken from the normal condition (100 °F) thermal analysis presented in Section A3.

The load resulting from the fuel assembly weight is applied as pressure on the fuel compartment plates. For the 0° orientation, the pressure acts only on the horizontal plates, and for the 90° orientation, the pressure acts only on the vertical plates. For the 30°, 45° and 60° orientation, the pressure was divided into components that act on both horizontal and vertical plates of the fuel compartments. The pressures for all orientations are calculated below for vertical and horizontal plates due to 1g and 3g lateral acceleration.

0° and 90° Drop Orientations

Pressure for 1g p = Fuel assembly weight / (Panel span × Panel length)
= 1300 lb./ (8.2375" x 144") = 1.096 psi
Pressure for 3g = 3 x 1.096 = 3.288 psi

30° Orientation

P_h on vertical plates for 1g = p sin 30° = 1.096 x 0.5 = 0.548 psi
P_v on horizontal plates for 1g = p cos 30° = 1.096 x 0.86603 = 0.949 psi
P_h on vertical plates for 3g = 3 x 0.548 = 1.644 psi
P_v on horizontal plates for 3g = 3 x 0.949 = 2.847 psi

45° Orientation

P_h on vertical plates for 1g = p sin 45° = 1.096 x 0.7071 = 0.775 psi
P_v on horizontal plates for 1g = p cos 45° = 1.096 x 0.7071 = 0.775 psi
P_h on vertical plates for 3g = 3 x 0.775 = 2.325 psi
P_v on horizontal plates for 3g = 3 x 0.775 = 2.325 psi

60° Orientation

P_h on vertical plates for 1g = p sin 60° = 1.096 x 0.86603 = 0.949 psi
P_v on horizontal plates for 1g = p cos 60° = 1.096 x 0.5 = 0.548 psi
P_h on vertical plates for 3g = 3 x 0.949 = 2.847 psi
P_v on horizontal plates for 3g = 3 x 0.548 = 1.644 psi

An increased axial acceleration is applied to the 15 inch section to simulate the compressive load due to the weight of the complete basket. The acceleration is (including the 3gs),

3.0 x (160/15) = 32 g

Note: This simplified approach yields the correct vertical reaction force, but yields a compressive stress in the plates varying from zero at the top-most elements to the full 0.39 ksi at the bottom-most elements. (See the end drop analysis in Section A4B.1.5.5. Multiplying the 1g stress of 0.13216 ksi by 3 gives a 3g stress of 0.39 ksi). In reality, at the lower extremities of the basket, the entire 15" modeled portion of the basket would have 0.39 ksi direct compressive stress.

The load distributions for the 0°, 30°, 45°, 60° and 90° analyses for the normal condition loads are shown in Figure A4B.1-8 through Figure A4B.1-12, respectively.

The accelerations applied in each run are as follows.

Orientation	Inertial Load (g)	a_x (g)	a_y (g)	a_z (g) (simulate 3g axial load)
0°	3g vert. & 3g lat.	0	3	32
30°	3g vert. & 3g lat.	-1.5	2.598	32
45°	3g vert. & 3g lat.	-2.121	2.121	32
60°	3g vert. & 3g lat.	-2.598	1.5	32
90°	3g vert. & 3g lat.	-3	0	32

A4B.1.5.2.4 ANSYS 3g Vertical & 3g Lateral Analyses and Results

Nonlinear (gap element) elastic analyses of the basket structure were performed using ANSYS for the 0°, 30°, 45°, 60° and 90° vertical and lateral load orientations.

The nodal stress intensity distribution in the stainless steel fuel compartments and transition rails are computed by ANSYS. The membrane plus bending stress intensity distributions for the 45° loading condition are shown in Figure A4B.1-13 and Figure A4B.1-14 as representative sample of the resulting stresses. The shell middle surface nodal stress intensity is the membrane stress intensity and the top or bottom surface stress intensity is the membrane plus bending stress intensity. The maximum membrane and membrane plus bending stresses for each load orientation are summarized in Table A4B.1-5.

A4B.1.5.3 BASKET ANALYSIS FOR THERMAL LOADS

A4B.1.5.3.1 Finite Element Model Description

Thermal Stress Model for Basket Fuel Compartments

A three-dimensional finite element model of the basket Fuel Compartments is constructed using shell elements. Due to symmetry, only ¼ of the model (see Figure A4B.1-15) is used in this analysis.

Thermal Stresses for Transition Rails

A three-dimensional finite element model of the transition rails is constructed using shell elements. Due to symmetry, only ¼ of the model (see Figure A4B.1-16) is used in this analysis.

A4B.1.5.3.2 Thermal Loads

The transition rails are attached to the basket with bolts in slotted holes, thus permitting free thermal growth of basket boxes. However, some thermal stresses in basket and rails can develop due to radial gradients (hot at center and cooler at periphery) for normal thermal conditions. Basket and Rail thermal stresses are calculated for the 100 °F (hot normal) and -40 °F (cold normal) ambient.

Elastic material properties described in Section A4B.1.5.2 are used, and conservative temperature gradients are applied (see paragraphs below).

Nodal Temperature for Basket Fuel Compartments

Analyses documented in Section A3 were used to obtain temperatures along a radial line from the basket center to the basket perimeter that give the largest radial thermal gradient. These temperatures were used to develop bounding polynomial curve-fit equations that give temperatures as a function of radial location. The temperature gradients were conservatively modified by increasing the temperature in the middle of the basket by 50 °F and decreasing the temperature at the perimeter of the basket by 50 °F for normal and off-normal thermal storage conditions. Temperatures between the middle of the basket and the perimeter of the basket were modified proportionally to maintain the same basic shape of the temperature distribution. For a given bounding curve, the conservative temperature gradient is mapped onto the basket models in all radial directions to give the largest gradient across the entire diameter of the basket. The temperature distribution in the fuel compartments for 100 °F and -40 °F are shown in Figure A4B.1-4 and Figure A4B.1-5, respectively.

Nodal Temperature for Transition Rails

The same conservative temperature gradient for fuel compartments is mapped onto the rail models in all radial directions to give the largest gradient across the entire diameter of the basket. The temperature distribution in the transition rails for 100 °F and -40 °F ambient conditions are shown in Figure A4B.1-6 and Figure A4B.1-7, respectively.

A4B.1.5.3.3 ANSYS Thermal Analysis and Results

Basket Compartment

Nodal stress intensity distributions in basket fuel compartments and the support plates are plotted at top or bottom surfaces in Figure A4B.1-17 and Figure A4B.1-18 for 100 °F and -40 °F ambient conditions, respectively.

The maximum stress intensities in the fuel compartments are 8.61 ksi and 8.74 ksi for 100 °F and -40 °F ambient conditions, respectively, and are shown in Table A4B.1-6.

Transition Rail

Nodal stress intensity distributions in transition rails are plotted at top or bottom surfaces in Figure A4B.1-19 and Figure A4B.1-20 for 100 °F and -40 °F ambient conditions, respectively.

The maximum stress intensities in the transition rails are 21.73 ksi and 21.60 ksi for 100 °F and -40 °F ambient conditions, respectively, and are shown in Table A4B.1-6.

A4B.1.5.4 SHEAR LOAD IN THE FUSION WELD

The results of the static load analyses were post-processed for Axial (FX) and Shear (FY and FZ) forces in the pipe elements representing the Fusion Welds. The maximum forces are listed as follow:

Maximum Force, FX = 314.92 lb. (at Element No. 32728, 60^o lateral load orientation)

Maximum Force, FY = 950.88 lb. (at Element No. 32723, 90^o lateral load orientation)

Maximum Force, FZ = 99.27 lb. (at Element No. 32553, 60^o lateral load orientation)

The results of the thermal load analyses were also post-processed for Axial (FX) and Shear (FY and FZ) forces in the pipe elements representing the Fusion Welds. The maximum forces are listed as follow:

Maximum Force, FX = 310.69 lb. (at Element No. 32559, 100 °F ambient)
Maximum Force, FY = 608.83 lb. (at Element No. 32548, -40 °F ambient)
Maximum Force, FZ = 99.01 lb. (at Element No. 32717, -40 °F ambient)

The maximum combined shear load in a fusion weld is computed by vectorially adding the maximum FX, FY, and FZ (irrespective of their location).

Maximum Combined Shear Force = $[(314.92 + 310.69)^2 + (950.88 + 608.83)^2 + (99.27 + 99.01)^2]^{1/2}$ = 1,683 lb = 1.68 kips (per fusion weld)

For the fusion weld load capacity test at room temperature, a factor of safety of 1.43 is applied and the material strength is corrected for room temperature testing. Therefore the, the required minimum fusion weld test load per weld is:

= 1,683 * (Factor of Safety) * (S_u at room temperature / S_u at 650^oF)

=1,683 lb * 1.43 * (75 ksi/63.4 ksi) = 2,847 lb.
≈ 2.9 kips

TABLE A4B.1-1
SUMMARY OF INDIVIDUAL LOADS FOR STORAGE CONDITIONS-BASKET

Individual Load	Loads
IL-1	50g Vertical (Bottom End Drop)
IL-2	3g Vertical (Lifting)
IL-3	3g Vertical + 3g Lateral (bounds all normal and off normal loads)
IL-4	Thermal Stress due to Hot Environment (100°F ambient)
IL-5	Thermal Stress due to Cold Environment (-40°F ambient)

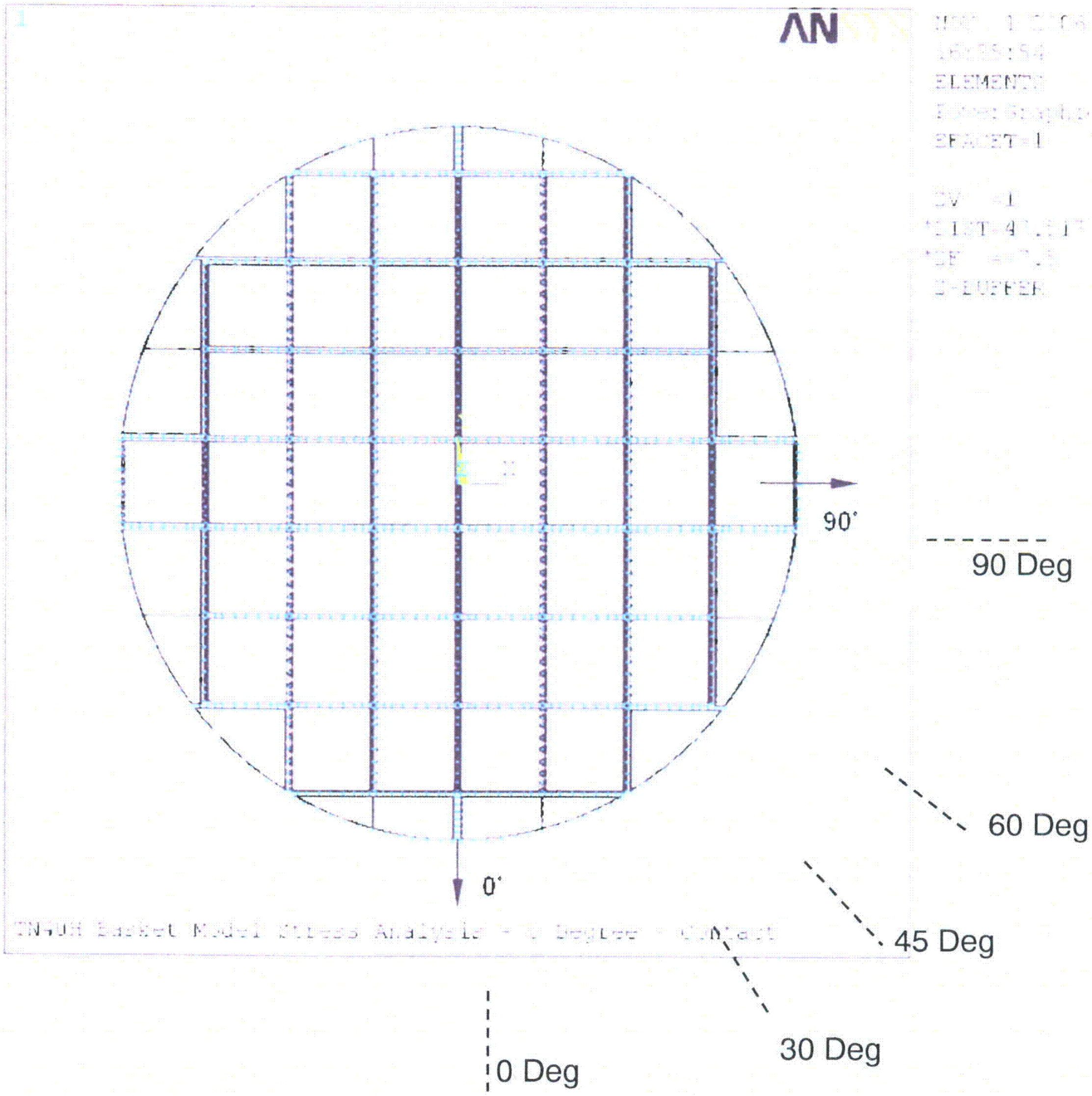

FIGURE A4B.1-2
TN-40HT BASKET LOADING ORIENTATION DEFINITIONS

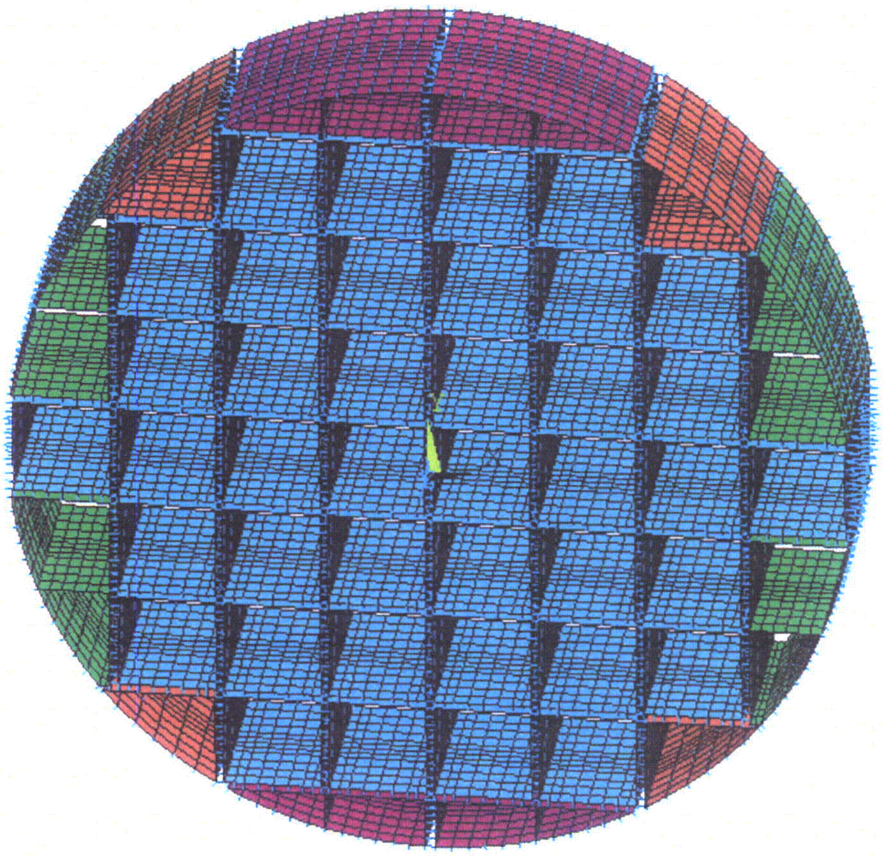

FIGURE A4B.1-3
BASKET FINITE ELEMENT MODEL

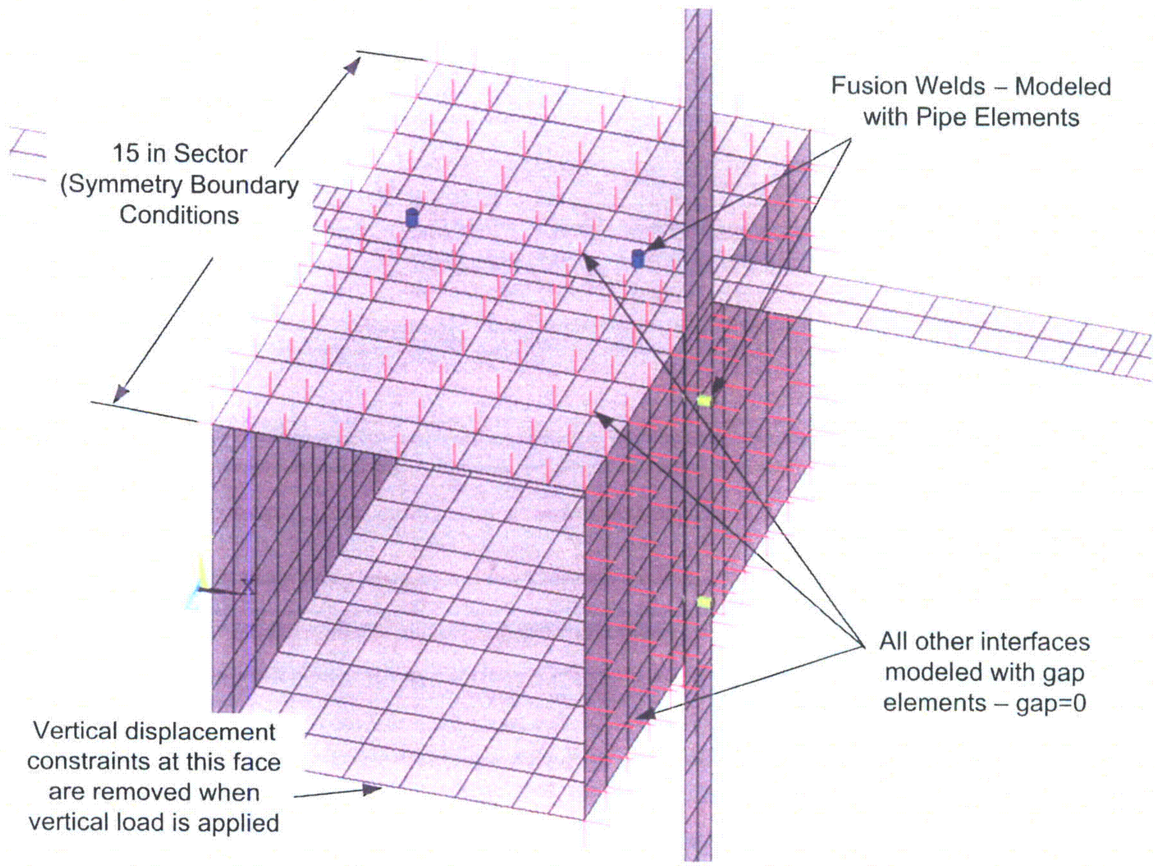

FIGURE A4B.1-21
INTERFACES FOR TYPICAL FUEL COMPARTMENT TUBE

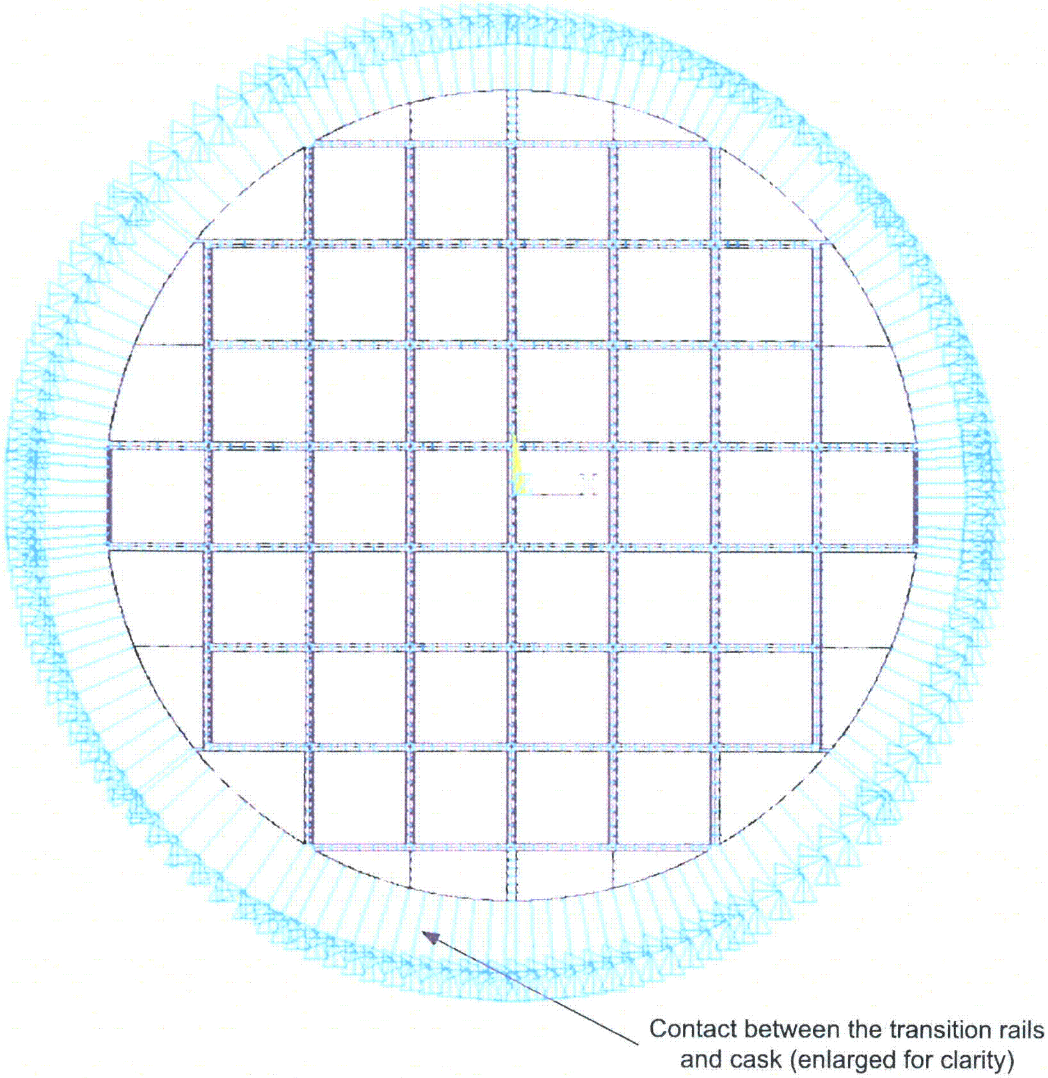

FIGURE A4B.1-22
BOUNDARY CONDITIONS FOR THE BASKET ANALYSIS
(SYMMETRY BOUNDARY CONDITIONS ARE NOT SHOWN FOR CLARITY)

Table A7.2-4 contains the masses of the various hardware materials present in the four principal regions of the fuel assembly. The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by the activation ratios 0.1, 0.2 and 0.2, respectively (Reference 5), to correct for the spatial and spectral changes of the neutron flux outside of the active fuel zone.

As an example, the effective cobalt mass in the top end region is determined by taking the mass of the material listed in Table A7.2-4 times the cobalt impurity percentage listed above, which are also shown on Table A7.2-3. The resultant product is then adjusted by the above activation ratio:

Co = 6.30 kg (stainless steel, Table A7.2-4)*0.08% (Table A7.2-3) +
0.508 kg (Inconel Table A7.2-4)* 0.469% (Table A7.2-3) = 0.0074 kg
effective Co (after applying the 0.1 activation ratio) = 0.00074 kg

The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis. The cobalt content for each fuel assembly region utilized in the source term calculation is obtained from Table A7.2-4 reduced by the activation ratios.

Fuel Insert Thimble Plug Device (TPD)

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.

Fuel Insert Burnable Poison Rod Assembly (BPRA)

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.

Reactor Coolant System Boron Concentration

Soluble boron in the reactor coolant system is the primary reactivity control mechanism in a PWR. It is important to include boron in the depletion model because it is a thermal neutron absorber and its presence hardens the neutron spectrum. The presence of boron reduces thermal absorption in U-235 and as a second-order effect, increases epithermal absorption in U-238, increasing the buildup of actinides such as Cm-244, the major contributor to the neutron source term.

Typical cycle average boron concentration is on the order of 900 ppm. 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. 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 under predictions are within 1% of those determined with the lower boron concentration and thus have essentially no effect on dose rates and cooling times.

The results of the fuel qualification sensitivity calculations with soluble boron concentration are shown in Table A7.2-12. Cases C1 through C8 are sensitivity calculations where the soluble boron concentration in Cases B1 through B7 on Table A7.2-11 was 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, i.e. Case A7 on Table A7.2-11, 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.

Reactor Coolant System Temperature

Moderator temperatures can vary between 500-600 °F. The long-term operating temperature affects the end-of-life reactivity (as a second order effect) and the total actinide inventory. Higher average moderator T_{av} reduces the average moderator density, which reduces neutron moderation during operation. This again 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. A moderator density of 0.733 g/cc (which corresponds to 566°F at an operating pressure of 2250 psia) is used in the SAS2H calculation.

The results of the fuel qualification sensitivity calculations with moderator temperature are shown in Table A7.2-12. Cases D1 through D8 are sensitivity calculations where the moderator temperature in Cases C1 through C7 on Table A7.2-12 was increased from 558 K (545°F, representative of a core average moderator temperature) to 590 K (602°F, representative of an average hot leg moderator temperature) and the moderator density was 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 and boron concentration results in an increase (when compared to corresponding cases B1 through B7) 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, i.e. Case A7 on Table A7.2-11, 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 566°F) for the design basis is justified because 566°F is representative of a core average moderator temperature.

Fuel and Cladding Temperature

Representative temperatures of 840 K and 620 K were selected for the fuel and clad, respectively.

A7.2.2 AIRBORNE RADIOACTIVE SOURCES

In addition to the source terms generated for the shielding analysis, the SAS2H irradiation also provides the bounding radiological source terms for confinement. These results are provided in Table A7.2-6.

A7.2.3 AXIAL SOURCE DISTRIBUTION

Reference 4 provides axial PWR burnup profiles as a function of fuel assembly burnup ranges. For the Prairie Island high burnup fuel to be stored in the TN-40HT cask, the burnup profile for > 46 GWD/MTU was selected for use in this analysis. Figure A7.2-1 represents this profile.

The conservative axial profile containing 18 axial zones is utilized in the shielding evaluation. The axial zones are approximately equal and each zone represents between 5% and 6% of the total active fuel zone. The peaking factors range from a slightly more than 0.5 at the bottom and top, to a maximum of 1.11 just below the middle. The gamma source is directly proportional to the burnup and the neutron source is proportional to the fourth power of the burnup. This data is presented in Table A7.2-7 and shown in Figure A7.2-2. For the gamma distribution, the average value is around 1.0. However, for the neutron distribution, the average value of the distribution is greater than 1.0. The average value of the axial neutron distribution may be interpreted as the ratio of the true total neutron source in an assembly to the neutron source calculated by SAS2H/ORIGEN-S for an average assembly burnup. Therefore, to properly correct the magnitude of the neutron source, the neutron source per assembly, reported in Table A7.2-8, is multiplied by the average value of the neutron source distribution as reported in Table A7.2-7.

A7.2.4 GAMMA SOURCE

The gamma source terms for the 14x14 design basis fuel assembly are provided in Table A7.2-8. The gamma source spectra are presented in the 18-group structure consistent with the SCALE 27n-18γ cross section library. The conversion of the source spectra from the default ORIGEN-S energy grouping to the SCALE 27n-18γ energy grouping is performed directly through the ORIGEN-S code. The SAS2H/ORIGEN-S input file is provided in Appendix A7B.

The gamma source for the fuel assembly hardware is primarily from the activation of cobalt. This activation contributes primarily to SCALE Energy Groups 36 and 37.

For this analysis, it is assumed that the fuel insert source is a composite source; a BPRA in the core region and a TPD in the plenum and top fitting region. This bounds both types since the TPD does not extend into the core region but has a higher source term in the plenum and top fitting regions than a BPRA. The cumulative burnup of the fuel assembly(s) where the BPRA resided during reactor operation is assumed to be 30 GWd/MTU and the BPRA has cooled for 18 years. The cumulative burnup of the TPD host assemblies is assumed to be 125 GWD/MTU and the TPD has cooled for 16 years. The fuel insert source is shown in Table A7.2-9.

A7.2.5 NEUTRON SOURCE

Table A7.2-8 provides the total neutron source for the 14x14 design basis fuel assembly. The neutron source term consists primarily of spontaneous fission neutrons (largely from Cm-244) with (α,O-18) sources of lesser importance, both causing secondary fission neutrons. The overall spectrum is well represented by the Cm-244 fission spectrum. For the MCNP analyses, the default Cm-244 energy spectrum is utilized.

A7.2.6 FUEL QUALIFICATION

This section provides the basis for qualification of the design basis fuel chosen for the shielding analysis of the TN-40HT cask. The objective is to demonstrate that the fuel assemblies with the parameters corresponding to design basis fuel result in the highest calculated dose rate so that bounding shielding analysis can be performed by utilizing these design basis source terms. In order to determine the bounding spent fuel parameters (design basis fuel assembly), the candidate assembly parameters are ranked by their relative radiation source strengths. A simple 2-D shielding calculation based on a representation of the TN-40HT cask is performed and the radiation dose rates at 2m from the side surface are determined. The spent fuel parameters that yield the highest total dose rate (gamma + neutron) are considered the design basis for shielding calculations.

These analyses were carried out using the SAS2H depletion module from the SCALE (Reference 1) computer software and MCNP (Reference 2). For all SAS2H calculations the SCALE 44 group ENDF/B-V (44groupndf5) library was used. MCNP calculations used the default cross section libraries.

A simplified MCNP model was utilized to calculate a response function at 2 meters from the side surface of a cask. The response function is simply a source-to-dose conversion function and is representative of the TN-40HT cask shielding configuration. A representative response function is adequate for ranking the impact of burnup, enrichment and cooling time combination on dose rate. In essence the dose rate at 2 meters is calculated for each gamma energy group of the source as calculated by SAS2H. For neutrons, since a bounding energy spectrum is used, the response function calculated is just a total source to dose factor.

The response function is then utilized with SAS2H models to estimate the 2 meter side dose rate as a function of different burnup, enrichment and cooling time combinations. Other information, such as the decay heat per fuel assembly and the cooling time are also collected.

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 and Table A7.2-12. 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 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 B9 and B10 represent BECT combinations at enrichments of 2.1 wt. % U-235 and 1.0 wt. % U-235. Their results are also bounded by A7.

The results of the sensitivity calculations – "C" and "D" cases are shown in Table A7.2-12. 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 (545°F) to 590 K (602°F, 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.

The calculated dose rate and decay heat along with the cooling time are then utilized to determine the bounding radiological source term. The final design basis radiological source term is generated by adding the fuel insert source term to the fuel/hardware source term.

The evaluation with the response function showed the design basis fuel assembly with parameters of 3.4 %wt U-235, 60 GWD/MTU burnup and cooled for 18 years to provide the appropriate bounding source terms. It is important to note that the decay heat for this bounding assembly is 845 watts which is in excess of the allowable limit as discussed in Section A3.

A7.2.7 RECONSTITUTED FUEL ASSEMBLIES

Reconstituted fuel assemblies are fuel assemblies that have replaced damaged fuel pins with either natural uranium dioxide 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 dioxide 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 dioxide 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, the 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).

Table A7.4-3 provides a summary of staffing levels assumed at various site locations along with the distance from the center of the ISFSI. The locations designated on Table A7.4-3 are shown on Figure A7.4-A. This information is considered to be a general estimate of station staffing and does not necessarily reflect actual staffing at any given time. Changes in staffing levels and locations can be expected to occur in the future without affecting the general estimates contained in this section.

Distance dependent dose rates are provided in Table A7A.7-2. The data is based on loading 48 TN-40HT casks over a 22-year period assuming 4 spent fuel casks are loaded every 2 years. The dose rates in Table A7.4-3 were conservatively (using linear interpolation) calculated using the dose rate vs. distance data corresponding to the "corner" detectors.

Table A7.4-4 summarizes the calculated total doses to full time and outage help at the various locations due to ISFSI operation. The collective onsite dose by location and the total onsite dose estimates are also shown on this table.

TABLE A7.2-1
PRAIRIE ISLAND FUEL ASSEMBLY DESIGN CHARACTERISTICS

Fuel Designations	Exxon Std (14x14)	Exxon Std High Burnup (14x14)	Exxon Toprod (14x14)	Westinghouse Standard (14x14)	Westinghouse OFA (including Vantage+) (14x14)
Max Length (in)	161.3	161.3	161.3	161.3	161.3
Max Width (in)	7.763	7.763	7.763	7.763	7.763
Fuel Density(1) (% Theoretical)	93.18	93.20	92.80	93.32	94.80
Rod Pitch (in)	0.556	0.556	0.556	0.556	0.556
No of Fueled Rods	179	179	179	179	179
Maximum Active Fuel Length (in)	144.0	144.0	144.0	144.0	144.0
Fuel Rod OD (in)	0.4240	0.4260	0.4170	0.4220	0.4000
Clad Thickness (in)	0.030	0.031	0.0295	0.0243	0.0243
Fuel Pellet OD (in)	0.3565	0.3565	0.3505	0.3659	0.3444
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4/ZIRLO
Guide Tube OD (in)	0.541	0.541	0.541	0.539	0.528
Guide Tube Wall Thickness (Zr-4) (in)	0.017	0.017	0.017	0.017	0.019
Guide Tube #	16	16	16	16	16
Instrument Tube #	1	1	1	1	1
Instr. Tube OD (in)	0.424	0.424	0.424	0.422	0.4015
Instr. Tube Wall Thickness (Zr-4) (in)	0.025	0.025	0.025	0.0243	0.0258
Maximum MTU/assembly(2)	0.370	0.370	0.370	0.410	0.360
Free Gas Volume m^3 @STP	0.226	0.226	0.226	0.107	0.107
Fill Gas	He	He	He	He	He

Notes:

(1) The fuel density values are "as-built", i.e., they incorporate the effect of the pellet dish and chamfer, as well as the theoretical density of the pellet. These densities were calculated from reload data provided by the fuel manufacturer, as follows:

As-built UO_2 density = total amount UO_2 in reload/ total volume of fuel in reload

Where fuel volume = (# fuel pins) (π/4) (pellet OD)^2 (active fuel length)

TABLE A7.2-10
RESPONSE FUNCTION FOR TN-40 HT CASK

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

Primary Gamma Energy Range (MeV)	Response Function ((mrem/hour) per particle) per cask
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 (1)	10.10	9.10	19.20
Ratio	0.75	0.80	0.77

(1) The neutron, gamma and total dose rates are obtained as an average of the dose rates shown in Table A7A.5-2. The dose rates at axial height ranging from -22.9 cm to 27.8 cm are included in the average calculations.

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
B9	44	2.1	12.0	687	10.03	14.39	24.42
B10	19	1.0	12.0	272	1.25	7.30	8.55

TABLE A7.2-12
FUEL QUALIFICATION SENSITIVITY 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
	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

A7A.4 MODEL SPECIFICATION

The neutron and gamma dose rates on the surface of the TN-40HT cask (Side, Top and Bottom surfaces) and at 1m and 2m from the side surface of the cask are evaluated with the 3-D Monte Carlo transport code MCNP (Reference 1). The flux-to-dose conversion factors specified by ANSI/ANS 6.1.1-1977 (Reference 2) are used and provided in Table A7A.4-1.

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP code. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made to account for secondary gamma radiation (n,γ), if desired. All of the near field dose rate calculations account for the dose rate due to secondary gamma radiation. For the far field (long distance) dose rate calculation, the dose rate contribution from the secondary gamma radiation is ignored because it is insignificant.

Figure A7A.4-1 is a plot of the TN-40HT cask MCNP model and shows the axial cross section of the cask. This shielding configuration is utilized to determine the radial (side surface) and axial (top and bottom surface) dose rates around the TN-40HT cask for normal, off-normal and accident conditions. This configuration is also utilized to determine the dose rates at long distances from the cask.

The following are key assumptions used in the development of the MCNP models:

- 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.

A7A.4.1 MCNP MODEL FOR NORMAL AND OFF-NORMAL CONDITIONS

A single shielding configuration is utilized for the TN-40HT design for both normal and off-normal conditions of storage. The cask bottom dose rates are determined as a consequence of off-normal conditions since these become important only during loading and transfer. During normal conditions of storage the cask is upright and is firmly seated on the concrete pad. A three-dimensional MCNP model which includes a discrete fuel assembly model of the TN-40HT cask was developed for this purpose. In the MCNP model, the TN-40HT cask axis is modeled along the Z-direction. The X and Y axes in the MCNP model represent the cask in the radial direction. The cask is assumed to sit on a concrete pad (Z-direction).

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 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. Cells 2051 through 2133 represent the discrete basket and fuel assembly zones.

Figure A7A.4-1 is a Y-Z plot of the MCNP model of the TN-40HT cask (axial section view). All the major details of the cask model are shown in this figure. The fuel basket extends to 160" in the axial direction from the bottom of the bottom fitting (-190.70 cm) to about 2" above the top of the fuel assembly (209.91 cm). The active fuel zone is 144" long with the center at 0 (± 182.88 cm). 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 cask trunnions and the resin cutouts (flats) are modeled explicitly. The neutron resin boxes/radial neutron shield, top neutron shield and protective cover are also modeled explicitly.

The basket is modeled discretely using the advanced geometry features of MCNP. The fuel is modeled as a cuboid based on a 7.763" square. The fuel compartment inside dimension is 8.05" and is modeled with stainless steel with a thickness of 0.187" surrounded by a 0.437"-thick basket aluminum plates. The Boral© (or any other poison material) plates were modeled as pure aluminum and the stainless steel strips (tie plates) were not modeled. The stainless steel and aluminum peripheral rails were modeled explicitly. A small air gap of 0.15" (0.38 cm) was assumed between the basket and the cask shell. Figure A7A.4-2 is a radial cross section (X-Y plot) of the MCNP gamma model at Z=181 cm that shows the upper trunnions. Figure A7A.4-3 is also a radial cross section plot which shows one quarter of the basket and the rails.

Above the basket, the stainless steel fuel compartments and aluminum plates surrounding the fuel assembly are replaced by air (void).

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.

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.

Tallies are based on the F4 type mesh tally feature of MCNP to provide the average flux in the defined volume. The radial tallies are located just off the cask surface, 1 m from cask outer surface and 2 m from cask outer surface. Similarly, axial tallies are located at the cask top (protective cover) and bottom which determine the average flux at the top and bottom surfaces, and 1m and 2m from the top surface.

A7A.4.2 MCNP MODEL FOR ACCIDENT CONDITIONS

The MCNP design basis model for accident conditions is almost identical to that of the normal (and off-normal) conditions except that all neutron shielding and the outer steel shell materials are replaced with void. This scenario is based on a cask drop accident that results in the complete removal of the neutron shielding materials including the polypropylene disk and the protective cover at the top. This is implemented in MCNP by replacing the appropriate materials in the MCNP models with void.

A7A.4.3 MCNP MODEL FOR LONG DISTANCES FROM THE CASK

The near field dose rate MCNP model described above is modified to determine the long distance dose rates. The near-field MCNP model is extended further to include additional volumetric detectors beyond the immediate vicinity of the cask and splitting cells are used to transport particles far from the cask (about 2000 m from the cask surface) to obtain dose rates at long distances. A schematic of the far field model is shown in Figure A7A.4-4. This modeling technique splits particles upwards and outwards to better simulate skyshine. At farther regions of the model, cones are used to split particles downward to the detector location.

The MCNP calculated dose rates at far distances consist of contributions from direct, air scatter (skyshine) and limited ground scatter (only in the immediate vicinity of the cask). Modifications (cell flagging) are also made to separate the direct and skyshine component. Any radiation that crosses an elevation of 20 feet (~1.5 meter above cask) is considered to be skyshine. This is a valid approximation for two main reasons. One, this height corresponds to the earth berm. Second, beyond this elevation it is highly probable the radiation will have to scatter back in order to reach far detectors.

Soil is modeled as the ground surface under and extending away from the cask. It is expected that the maximum contribution to the ground scatter component occurs at the immediate vicinity of the cask. A layer of soil more than 4 feet thick is adequate to account for "ground shine." Actual soil depth modeled in MCNP models is about 6 feet. Axial splitting is not modeled in the soil.

The doses due to capture gamma sources are not calculated since they are insignificant at large distances in comparison to primary gamma and neutron sources.

The dose is calculated as F4 tallies (that calculate the volumetric dose rates) in an annular cylindrical detector, 20 cm thick and 3 feet high at about 6 feet off the ground. The dose rates are calculated at distances ranging from 10m to 2000m from the edge of the cask.

A7A.4.3 SHIELD REGIONAL DENSITIES

Table A7.2-4 shows the fuel assembly material composition for the four fuel assembly regions. Based on these material compositions and material densities, atom fractions for the fuel assembly regions are determined and provided in Table A7A.4-2. The mass of materials in each fuel assembly region is homogenized over the volume of the region (area = 60.26 in^2, based on a 7.763" x 7.763" cross-section). Table A7A.4-3 provide the shield regional densities for the TN-40HT cask. The actual fuel layout in the TN-40HT cask is an array of fuel assemblies inside stainless steel compartments surrounded by sheets of aluminum material.

The radial resin and aluminum boxes are discreetly modeled based on the material composition of the polyester resin and the aluminum.

Credit is taken for the ISFSI berm. The berm is modeled as an isosceles trapezoid with 14.02 and 2.44 meters bottom and top segments, respectively. The berm is assumed 6.25 meters height and made from soil. Based on the coordinate system depicted on Figure A7A.7-1 (the berm is not shown on the Figure), the planes of symmetry along the west and east sides of the berm are X=-114.30 and X=121.61 meters, respectively. Similarly, planes of symmetry along the north and south sides of the berm are at Y=-58.22 and Y=58.22 meters, respectively. -

Radiological sources from Bottom Nozzle, In-core, Plenum and Top Nozzle axial regions of the TN40 HT design basis fuel assembly are calculated for cooling times ranging from 18 to 40 years, with 2 years increments and shown in Table A7A.7-4. The SAS2H\ORIGEN-S models utilized in the design basis fuel source term calculations are also employed herein.

In order to simplify the source term specification in the MCNP models, the total gamma and neutron radiation source terms strength can be approximated with an exponential function as a function of decay time. 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 the SAS2H calculations. Decay constants for calculating neutron, in-core gamma (active fuel region), and fittings (top and bottom nozzle and plenum regions) gamma radiation source terms strength are calculated to be equal to 0.0358 years^{-1} (corresponds to ~19.4 years half-life), 0.025 years^{-1} (corresponds to ~27.7 years half-life) and 0.1315 years^{-1} (corresponds to ~5.27 years half-life) respectively. These decay constants are calculated in such a way that exponential function produces source terms that match, within 1%, the source terms calculated using SAS2H\ORIGEN-S models.

For ease of input specification in the MCNP models, the total gamma source strength (4.8937e+18 gammas per second including contribution from BPRAs and TPDs.) is calculated utilizing the exponential functions. Spectrum due to the design basis fuel assembly is used to conservatively describe the spectral distribution of gamma radiation source terms in all the casks where source is specified.

For the neutron source specification, the total neutron source strength calculated directly (1.9253e+12 neutrons per second, including sub-critical multiplication and axial source profile) with the SAS2H\ORIGEN-S models are utilized. Spectral distribution of neutron radiation source terms is due to Cm-244. MCNP provides built-in parameters to describe this distribution in the calculational models.

The assumptions for the MCNP methodology are summarized below.

- Due to symmetry of the cask loading configurations, the source specification is simplified and considered only for 12 of the 48 casks while the detectors are positioned to obtain the results as a result of this simplified source specification.

- The "universe" is a sphere surrounding the ISFSI. The radius of this sphere (r=5.0 kilometers) is more than 20 mean free paths for neutrons and more than 20 mean free paths for gammas in energy groups that contribute the most to the gamma radiation dose rates. This ensures that the model is of a sufficient size to include all interactions affecting the dose rate at the detector locations.

- "Detectors" located at distances less than or equal to 45 meters are within the ISFSI berm perimeter. Dose rates points at larger distances are behind the berm. It is likely that the dose rates beyond the berm perimeter are almost entirely due to skyshine component.

- Dose rates include contribution due to primary gamma and neutron sources as well as due to the secondary gamma radiation from (n,g) interactions.

A7A.7.2 MCNP ARRAY METHOD – DOSE RATE AS A FUNCTION OF DISTANCE

The MCNP results provide the dose rate as a function of distance at all the locations (including sides and corner) around the ISFSI. The total and skyshine dose rate results for the N/S sides, E/W sides and the corners are shown in Table A7A.7-2. The skyshine doses are shown in Table A7A.7-3 only for comparison. Due to presence of the ISFSI berm, it is expected that the dose rates at distances greater than 100m are dominated by the skyshine component and these results serve to demonstrate this assertion. Co-ordinate transformation is performed such that the results reported in Table A7A.7-2 are based on the distance from the center of the ISFSI.

A7A.8 CONFINEMENT

A7A.8.1 CONTAINMENT BOUNDARY

The containment boundary consists of the inner shell and bottom plate, shell flange, lid outer plate, lid bolts, penetration cover plate and bolts and the inner portions of the lid seal and the two lid penetrations (vent and drain). The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. Helium assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation.

A7A.8.2 SEALS AND WELDS

The containment boundary welds consist of the circumferential welds attaching the bottom closure and the top flange to the vessel shell. Also, the longitudinal weld(s) on the rolled plate, closing the cylindrical vessel shell, and the circumferential weld(s) attaching the rolled shells together are containment welds.

Double metal seals are utilized on the lid and the two lid penetrations. Helicoflex HND or equivalent seals may be used. The seals are shown in Figure A7A.8-1. The internal spring and lining maintain the necessary rigidity and sealing force, and provide some elastic recovery capability. The outer aluminum jacket provides a ductile material to seal against the sealing surfaces. The jacket also provides a connecting sheet between the inner outer seals.

Holes in this sheet allow for attachment screws and for communication between the overpressure system and the space between the seals. This sheet, which is about 0.020 in. thick, has insufficient strength to transmit radial forces great enough to overcome the axial compressive forces on the seals, which are over 1000 lb/in. of seal length. Additional information on the seals is provided in Section A3.3.2. The overpressure port seal is a single metal seal of the same design, Helicoflex HN200 or equivalent.

All TN-40HT surfaces which mate with the metal seals are stainless steel.

The use of a double seal system allows the TN-40HT cask to have a pressure monitoring of the interspace between the seals. This combined cover-seal pressure monitoring system always meets or exceeds the requirement of a double barrier closure which guarantees tight, permanent confinement. When the cask is placed in storage a pressure greater than the cavity pressure is set up in the gaps (interspace) between the double metal seals of the lid and the lid penetrations. A decrease in the pressure of the monitoring system would be signaled by a pressure transducer in the overpressure system.

The lid and penetration seals described above are contained in grooves. A high level of sealing over the storage period is assured by utilizing seals in a deformation-controlled design. The deformation of the seals is constant since bolt loads assure that the mating surfaces remain in contact. The seal deformation is set by the original diameter and the depth of the groove.

The nominal cross-section diameter of the lid seal is 0.26 in. and the nominal groove depth is 0.22 in. At 0.04 in. compression, the sealing force is 1,399 lb/inch (Reference 6). The total force of the double seal is 660,129 lb. The total minimum preload of the 48 lid bolts is 71,111 lb/bolt b, which is greater than the combined force of the seals and internal pressure, 23,175 lb/bolt (Appendix A4A.4).

The nominal cross-section diameter of the port seals is 0.161 in. and the nominal groove depth is 0.126 in. At .035 in. compression, the sealing force is 1,142 lb/inch. The total force of the double seal is 37,922 lb. The total minimum preload of the 8 cover bolts is 7,111 lb/bolt, which is greater than the combined force of the seals and internal pressure, 5,058 lb/bolt (Appendix A4A.5).

The maximum radial force on the seals is from the 5.5 atm abs overpressure system. This results in a force per unit seal length of about 15 lb./in., which is negligible compared to the compressive (axial) force of over 1000 lb/inch. Because the maximum pressure is between the two seals, the direction of this force is such that the seals are supported by the walls of the seal groove. However, the seals are designed to retain pressure in either direction.

Helicoflex metal seals are all capable of limiting leak rates to much less than 1×10^{-5} ref cm^3/s. After loading, all lid and cover seals are leak tested and the acceptable total cask leakage (both inner and outer seals combined) is 1×10^{-5} ref cm^3/s with a minimum test sensitivity of 5×10^{-6} atm-cc/sec.

A7A.8.3 CLOSURE

The containment vessel contains an integrally-welded bottom closure and a bolted and flanged top closure (lid). The lid plate is attached to the cask body with 48 bolts. The bolt torque required to seal the metal seals located in the lid and maintain confinement under normal and accident conditions is provided in Drawing TN40HT-72-1 in Section A1.5. The closure bolt analysis is presented in Section A4A.4.

The lid contains two penetrations which are sealed by flanged covers fastened to the lid by 8 bolts each. The bolt torque required to seal the metal seals in the penetration covers and maintain confinement under normal and accident conditions is provided in Drawing TN40HT-72-1 in Section A1.5. The penetration bolt analysis is presented in Section A4A.5.

A7A.8.4 MONITORING OF SYSTEM CONFINEMENT

An overpressure (OP) monitoring system is part of the TN-40HT design. The pressure in the monitoring system is greater than that of the cask cavity and the cask cavity pressure is greater than ambient. In this configuration, neither in-leakage of air nor out-leakage of cavity gas is possible.

If a leak existed in the seals, the design of the TN-40HT overpressure system is such that the leak will either be to the atmosphere or to the cask cavity. Leakage from the cask cavity past the higher pressure of the OP system is physically impossible.

The seals are collectively leak tested to 1 x 10^{-5} ref cm^3/s. Using the methodology of ANSI N14.5 (Reference 7), an equivalent maximum hole size is estimated based upon test conditions of equivalent air leaking from 1 atm abs to 0.01 atm abs in ambient temperature conditions (77 °F or 25 °C) and the maximum acceptable leak of 1 x 10^{-5} ref cm^3/s. The leakage hole length is assumed to be the same as the metal seal width, 0.5 cm. The equivalent maximum hole size is calculated below.

$$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u) \text{ cc/sec at } T_u, P_u$$

Other definitions:

- L_u = upstream volumetric leakage rate, cc/sec = 1 x 10^{-5} ref cm^3/s (Test Leak Rate)
- F_c = coefficient of continuum flow conductance per unit pressure, cc/atm-sec
- F_m = coefficient of free molecular flow conductance per unit pressure, cc/atm-sec
- P_u = fluid upstream pressure, atm abs = 1.0 atm abs
- P_d = fluid downstream pressure, atm abs = 0.01 atm abs
- D = leakage hole diameter, cm
- a = leakage hole length, cm = 0.5 cm (assuming leak path length is on the order of the metal seal width)
- μ = fluid viscosity, cP = 0.0185 cP (from ANSI N14.5, Table B.1)
- T = fluid absolute temperature, K = 298 K
- M = molecular weight, g/mol = 29.0 g/mol (from ANSI N14.5, Table B.1)
- P_a = average stream pressure = ½ ($P_u + P_d$), atm abs = 0.505 atm abs

$$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u) \text{ cc/sec}$$

where:

$$F_c = (2.49\text{x}10^6 \times D^4)/(a\mu) \text{ cc/atm-sec}$$
$$F_m = \{3.81\text{x}10^3 \times D^3 \times (T/M)^{0.5}\} / (aP_a) \text{ cc/atm-sec}$$

Substituting:

$$F_c = (2.49\text{x}10^6 \times D^4)/(0.5 \times 0.0185) = 2.69 \times 10^8\ D^4$$
$$F_m = \{3.81\text{x}10^3 \times D^3 \times (298/29.0)^{0.5}\} / \{0.5 \times 0.505) = 4.84 \times 10^4\ D^3$$

$$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u) \text{ cc/sec}$$

A7A.8.5 CONFINEMENT REQUIREMENTS FOR NORMAL CONDITIONS OF STORAGE

The TN-40HT dry storage cask is designed to provide storage of spent fuel for at least 25 years. The cask cavity pressure is always above ambient during the storage period as a precaution against the in-leakage of air which might be harmful to the fuel. Since the containment vessel consists of a steel cylinder with an integrally-welded bottom closure, the cavity gas can escape only through the lid closure system. In order to ensure no release of radioactivity material, two systems are employed. First, all bolted closures are provided with double seals. Second, the interspace between the seals is pressurized to provide a positive pressure gradient. If the inner seals were to leak, helium would flow into the cask cavity and radioactive material would not be released. If the outer seals were to leak, helium would leak from the overpressure system to the exterior, and no radioactive material would be released.

The cask loadings for normal conditions of storage are given in Section A3.2.5. It is shown that the seals are not disturbed by any of the loadings and thus, the cask confinement is maintained.

A7A.8.6 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

A7A.8.6.1 SOURCE TERMS FOR CONFINEMENT CALCULATIONS

Table A7.2-6 lists the activity representing the fission gases, volatiles, and fines contributing more than 0.1% of the activity contained in a design basis fuel, plus Iodine 129.

The releasable source term is first determined. The release fractions (References 8 and 9) applied to the source term are provided below.

Variable	Off-Normal Conditions	Accident Conditions
Fraction of crud that spalls off rods, f_C	0.15	1.0
Fraction of Rods that develop cladding breaches, f_B	0.10	1.0
Fraction of Gases that are released due to a cladding breach, f_G	0.3	0.3
Fraction of Fines that are released due to a cladding breach, f_F	3×10^{-6}*	3×10^{-6}*
Fraction of Volatiles that are released due to a cladding breach, f_V	2×10^{-4}	2×10^{-4}

* Per NUREG-1617, 3×10^{-5} of the fines are released during a cladding breach. Per SAND90-2406, (Reference 10), page IV-7, of the 3×10^{-5} of the fines released recommends that only 10% of the fuel fines ejected remain airborne.

A7A.8.6.2 RELEASE OF CONTENTS

Two scenarios are considered:

- Off-Normal Conditions – This condition exists over a 45 day period, seals are leaking at the test leak rate of 1×10^{-5} ref cm^3/s and the fraction of rods that have failed is 10%. Stability category D and a 5 m/s wind speed are used for this analysis. This scenario assumes one cask is in off-normal condition at the ISFSI. The 45 day exposure duration will serve as the bases for the allowed completion times for the Cask Interseal Pressure Technical Specification.

- Hypothetical Accident Conditions – This condition exists over a 30 day period, seals are leaking at the test leak rate of 1×10^{-5} ref cm^3/sec, the fraction of rods that have failed is 100%, and the temperature inside the cask is comparable to the fire accident conditions. Stability category F and 1 m/s wind speed is used for this analysis. This scenario assumes one cask is in the hypothetical accident condition at the ISFSI.

In the first scenario, the release is assumed to occur for more than a 20 minute period. The methodology of Reg Guide 1.145 (Reference 11) is applied. The atmospheric diffusion from a ground level point source at 110 meters is based on the following parameters.

Wind speed = 5 meter/second
σ_y = 9 meters (Reference 11, Figure 1)
σ_z = 5 meters (Reference 11, Figure 2)
M = 1.1, (Reference 11, Figure 3]
$\Sigma_y = M\sigma_y$ = 9.9 meters
A = cross sectional area of the TN-40HT = $9.1m^2$

Using the methodology of Reg Guide 1.145, $\{\chi/Q\}_{110\ meters}$ during off-normal conditions is 1.29E-03 sec/m^3. Similarly, the atmospheric diffusion for 700 meters which corresponds to the distance from the ISFSI to the nearest residence (700 meters is a conservative value, 724 meters is the distance from the center of the ISFSI to the nearest resident), during off-normal conditions is calculated using the following parameters.

Wind speed = 5 meter/second
σ_y = 50 meters
σ_z = 25 meters
M = 1.1
$\Sigma_y = M\sigma_y$ = 55 meters

During off normal conditions $\{\chi/Q\}_{700\ meters}$ is 4.63E-05 sec/m^3.

In the second scenario the release is assumed to be a short term ground level release (occurring, however, over a 30 day period) assuming the methodology of Regulatory Guide 1.25 (Reference 12). The atmospheric stability classification of F and a wind speed of 1 m/sec are used. The atmospheric diffusion from a ground level point source at 110 and 724 meters is taken from Table 8.2-1 and Table 2.3-2 to be:

$\chi/Q_{110\ \text{meters}}$ = 6.63E-03 sec/m^3 and

$\chi/Q_{0.45\text{miles}}$ = 2.66E-04 sec/m^3

(nearest resident is 0.45 miles, about 724 meters from center of ISFSI)

A7A.8.6.2.1 DOSE CALCULATIONS

Dose components are calculated following the method of Regulatory Guide 1.109 (Reference 13) and utilizing dose conversion factors from EPA Federal Guidance Reports Numbers 11 and 12 (References 14 and 15). (Note: Two sets of dose conversion factors (DCFs) depending upon the chemical state of Sr-90 are reported in Federal Guidance Report Number 11. One set of DCF values is for Sr in the form of $SrTiO_3$ and the other set is for Sr in all other forms. The Sr-90 fission product should not form $SrTiO_3$ within the storage cask and therefore the DCF for this compound was not used.)

To determine the committed doses (from air inhalation), the following equation is used:

$$\text{Dose}_{\text{inhalation}} = R \times \chi/Q \times Q \times \text{DCF}_{\text{inhalation}} \times \text{Time}$$

Where:

R = Inhalation Rate = 8,000 m^3/year = 2.54E-04 m^3/sec (References 8 and13)

χ/Q = Short term average centerline value of atmospheric diffusion for a ground level release (sec/m^3)

Q = amount of material released (μCi/sec)

$\text{DCF}_{\text{inhalation}}$ = Exposure Dose Conversion Factor (mrem/μCi), from (Reference 14).

Time = Time of Exposure (Seconds)

To determine the deep doses (from air immersion), the following equation is used:

$$\text{Dose}_{\text{air immersion}} = \{\chi/Q \times Q \times \text{DCF}_{\text{air immersion}}\} \times \text{Time}$$

Where:

χ / Q = Short term average centerline value of atmospheric diffusion for a ground level release (sec/m^3)

Q = amount of material released (μCi/sec)

$\text{DCF}_{\text{immersion}}$ = Exposure Dose Conversion Factor (mrem/year per μCi/cm^3), (Reference 15)

Time = Time of Exposure (Seconds)

The maximum 30-day TEDE value is 0.024 rem. The corresponding 10CFR 72.106 limit is 5 rem.

The maximum 30-day Lens Dose Equivalent value is 0.0241 rem. The corresponding 10CFR 72.106 limit is 15 rem.

The maximum 30-day dose to any organ/tissue is 0.244 rem and it occurs at the bone surface. The corresponding 10CFR 72.106 limit is 50 rem.

Therefore all the criteria of 72.106 are met at 110 m.

A summary of the doses at 110 m and their corresponding regulatory limits is shown below.

Off-Normal Conditions		
Organ	**10CFR72.104(a) Limit (mrem)**	**110 meter Dose (mrem)**
Whole Body (TEDE)	25	0.30
Thyroid	75	0.01
Other Critical Organ	25	2.90 (Bone Surface)
Accident Conditions		
Organ	**10CFR72.106(b) Limit (mrem)**	**110 meter Dose (mrem)**
Whole Body (TEDE)	5000	24.0
Organ (TODE)	50000	244 (Bone Surface)
Lens of Eye (LDE)	15000	24.1
Skin (SDE)	50000	0.14

A7A.8.6.3 LATENT SEAL FAILURE

By design the overpressure monitoring system does not immediately alarm if there is a leak in a seal or the overpressure system. The time period from when a leak begins to occur and when the overpressure system alarm is activated is dependent on the size of the leak. Two conditions which could exist within the TN-40HT confinement system are:

(1) The outer seal (or the overpressure system) is leaking to the atmosphere. In this case the inner seal is intact and there is no release of the contents of the cask cavity to the atmosphere.

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

TABLE A7A.8-1
TN-40HT RELEASABLE SOURCE TERM FOR OFF-NORMAL CONDITIONS – DESIGN BASIS 14X14 FUEL

Isotope	Activity (Ci/assembly)	Release Fraction	Concentration in Void Space of TN-40HT[1] (Ci/cm^3)	Material Released[2] Q ($\mu Ci/sec$)
H 3	1.78E+02	0.30	3.80E-05	5.76E-04
Co60[(3)]	6.73E+00	1.50E-01	7.18E-06	1.09E-04
Pu238	3.02E+03	3.00E-06	6.44E-09	9.78E-08
Pu239	1.35E+02	3.00E-06	2.88E-10	4.37E-09
Pu240	2.67E+02	3.00E-06	5.70E-10	8.65E-09
Pu241	3.19E+04	3.00E-06	6.81E-08	1.03E-06
Am241	1.50E+03	3.00E-06	3.20E-09	4.86E-08
Am243	4.17E+01	3.00E-06	8.90E-11	1.35E-09
Cm244	5.28E+03	3.00E-06	1.13E-08	1.71E-07
Kr 85	1.78E+03	0.30	3.80E-04	5.77E-03
Sr 90	3.11E+04	2.00E-04	4.42E-06	6.72E-05
Y 90	3.11E+04	3.00E-06	6.64E-08	1.01E-06
I129	2.40E-02	0.30	5.12E-09	7.77E-08
Cs134	4.01E+02	2.00E-04	5.70E-08	8.66E-07
Cs137	5.27E+04	2.00E-04	7.50E-06	1.14E-04
Ba137m	4.97E+04	3.00E-06	1.06E-07	1.61E-06
Pm147	6.53E+02	3.00E-06	1.39E-09	2.12E-08
Eu154	1.33E+03	3.00E-06	2.84E-09	4.31E-08
Np239	4.17E+01	3.00E-06	8.90E-11	1.35E-09

1. Values are based on 10% failure of the fuel rods and cask free volume of 5.63 m^3.

2. Values are based on 1.518E-05 cm^3/sec helium leak from containment.

3. The Co-60 source is calculated using the methodology of Reference (Reference 8). It is based on a 14x14 PWR fuel assembly with surface area of 1300 cm^2/rod and a crud surface concentration of 140μCi /cm^2 (per Table 7.1 of Reference 8) at the time of discharge. (The value listed above includes a minimum cooling time of twelve years.)

TABLE A7A.8-3
OFF-SITE AIRBORNE DOSES FROM OFF-NORMAL CONDITIONS AT 110 M (INTERNAL + EXTERNAL)

Design Basis 14x14 Fuel, Committed Doses (Internal) + Deep Dose (External)
mrem for 45 days

Isotope	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remainder	Effective	Skin
H3	4.68E-05	4.68E-05	4.69E-05	4.68E-05	4.68E-05	4.68E-05	4.68E-05	4.68E-05	0.00E+00
Co60	2.69E-03	9.71E-03	1.77E-01	9.06E-03	7.28E-03	8.56E-03	1.87E-02	3.05E-02	2.93E-04
Pu238	1.29E-02	4.60E-07	1.47E-01	6.99E-02	8.73E-01	4.42E-07	3.23E-02	4.87E-02	7.40E-11
Pu239	6.53E-04	1.89E-08	6.64E-03	3.47E-03	4.34E-02	1.86E-08	1.55E-03	2.38E-03	1.51E-12
Pu240	1.29E-03	3.87E-08	1.31E-02	6.87E-03	8.58E-02	3.68E-08	3.07E-03	4.71E-03	6.27E-12
Pu241	3.31E-03	1.49E-07	1.54E-02	1.63E-02	2.04E-01	6.02E-08	6.36E-03	1.08E-02	2.24E-12
Am241	7.42E-03	6.11E-07	4.20E-03	3.97E-02	4.95E-01	3.66E-07	1.79E-02	2.74E-02	1.15E-09
Am243	2.07E-04	9.65E-08	1.13E-04	1.10E-03	1.38E-02	5.27E-08	4.91E-04	7.55E-04	6.87E-11
Cm244	1.28E-02	8.36E-07	1.55E-02	7.54E-02	9.40E-01	8.12E-07	3.84E-02	5.38E-02	1.24E-10
Kr85	1.25E-05	1.43E-05	1.22E-05	1.16E-05	2.35E-05	1.26E-05	1.16E-05	1.27E-05	1.41E-03
Sr90	8.33E-04	8.33E-04	1.18E-03	1.06E-01	2.29E-01	8.33E-04	1.81E-03	1.11E-01	1.14E-05
Y90	4.86E-08	4.92E-08	4.41E-05	1.32E-06	1.32E-06	4.86E-08	1.83E-05	1.08E-05	1.16E-06
I129	3.24E-08	7.73E-08	1.15E-07	5.14E-08	5.20E-08	5.70E-04	4.34E-08	1.71E-05	1.58E-09
Cs134	5.41E-05	4.53E-05	4.92E-05	4.92E-05	4.67E-05	4.64E-05	5.77E-05	5.21E-05	1.51E-06
Cs137	4.68E-03	4.19E-03	4.72E-03	4.44E-03	4.25E-03	4.24E-03	4.88E-03	4.62E-03	1.82E-05
Ba137m	8.40E-07	9.59E-07	8.34E-07	8.13E-07	1.38E-06	8.58E-07	7.98E-07	8.58E-07	1.11E-06
Pm147	2.16E-12	3.95E-12	7.69E-06	8.11E-07	1.01E-05	2.23E-12	5.85E-07	1.05E-06	3.17E-10
Eu154	2.42E-06	3.19E-06	1.61E-05	2.15E-05	1.06E-04	1.49E-06	2.29E-05	1.57E-05	6.61E-08
Np239	6.61E-10	3.22E-10	1.52E-08	1.48E-09	1.34E-08	2.36E-10	6.26E-09	4.50E-09	4.00E-10
Total	4.69E-02	1.48E-02	3.85E-01	3.32E-01	2.90E+00	1.43E-02	1.26E-01	2.95E-01	1.73E-03

TABLE A7A.8-4
OFF-SITE AIRBORNE DOSES FROM OFF-NORMAL CONDITIONS AT 110 M (EXTERNAL)

Design Basis14x14 Fuel, Deep Doses (External)
mrem for 45 days

Isotope	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remainder	Effective	Skin
H3	0.00E+00	0.00E+00	2.93E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.53E-09	0.00E+00
Co60	2.48E-04	2.81E-04	2.50E-04	2.48E-04	3.59E-04	2.56E-04	2.42E-04	2.54E-04	2.93E-04
Pu238	1.19E-11	2.30E-11	1.92E-12	3.04E-12	1.68E-11	7.26E-12	3.60E-12	8.83E-12	7.40E-11
Pu239	3.92E-13	6.11E-13	2.14E-13	2.16E-13	7.66E-13	3.14E-13	2.31E-13	3.43E-13	1.51E-12
Pu240	1.02E-12	1.97E-12	1.74E-13	2.64E-13	1.48E-12	6.27E-13	3.14E-13	7.60E-13	6.27E-12
Pu241	1.37E-12	1.66E-12	1.24E-12	1.08E-12	4.19E-12	1.33E-12	1.16E-12	1.39E-12	2.24E-12
Am241	7.72E-10	9.62E-10	6.06E-09	4.68E-10	2.58E-09	7.04E-10	5.70E-10	7.36E-10	1.15E-09
Am243	5.47E-11	6.52E-11	4.80E-11	3.87E-11	1.87E-10	5.22E-11	4.47E-11	5.45E-11	6.87E-11
Cm244	2.18E-11	4.21E-11	2.24E-12	4.62E-12	2.79E-11	1.33E-11	5.73E-12	1.55E-11	1.24E-10
Kr85	1.25E-05	1.43E-05	1.22E-05	1.16E-05	2.35E-05	1.26E-05	1.16E-05	1.27E-05	1.41E-03
Sr90	9.67E-09	1.18E-08	8.00E-09	6.76E-09	2.83E-08	9.11E-09	7.59E-09	9.36E-09	1.14E-05
Y90	3.52E-09	4.10E-09	3.30E-09	3.02E-09	8.28E-09	3.49E-09	3.13E-09	3.54E-09	1.16E-06
I129	6.95E-10	9.58E-10	3.08E-10	2.36E-10	1.58E-09	5.55E-10	3.31E-10	5.47E-10	1.58E-09
Cs134	1.19E-06	1.35E-06	1.18E-06	1.15E-06	1.92E-06	1.21E-06	1.13E-06	1.21E-06	1.51E-06
Cs137	1.68E-08	2.04E-08	1.41E-08	1.20E-08	4.82E-08	1.59E-08	1.34E-08	1.63E-08	1.82E-05
Ba137m	8.40E-07	9.59E-07	8.34E-07	8.13E-07	1.38E-06	8.58E-07	7.98E-07	8.58E-07	1.11E-06
Pm147	2.93E-13	3.74E-13	2.13E-13	1.75E-13	8.53E-13	2.64E-13	2.06E-13	2.71E-13	3.17E-10
Eu154	4.78E-08	5.43E-08	4.78E-08	4.68E-08	7.52E-08	4.90E-08	4.58E-08	4.90E-08	6.61E-08
Np239	1.88E-10	2.18E-10	1.79E-10	1.62E-10	5.00E-10	1.88E-10	1.69E-10	1.92E-10	4.00E-10
Total	2.63E-04	2.97E-04	2.65E-04	2.62E-04	3.86E-04	2.71E-04	2.56E-04	2.69E-04	1.73E-03

TABLE A7A.8-5
OFF-SITE AIRBORNE DOSES FROM ACCIDENT CONDITIONS AT 110 M (INTERNAL)

Design Basis 14x14 Fuel, mrem/30 Days, Committed Doses (Internal)

Isotope	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remainder	Effective
H3	3.95E-03	3.95E-03	3.95E-03	3.95E-03	3.95E-03	3.95E-03	3.95E-03	3.95E-03
Co60	1.37E-01	5.30E-01	9.94E+00	4.96E-01	3.89E-01	4.67E-01	1.04E+00	1.70E+00
Pu238	1.09E+00	3.88E-05	1.24E+01	5.89E+00	7.37E+01	3.73E-05	2.72E+00	4.11E+00
Pu239	5.51E-02	1.60E-06	5.60E-01	2.93E-01	3.66E+00	1.56E-06	1.31E-01	2.01E-01
Pu240	1.09E-01	3.26E-06	1.11E+00	5.79E-01	7.23E+00	3.10E-06	2.59E-01	3.98E-01
Pu241	2.79E-01	1.25E-05	1.30E+00	1.38E+00	1.72E+01	5.08E-06	5.36E-01	9.13E-01
Am241	6.26E-01	5.14E-05	3.54E-01	3.35E+00	4.18E+01	3.08E-05	1.51E+00	2.31E+00
Am243	1.75E-02	8.14E-06	9.53E-03	9.26E-02	1.16E+00	4.44E-06	4.14E-02	6.37E-02
Cm244	1.08E+00	7.05E-05	1.31E+00	6.36E+00	7.93E+01	6.85E-05	3.24E+00	4.54E+00
Kr85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	7.03E-02	7.03E-02	9.93E-02	8.94E+00	1.93E+01	7.03E-02	1.53E-01	9.34E+00
Y90	3.80E-06	3.80E-06	3.72E-03	1.11E-04	1.11E-04	3.80E-06	1.54E-03	9.10E-04
I129	2.68E-06	6.44E-06	9.67E-06	4.31E-06	4.25E-06	4.81E-02	3.64E-06	1.44E-03
Cs134	4.46E-03	3.71E-03	4.05E-03	4.05E-03	3.77E-03	3.81E-03	4.77E-03	4.29E-03
Cs137	3.95E-01	3.54E-01	3.98E-01	3.74E-01	3.58E-01	3.58E-01	4.11E-01	3.89E-01
Ba137m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm147	1.58E-10	3.02E-10	6.49E-04	6.84E-05	8.55E-04	1.66E-10	4.94E-05	8.89E-05
Eu154	2.00E-04	2.65E-04	1.35E-03	1.81E-03	8.93E-03	1.22E-04	1.93E-03	1.32E-03
Np239	3.99E-08	8.73E-09	1.26E-06	1.11E-07	1.09E-06	4.08E-09	5.13E-07	3.63E-07
Total	3.86E+00	9.62E-01	2.75E+01	2.78E+01	2.44E+02	9.51E-01	1.00E+01	2.40E+01

TABLE A7A.8-6
OFF-SITE AIRBORNE DOSES FROM ACCIDENT CONDITIONS AT 110 M (EXTERNAL)

Design Basis 14x14 Fuel, mrem/30 Days, Deep Doses (External)

	Gonad	Breast	Lung	R. Marrow	B. Surface	Thyroid	Remainder	Effective	Skin
H3	0.00E+00	0.00E+00	2.47E-06	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.98E-07	0.00E+00
Co60	1.40E-02	1.58E-02	1.41E-02	1.40E-02	2.02E-02	1.44E-02	1.36E-02	1.43E-02	1.65E-02
Pu238	1.00E-09	1.94E-09	1.62E-10	2.56E-10	1.42E-09	6.12E-10	3.04E-10	7.45E-10	6.24E-09
Pu239	3.30E-11	5.15E-11	1.81E-11	1.82E-11	6.46E-11	2.65E-11	1.95E-11	2.89E-11	1.27E-10
Pu240	8.58E-11	1.66E-10	1.47E-11	2.23E-11	1.25E-10	5.29E-11	2.65E-11	6.41E-11	5.29E-10
Pu241	1.16E-10	1.40E-10	1.05E-10	9.08E-11	3.53E-10	1.13E-10	9.82E-11	1.17E-10	1.89E-10
Am241	6.51E-08	8.11E-08	5.11E-07	3.95E-08	2.18E-07	5.94E-08	4.81E-08	6.20E-08	9.71E-08
Am243	4.62E-09	5.50E-09	4.05E-09	3.27E-09	1.57E-08	4.41E-09	3.77E-09	4.60E-09	5.80E-09
Cm244	1.84E-09	3.55E-09	1.89E-10	3.90E-10	2.35E-09	1.12E-09	4.83E-10	1.31E-09	1.04E-08
Kr85	1.05E-03	1.21E-03	1.03E-03	9.81E-04	1.98E-03	1.06E-03	9.81E-04	1.07E-03	1.19E-01
Sr90	8.15E-07	9.95E-07	6.75E-07	5.70E-07	2.39E-06	7.68E-07	6.40E-07	7.89E-07	9.64E-04
Y90	2.97E-07	3.46E-07	2.78E-07	2.55E-07	6.98E-07	2.94E-07	2.64E-07	2.99E-07	9.81E-05
I129	5.86E-08	8.08E-08	2.60E-08	1.99E-08	1.33E-07	4.68E-08	2.79E-08	4.61E-08	1.33E-07
Cs134	1.00E-04	1.14E-04	9.96E-05	9.72E-05	1.62E-04	1.02E-04	9.54E-05	1.02E-04	1.28E-04
Cs137	1.41E-06	1.72E-06	1.19E-06	1.01E-06	4.07E-06	1.34E-06	1.13E-06	1.37E-06	1.53E-03
Ba137m	7.09E-05	8.09E-05	7.04E-05	6.86E-05	1.16E-04	7.24E-05	6.73E-05	7.24E-05	9.37E-05
Pm147	2.47E-11	3.16E-11	1.80E-11	1.47E-11	7.20E-11	2.23E-11	1.74E-11	2.29E-11	2.68E-08
Eu154	4.03E-06	4.58E-06	4.03E-06	3.95E-06	6.34E-06	4.14E-06	3.87E-06	4.13E-06	5.57E-06
Np239	1.59E-08	1.84E-08	1.51E-08	1.37E-08	4.22E-08	1.59E-08	1.43E-08	1.62E-08	3.37E-08
Total	1.52E-02	1.72E-02	1.53E-02	1.51E-02	2.25E-02	1.57E-02	1.48E-02	1.55E-02	1.38E-01

A8.2.8 HYPOTHETICAL CASK DROP ACCIDENT

A8.2.8.1 CAUSE OF ACCIDENT

The stability of the TN-40HT storage cask in the upright position on the ISFSI concrete storage pad is demonstrated in Section A3.2. The effects of tornado wind and missiles, flood water and earthquakes are described in Sections A3.2.1, A3.2.2 and A3.2.3, respectively. It is shown in those sections that the cask will not tip over under the most severe natural phenomena specified in the Prairie Island Updated Safety Analysis Report.

The cask is lifted at Prairie Island using a single failure proof crane. The upper trunnions are designed to meet the requirements of NUREG-0612 (Reference 5) for non-redundant lifting fixture. This is accomplished by evaluating the trunnions to the stress design factors required by ANSI N14.6 (Reference 3), i.e. capable of lifting 6 times and 10 times the cask weight without exceeding the yield and ultimate strengths of the material, respectively. The loaded cask will be handled by the transport vehicle in a vertical orientation and not lifted higher than 18 in.

However section of the SAR considers design events of the third and fourth types (includes accidents) as defined in ANSI/ANS 57.9. The third type of events are those that could reasonably be expected to occur over the lifetime of the ISFSI (does not include dropping of the cask). The fourth type of event includes severe natural phenomena (described in Section A8.2.1 through A8.2.5) and man-induced low probability events postulated because their consequences could result in the maximum potential impact on the immediate environs. Therefore the cask is examined for a dropping accident which is an impact event that is extremely unlikely to occur.

A8.2.8.2 ACCIDENT ANALYSIS

In this section the cask is evaluated under bottom end impact on the ISFSI storage pad after a drop from a height of 18 in. The storage pad is the hardest concrete surface outside of the containment building. The cask is always oriented vertically and is never lifted higher than 18 in. once it leaves the containment building. Therefore this case is an upper bound drop event since impact onto a softer surface would result in lower cask deceleration and a lower impact force.

A8.2.8.2.1 DYNAMIC IMPACT LOADS

The peak decelerations in the cask and basket during the 18 inch end drop were calculated by a dynamic nonlinear analysis described in Section A4A.10. The analysis showed a maximum acceleration in the TN40HT cask body of 44.1g. This occurred in the bottom plate. The highest acceleration in the basket and fuel was 28.8g. However, since the basket and fuel were not modeled explicitly, the maximum acceleration (28.8g) must be multiplied by the dynamic load factor of 1.52 resulting in a maximum loading of 43.8g.

A8.2.8.2.2 CASK BODY ANALYSIS

The cask is analyzed conservatively for a 50 g vertical load simulating the end drop. The evaluation is presented in Section A4.2.3.4. All calculated stresses meet code allowables.

A8.2.8.2.3 LID BOLT ANALYSIS

During a bottom end drop, the rim of the lid is forced against the flange of the cask body. The lid is initially seated against the flange by preloading (torquing) the bolts. The bolt preload will not be affected if compressive yielding of the contact bearing area does not occur.

The evaluation of the cask presented in Section A4.2.3.4 shows that during a drop cask accident the maximum stresses in the lid outer plate and the shell flange are less than the yield stress. Thus the bolt preload will not be affected by the bottom drop. Therefore, this hypothetical accident case will not affect the bolts.

A8.2.8.2.4 BASKET ANALYSIS

The basket is analyzed conservatively for a 50 g vertical load simulating the end drop. The evaluation is presented in, Section A4.2.3.4. All calculated stresses meet code allowables.

A8.2.8.3 ACCIDENT DOSE CALCULATIONS

Cask drop will not breach the cask confinement barrier. No radioactivity will be released and no resultant doses will occur.

However, a bounding dose can be determined. The loss of neutron shielding (322 mrem from Section A8.2.5) is combined with the total effective dose equivalent (TEDE) from the loss of one confinement barrier and 100% fuel cladding failure (24 mrem from Section A8.2.9). The resulting site boundary accident dose, 346 mrem, is below the 5 rem TEDE limit as specified in 10 CFR 72.106(b) (Reference 2).

A8.2.9 LOSS OF CONFINEMENT BARRIER

A8.2.9.1 CAUSE OF ACCIDENT

A combined event of failure of one of the seals in addition to a failure of the pressure monitoring system is assessed. This could also be a failure of the pressure boundary of the overpressure system.

A8.2.9.2 ACCIDENT ANALYSIS

Analysis has been performed in Appendix A4A to show that the bolts will be able to maintain the seal under accident conditions. Thus the leak rate is limited to the test leak rate of 1 $\times 10^{-5}$ ref cm^3/s.

A description of the three possible leaks which could occur is presented below:

- <u>In any of the inner containment seals (lid seal, inner vent seal or inner drain seal)</u>

  The lid and lid penetration cover bolts and seals are designed to prevent leakage during all postulated accident events. Therefore, this is a very unlikely event.

  In this case the overpressure system, which has a higher pressure than the cask cavity, would leak helium into the cask cavity. Since the pressure is higher in the overpressure tank, it would prevent leakage of radioactive materials out of the cask cavity until the pressure between the overpressure tank and the cask cavity equalized. This would take several years, depending on the size of the leak. At the test leak rate, the overpressure system pressure would always exceed the cask cavity pressure, as shown in Appendix A7A. Therefore no leakage of radioactive material can occur, even if the alarm were to fail. Appendix A7A also demonstrates that even if the inner seal has experienced a latent seal failure there is ample time for identifying the leak through routine surveillances.

- <u>In any of the outer seals (lid, overpressure port cover, vent cover or drain cover)</u>

  The lid and lid penetration cover bolts and seals are designed to prevent leakage during all postulated accident events. Therefore, this is a very unlikely event.

  In this case, leakage out of the interspace to the atmosphere would occur. This would not result in release of radioactive material from the cask cavity since the inner seal is intact. Again, as demonstrated in Appendix A7A, a latent seal failure of the outer seals would not result in a release of any radioactive material to the environment. There is also ample time for identifying the leak through routine surveillances.

- <u>A leak in the overpressure system</u>

  This is the most likely cause of a leak, since it is a non safety related component and not designed to withstand accident loadings.

  In this case two scenarios could exist:

  - The overpressure system is not functioning and the inner seal is intact. In this case there is no release of radioactive material to the environment; or

  - The overpressure system is not functioning and the inner seal is leaking at some rate.

In this latter case, leakage out of the interspace to the atmosphere and the cask cavity could occur. This would not result in release of radioactive material from the cask cavity until the pressure fell to the cask cavity pressure.

At the test leak rate of 1×10^{-5} ref cm^3/s, this would not occur during a 25 year storage period. However, a leak of this magnitude in combination with a loss of the over pressure system has been evaluated in Appendix A7A.

A8.2.9.3 ACCIDENT DOSE CALCULATIONS

The results of the calculations in Appendix A7A assuming accident conditions indicated that at the site boundary (110m from the cask), for a 30 day release, the total effective dose equivalent is 24 mrem. The total organ dose equivalent to any individual organ (the critical organ in this case is the bone surface) is 244 mrem for a 30 day release. The lens dose equivalent to the lens of the eye is 24.1 mrem for a 30 day release. These values are well below the limiting off site doses defined in 10 CFR 72.106(b).

Another accident condition under consideration is that the overpressure system is not functioning and the inner seal has experienced a latent seal failure. This analysis is presented in Appendix A7A. This accident analysis demonstrates that a latent failure up to 100 times greater that the test value could occur and there is ample time for recovery before the limiting off site doses in 10 CFR 72.106(b) are met. The probability that a gross leak of an inner seal in combination with a gross leak in the outer seal is not considered a credible event.

A9.7 ADDITIONAL FABRICATION TESTING AND INSPECTIONS

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.

A9.7.2 WELDING REQUIREMENTS AND INSPECTIONS

Qualification of welding procedures and welders shall be determined using Section IX of the ASME Code, Reference 4.

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 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.

Non-confinement welds shall be inspected in accordance with ASME Code Subsection NF including alternatives to the Code as specified in SAR Section A3.5.

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.

A9.7.3 NEUTRON ABSORBER REQUIREMENTS

The neutron absorber used for criticality control in the TN-40HT basket may consist any of the following types of material:

(a) Boron-aluminum alloy (borated aluminum)

(b) Boron carbide / aluminum metal matrix composite (MMC)

(c) Boral®

The TN-40HT safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials.

To assure performance of the neutron absorber's design function only visual inspections, thermal conductivity testing, and the presence / uniformity of B10 need to be verified with testing requirements specific to each material.

References to metal matrix composites throughout this chapter are not intended to refer to borated aluminum or Boral®.

A9.7.3.1 Boron Aluminum Alloy (Borated Aluminum)

Description

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating as a uniform fine dispersion of discrete aluminum diboride (AlB_2) or Titanium diboride (TiB_2) particles in the matrix of aluminum or aluminum alloy. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The boron may have the natural isotopic distribution or may be enriched in B10.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section A9.7.4.3.

Requirements

The boron content in the aluminum or aluminum alloy shall not exceed 5% by weight.

The neutron absorbers shall be 100% visually inspected in accordance with the inspection requirements described in Section A9.7.4.1.

The thermal conductivity of the material shall be tested in accordance with the testing requirements in Section A9.7.4.2.

The minimum B10 areal density specified in Table A3.3-17 shall be confirmed via neutron transmission testing as described in Section A9.7.4.3.

A9.7.3.2 BORON CARBIDE / ALUMINUM METAL MATRIX COMPOSITES (MMC)

Description

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. It is a low-porosity product, with a metallurgically bonded matrix.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section A9.7.4.3.

Requirements

For non-clad MMC products, the boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

Non-clad MMC products shall have a density greater than 98% of theoretical density , with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

Boron carbide particles for the products considered here typically have an average size in the range 10-40 microns, although the actual specification may be by mesh size, rather than by average particle size. No more than 10% of the particles shall be over 60 microns.

The neutron absorbers shall be 100% visually inspected in accordance with the inspection requirements described in Section A9.7.4.1.

The thermal conductivity of the material shall be tested in accordance with the testing requirements in Section A9.7.4.2.

The minimum B10 areal density specified in Table A3.3-17 shall be confirmed via neutron transmission testing as described in Section A9.7.4.3.

The MMCs material shall be qualified in accordance with the requirements specified in Section A9.7.5, and shall subsequently be subject to the process controls specified in Section A9.7.6.

A9.7.3.3 BORAL®

Description

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. The average size of the boron carbide particles in the finished product is approximately 50 microns after rolling.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®.

Requirements

The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The neutron absorbers shall be 100% visually inspected in accordance with the inspection requirements described in Section A9.7.4.1.

The thermal conductivity of the material shall be tested in accordance with the testing requirements in Section A9.7.4.2.

The minimum B10 areal density specified in Table A3.3-17 shall be confirmed via chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing described in Section A9.7.4.3. Areal density testing shall be performed on a coupon taken from the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

A9.7.4 NEUTRON ABSORBERS ACCEPTANCE TESTING

A9.7.4.1 VISUAL INSPECTIONS OF NEUTRON ABSORBERS

For borated aluminum and MMCs, visual inspections shall follow the recommendations in Aluminum Standards and Data (Reference 6), Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings". Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be treated as non-conforming. Inspection of MMCs with an integral aluminum cladding shall also include verification that the matrix is not exposed through the faces of the aluminum cladding and that solid aluminum is not present at the edges.

For Boral®, visual inspection shall verify that there are no cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet.

A9.7.4.2 THERMAL CONDUCTIVITY TESTING OF NEUTRON ABSORBERS

Testing shall conform to ASTM E1225 (Reference 7), ASTM E1461 (Reference 8), or equivalent method, performed at room temperature on coupons taken from the rolled or extruded production material. Previous testing of borated aluminum and metal matrix composite, Table A9.7-1, shows that thermal conductivity increases slightly with temperature. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.

If a thermal conductivity test result is below the specified minimum, additional tests may be performed on the material from that lot. If the mean value of those tests falls below the specified minimum the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the boron appearing in the same phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

The thermal analysis in Chapter A3.3.2.2 considers a dual plate basket construction base model with 0.125" thick neutron absorber with a 0.312" thick aluminum 1100 plate. This model gives the bounding values for the maximum component temperatures. Either a dual plate basket construction or an alternate single plate (borated aluminum or MMC) construction basket may be utilized. For the dual plate construction, the specified thickness of the neutron absorber may vary, and the thermal conductivity acceptance criterion for the neutron absorber will be based on the nominal thickness specified. In either construction type, to maintain the thermal performance of the basket, the minimum thermal conductivity shall be such that the total thermal conductance (sum of conductivity * thickness) of the neutron absorber and the aluminum 1100 plate shall at least equal the conductance assumed in the analysis for the base model. Samples of the acceptance criteria for various neutron absorber thicknesses are highlighted in Table A9.7-2.

The aluminum 1100 plate does not need to be tested for thermal conductivity; the material may be credited with the values published in the ASME Code Section II part D. The neutron absorber material need not be tested for thermal conductivity if the nominal thickness of the aluminum 1100 plate is 0.359 inch or greater.

A9.7.4.3 Neutron Transmission Testing of Neutron Absorbers

Neutron Transmission acceptance testing procedures shall be subject to approval by Transnuclear . Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in a lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes. The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1 inch diameter.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard. Standards will be calibrated, traceable to nationally recognized standards, or by attenuation of a monoenergetic neutron beam correlated to the known cross section of boron 10 at that energy.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 0.75 sq. inch.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. If a goodness-of-fit test demonstrates that the sample comes from a normal population, the one-sided tolerance limit for a normal distribution may be used for this purpose. Otherwise, a non-parametric (distribution-free) method of determining the one-sided tolerance limit may be used. Demonstration of the one-sided tolerance limit shall be evaluated for acceptance in accordance with Transnuclear 's QA procedures.

The following illustrates one acceptable method and is intended to be utilized as an example. The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined as the mean value of B10 volume density for the sample less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence (Reference 9).

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than the statistically derived minimum thickness or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness.

Non-conforming material shall be evaluated for acceptance in accordance with Transnuclear 's QA procedures.

A9.7.5 QUALIFICATION TESTING OF METAL MATRIX COMPOSITES

A9.7.5.1 APPLICABILITY AND SCOPE

Prior to initial use in a spent fuel dry storage system, new MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process controls shall be identified per Section A9.7.6 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage system.

ASTM methods and practices are referenced below for guidance. Alternative methods may be used with the approval of Transnuclear .

A9.7.5.2 DURABILITY

There is no need to include accelerated radiation damage testing in the qualification. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for unclad MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842 °F (Reference 10), well above the basket temperature under normal conditions of storage or transport.

Corrosion testing is not required for MMCs (clad or unclad) consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear (Reference 11).

A9.7.5.3 DELAMINATION TESTING OF CLAD MMC

Clad MMCs shall be subjected to thermal damage testing following water immersion to ensure that delamination does not occur under normal conditions of storage.

A9.7.5.4 REQUIRED TESTS AND EXAMINATIONS TO DEMONSTRATE MECHANICAL INTEGRITY

At least three samples, one each from approximately the two ends and middle of the test material production run shall be subjected to:

a) room temperature tensile testing (ASTM- B557 (Reference 12)) demonstrating that the material:

- has a 0.2% offset yield strength no less than 1.5 ksi;
- has an ultimate strength no less than 5.0 ksi; and
- has minimum elongation in two inches no less than 0.5%.

As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290 (Reference 13). The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

b) testing by ASTM-B311 (Reference 14) to verify more than 98% theoretical density for non-clad MMCs and 97% for the matrix of clad MMCs. Testing or examination for interconnected porosity on the faces and edges of unclad MMC, and on the edges of clad MMC shall be performed by a method to be approved by Transnuclear. The maximum interconnect porosity is 0.5 volume %.

And for at least one sample,

c) for MMCs with an integral aluminum cladding, thermal durability testing demonstrating that after a minimum 24 hour soak in either pure or borated water, then insertion into a preheated oven at approximately 825°F for a minimum of 24 hours, the specimens are free of blisters and delamination and pass the mechanical testing requirements described in test 'a' of this section.

A9.7.5.5 REQUIRED TESTS AND EXAMINATIONS TO DEMONSTRATE B10 UNIFORMITY

Uniformity of the boron distribution shall be verified either by:

(a) Neutron radioscopy or radiography (ASTM E94 (Reference 15), E142 (Reference 16), and E545 (Reference 17)) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

(b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section A9.7.4.3, or by chemical analysis for boron carbide content in the composite.

A9.7.5.6 APPROVAL OF PROCEDURES

Qualification procedures shall be subject to approval by Transnuclear.

A9.7.6 PROCESS CONTROLS FOR METAL MATRIX COMPOSITES

This section provides process controls to ensure that the material delivered for use is equivalent to the qualification test material.

A9.7.6.1 APPLICABILITY AND SCOPE

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. Transnuclear shall determine whether a complete or partial re-qualification program per Section A9.7.5 is required, depending on the characteristics of the material that could be affected by the process change.

A9.7.6.2 DEFINITION OF KEY PROCESS CHANGES

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduced density, reduced corrosion resistance, or reduce the mechanical strength or ductility of the MMC.

A9.7.6.3 IDENTIFICATION AND CONTROL OF KEY PROCESS CHANGES

The manufacturer shall provide Transnuclear with a description of materials and process controls used in producing the MMC. Transnuclear and the manufacturer shall identify key process changes as defined in Section A9.7.6.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change.

The following are examples of other changes that are established as key process changes, as determined by Transnuclear's review of the specific applications and production processes:

(a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns, or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit;

(b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering;

(c) Change in the nominal matrix alloy;

(d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for powder metallurgy or thermal spray MMCs that were qualified with extruded material, or a change to direct rolling from the billet;

(e) For MMCs using a magnesium-alloyed aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature;

(f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending; and

(g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

A9.7.7 Radial Neutron Shielding Tests

The shielding performance of the radial polyester resin can be verified adequately by chemical analysis and verification of density. Uniformity is assured by installation process control.

Testing Requirements

Chemical analysis shall be performed on the first batch mixed with a given set of components, and thereafter whenever a new lot of one of the major components is introduced. The acceptance values for the chemical composition of the polyester resin are listed in the following table. Note that the chemical composition used in the shielding models (i.e. listed in Table A7A.4-3) are included in the following table for comparison.

Table A7A.4-3 values		Acceptance Testing Values		
Element	nominal wt %	Element	wt %	acceptance range (wt %)
H	5.05	H	5.05	-10 / +20
B	1.05	B	1.05	± 20
C	35.13	C	35.13	± 20
Al	14.93	Al	14.93	± 20
O	41.73	O+Zn (balance)	43.84	± 20
Zn	2.11			
Total	100.0%		100%	

A density measurement shall be performed on every mixed batch of the polyester resin. The minimum polymer density measured shall be greater than 1.547 g/cm^3.

Process Controls

Qualification tests of the personnel and procedure used for mixing and pouring the polyester resin shall be performed. Qualification testing shall include verification that the chemical composition and density is achieved, and the process is performed in such a manner as to prevent voids.

A9.8 REFERENCES

The references 1 through 3 listed in Section 9.7 are independent of cask design. The new references associated with Section A9 are:

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

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

6. "Aluminum Standards and Data, 2003" The Aluminum Association.

7 ASTM E1225, "Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique"

8. ASTM E1461, "Thermal Diffusivity of Solids by the Flash Method"

9. Natrella, "Experimental Statistics," Dover, 2005.

10. Pyzak and Beaman, "Al-B-C Phase Development and Effects on Mechanical Properties of B_4C/Al Derived Composites," J. Am. Ceramic Soc., 78[2], 302-312 (1995)

11. "Hydrogen Generation Analysis Report for TN-68 Cask Materials," Test Report No. 61123-99N, Rev 0, Oct 23, 1998, National Technical Systems.

12. ASTM B557, "Standard Test Methods of Tension Testing Wrought and Cast Aluminum- and Magnesium-Alloy Products"

13. ASTM E290, "Standard Test Methods for Bend Testing of Material for Ductility"

14. ASTM B311, "Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less Than Two Percent Porosity"

15. ASTM E94, "Recommended Practice for Radiographic Testing"

16. ASTM E142, "Controlling Quality of Radiographic Testing"

17. ASTM E545, "Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing"

18. Thermal Conductivity Measurements of Boron Carbide/Aluminum Specimens, Oct 1998, testing by Precision Measurements and Instruments Corp. for Transnuclear, Inc., Purchase Order Number 98037

19. Eagle Picher Report AAQR06, "Qualification of Thermal Conductivity, Borated Aluminum 1100", May 2001

Table A9.7-1
Thermal Conductivity for Sample Neutron Absorbers

Temperature °C	Material			
	1	2	3	4
20	193	170	194	194
100	203	183	207	201
200	208	-	-	
250	-	201	218	206
300	211	204	220	203
314	-	-	-	202
342	-	-	-	202

Units: W/mK

Materials:
1) Boralyn® MMC, aluminum 1100 with 15% B_4C
2) Borated aluminum 1100, 2.5% boron as TiB_2
3) Borated aluminum 1100, 2.0% boron as TiB_2
4) Borated aluminum 1100, 4.3% boron as AlB_2

Sources:
References 18 and 19

TABLE A9.7-2
SAMPLE DETERMINATION OF THERMAL CONDUCTIVITY ACCEPTANCE CRITERION

Single Plate Model	Al 1100	n absorber	total
thickness (inch)	0	0.437	0.437
conductivity at 70°F (Btu/hr-in-°F)	n/a	9.11	n/a
conductance (Btu/hr-°F)	0	3.98	3.98*

Dual Plate Construction	Al 1100	n absorber	total	
thickness (inch)	0.312	0.125	0.437	as modeled
conductivity at 70°F (Btu/h-.in-°F)	11.09	4.17	n/a	
conductance (Btu/hr-°F)	3.46	0.52	3.98	

thickness (inch)	0.187	0.250	0.437	thicker neutron absorber
conductivity at 70°F (Btu/hr-in-°F)	11.09	7.62	n/a	
conductance (Btu/hr-°F)	2.07	1.91	3.98	

thickness (inch)	0.359	0.078	0.437	thinner neutron absorber
conductivity at 70°F (Btu/hr-in-°F)	11.09	0	n/a	
conductance (Btu/hr-°F)	3.98	0	3.98	

The acceptance criterion is identified by boldface type for each thickness.