TRANSNUCLEAR
AN AREVA COMPANY

December 10, 2009
E-28034

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Transnuclear, Inc. (TN) Application for the TN-40 Transportation Packaging for Spent Fuel, Revision 3, Docket No. 71-9313, TAC No. L24106

Reference: Letter from Meraj Rahimi (NRC) to Robert Grubb (TN), "Second Round Request for Additional Information for Review of the Model No. TN-40 Transportation Package," dated January 29, 2009

This submittal provides responses to the request for additional information (RAI) forwarded by the referenced letter. Enclosures 4 and 5 herein provide each of the NRC staff RAI followed by a TN response for non-proprietary items and proprietary items, respectively. Enclosure 2 lists the RAI items and shows associated changed Safety Analysis Report (SAR) page numbers.

Enclosure 6 provides instructions for SAR page removal and insertion. Enclosure 7 provides changes to the TN-40 SAR. In the SAR, replacement and new pages are annotated as Revision 3, with changed areas indicated by italicized text and revision bars.

In addition to changes related to the RAI, Enclosure 3 describes two cases which also caused changes to the SAR.

This submittal includes proprietary information which may not be used for any purpose other than to support your staff's review of the application. In accordance with 10 CFR 2.390, I am providing an affidavit (Enclosure 1) specifically requesting that you withhold this proprietary information from public disclosure. This submittal also includes security-related information. Accordingly, public versions of TN SAR drawings and pages are provided in Enclosure 8, and public versions of certain RAIs and responses, TN calculations and TN reports are provided in Enclosure 21. Because Enclosure 10 consists of proprietary information not owned by TN, and because Enclosure 18 consists of computer disks that are entirely proprietary, public versions are not provided.

Additional enclosures listed are referenced from within RAI responses.

Should the NRC staff require additional information to support review of this application, please do not hesitate to contact Mr. Don Shaw at 410-910-6878 or me at 410-910-6881.

Sincerely,

Jayant Bondre

Jayant Bondre, PhD
Vice President - Engineering

7135 Minstrel Way, Suite 300, Columbia, MD 21045
Phone: 410-910-6900 • Fax: 410-910-6902

NMSS01

NMSS

cc: Michele Sampson (NRC SFST) (8 copies of this cover letter and Enclosures 1 through 7 and one copy of Enclosures 9 through 20, provided in a separate mailing)

Enclosures:

1. Affidavit Pursuant to 10 CFR 2.390
2. List of RAIs and Affected SAR Pages
3. Additional, non-RAI Items Addressed in this Submittal
4. RAI Responses (non-proprietary items)
5. RAI Responses (proprietary items)
6. SAR Page Replacement Instructions
7. Changed Pages and Drawings for the TN-40 Application Safety Analysis Report, Revision 3
8. Public Versions of Changed Drawings, Appendix 2.10.7 Pages, and Chapter 6 Pages, for the TN-40 Application Safety Analysis Report, Revision 3
9. TN Technical Report No. E-25768, Rev. 0, "Evaluation of Creep of NUHOMS Basket Aluminum Components under Long Term Storage Conditions," November 2007 (associated with RAI 2-2) (proprietary)
10. XN-NF-78-34 and XN-NF-83-87 (associated with RAI 1-1) (proprietary)
11. Six Pages from XN-NF-79-67 (associated with RAI 1-1)
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14. Transnuclear, Inc. Calculation 10421-038 (associated with RAI 2-4)
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21. Public versions of Enclosures 5, 9, 15, 16 and 17

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. (TN), 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 5, 7, 9, 10, 15, 16, 17, and 19, as listed below:

1. Enclosure 5, proprietary NRC Request for Additional Information (RAI) items and responses,
2. Enclosure 7, certain changed pages for the TN-40 application Safety Analysis Report,
3. Enclosure 9, TN Technical Report No. E-25768, Rev. 0, "Evaluation of Creep of NUHOMS Basket Aluminum Components under Long Term Storage Conditions," November 2007,
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5. Enclosure 15, Transnuclear, Inc. Calculation 10421-024, Revision 2,
6. Enclosure 16, Transnuclear, Inc. Calculation 10421-023, Revision 2,
7. Enclosure 17, Transnuclear, Inc. Calculation 10421-051, Revision 0,
8. Enclosure 19, computer input and output files.

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 involves proprietary RAIs and responses, portions of certain TN-40 spent fuel transportation cask SAR chapters and design analyses, plus certain computer files associated with those analyses, a TN technical report assessing the aging of certain system components, 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 transportation system.

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   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
Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 10th day of December, 2009.

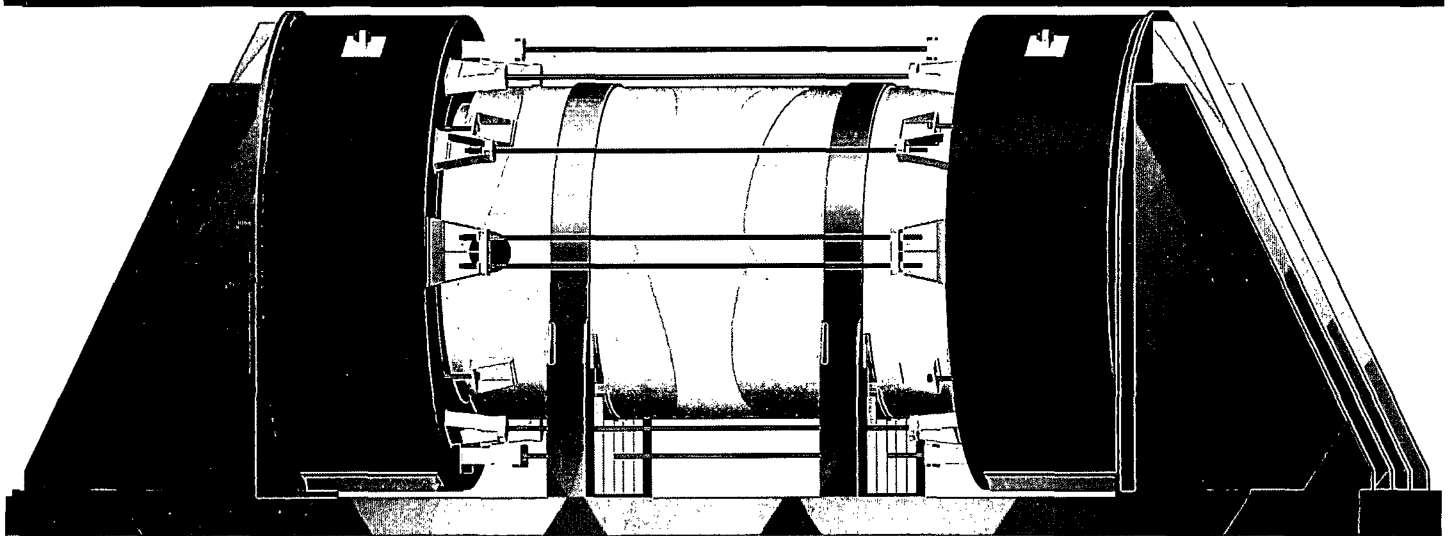
Arleastrice M. Taylor
Notary Public

My Commission Expires 10 / 14 / 2012

ARLEASTRICE M. TAYLOR NOTARY PUBLIC

NON-PROPRIETARY

TN-40 TRANSPORTATION PACKAGING



SAFETY ANALYSIS REPORT

AREVA

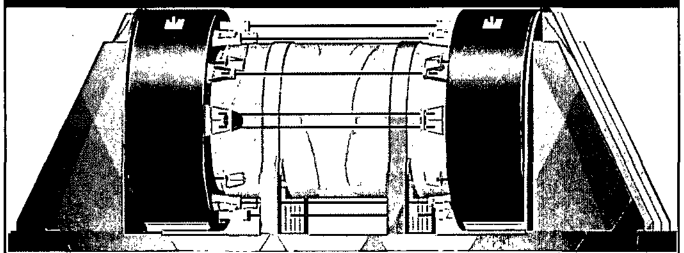
TRANSNUCLEAR, INC.

E-23861

NON- PROPRIETARY

TN-40

TRANSPORTATION PACKAGING



SAFETY ANALYSIS REPORT

AREVA

TRANSNUCLEAR, INC.

E-23861

NON-PROPRIETARY

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TN-40

TRANSPORTATION PACKAGING

SAFETY ANALYSIS REPORT

Revision 3

December 2009

7135 Minstrel Way, Suite 300 • Columbia, MD 21045

TRANSNUCLEAR
AN AREVA COMPANY

December 10, 2009
E-28034

U. S. Nuclear Regulatory Commission
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Vice President - Engineering

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Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 10th day of December, 2009.

Notary Public

My Commission Expires 10 / 14 / 2012

TN-40 RAI No. 2 – RAIs and Associated SAR Pages

Associated RAI(s)	SAR Change?	SAR Page
1-1	yes	1-8
1-2	yes	1-7 1-8
1-3	no	n/a
1-4	no	n/a
2-1	no	n/a
2-2	no	n/a
2-3	no	n/a
2-4	no	n/a
2-5	yes	2-29 2-30 2-31 2.10.9-21
2-6	yes	2-29 2-30 2-31 2.10.9-21
2-7	yes	2.10.5-11C 2.10.5-11D 2.10.5-11E 2.10.5-11F 2.10.5-12 2.10.5-58 2-49
2-8	no	n/a
2-9	no	n/a
2-10	no	n/a
2-11	no	n/a
2-12	yes	2.10.5-4
2-13	yes	See RAI 2-7
2-14	yes	2.10.5-54
2-15	yes	2.10.5-43
2-16	no	n/a

TN-40 RAI No. 2 – RAIs and Associated SAR Pages

Associated RAI(s)	SAR Change?	SAR Page
2-17 through 2-20	yes	2.10.7-i 2.10.7-ii 2.10.7-3 2.10.7-4 2.10.7-5 2.10.7-6 2.10.7-6a 2.10.7-6b 2.10.7-7 2.10.7-7a 2.10.7-9 2.10.7-13 2.10.7-15 2.10.7-16 2.10.7-22 2.10.7-23 2.10.7-24 2.10.7-25 2.10.7-26 2.10.7-27 2.10.7-28 2.10.7-29 2.10.7-30 2.10.7-31 2.10.7-33 2.10.7-34 2.10.7-35 2.10.7-36
3-1	no	n/a
3-2	yes	3.7.1-5 3.7.1-6 3.7.1-7 3.7.1-8 3.7.1-9 3.7.1-10 3.7.1-11 3.7.1-12 3.7.1-13 3.7.1-14 3.7.1-15
4-1	yes	4-4
5-1	yes	1-4 1-4a

TN-40 RAI No. 2 – RAIs and Associated SAR Pages

Associated RAI(s)	SAR Change?	SAR Page
5-2	yes	5-4 5-56 5-57 5-58 5-59
5-3	yes	5-6 through 5-8, 5-66C
5-4	yes	5-51
5-5	yes	Drawing 10421-71-1 Drawing 10421-71-2 Drawing 10421-71-3 5-1 5-8 5-9A 5-9B 5-10 5-66A 5-69 5-70
5-6	yes	5-1 5-2 5-9B 5-10 5-66A
5-7	yes	1-7 1-8
5-8	yes	5-7 5-9B 5-51 5-66A
5-9	yes	5-1 5-2 5-5 5-6 5-7 5-8 5-9B 5-9C 5-51 5-56 5-58 5-66A 5-66B

TN-40 RAI No. 2 – RAIs and Associated SAR Pages

Associated RAI(s)	SAR Change?	SAR Page
5-10	yes	5-9D 5-9E 5-10
All Chapter 6 RAIs, as a group	yes	6-2 6-3 6-5 6-6 6-7 6-8 6-9 6-10 6-11 6-17 6-18 6-18B 6-21 6-50 6-52 6-53 6-54 6-55 6-96
6-1	yes	6-12 6-58 6-59 6-60 6-61 6-62 6-63 6-64 6-65 6-66 6-67 6-68 6-69
6-2	yes	6-12 6-69

TN-40 RAI No. 2 – RAIs and Associated SAR Pages

Associated RAI(s)	SAR Change?	SAR Page
6-3	yes	6-12 6-13 6-14 6-15 6-16 6-18C 6-23 6-24 6-25 6-30 6-31 6-32 6-33 6-33A 6-33B 6-33C 6-33D 6-69 6-70 6-71 6-71A through 6-71L 6-72 6-79 6-80 6-81 6-82 6-83 6-95 6-113 6-114 6-115 6-116 6-117
6-4	yes	6-21
6-5	yes	6-26 6-27 6-85 6-118

TN-40 RAI No. 2 – RAIs and Associated SAR Pages

Associated RAI(s)	SAR Change?	SAR Page
6-6	yes	6-33 6-33A 6-33B 6-33E 6-33F 6-33G 6-93 6-94 6-95 6-95A 6-95B 6-95C 6-95D 6-95F 6-95G 6-95H
6-7	yes	6-33 6-33A 6-33B 6-33E 6-33F 6-33G 6-91 6-92 6-92A 6-92B 6-92C 6-92D 6-92E 6-93 6-94 6-95 6-95A 6-95B 6-95C 6-95D 6-95F 6-95G 6-95H
7-1	yes	7-7
7-2	yes	Drawing 10421-71-2
7-3	yes	7-9 7-10

TN-40 RAI No. 2 – RAIs and Associated SAR Pages

Associated RAI(s)	SAR Change?	SAR Page
7-4	yes	7-1 7-2 7-11
8-1	yes	8-1
8-2	yes	8-2 8-8
8-3	yes	1-1 7-13
8-4	no	n/a
8-5	yes	3-14 3-14A 8-5
8-6	yes	1-1 8-7
8-7	yes	1-1 8-7

Additional, non-RAI Items Addressed in this Submittal

Subject	Discussion	Changed SAR Pages
Fire Emissivity	Recent discussions occurred between NRC staff and TN regarding fire emissivity and fire temperature values used to meet 10 CFR 71.73 (c)(4), as they pertained to TN-40. TN has performed a sensitivity analysis to consider the effects of increasing the fire emissivity to 1.0 and included resultant SAR changes in this submittal.	3-15 3-20 3.7.3-i 3.7.3-1 3.7.3-2 3.7.3-3 3.7.3-4
Leak Rate Criteria	Leak rate inconsistencies noted by the Staff in Section 4.4 and 8.1.3 have been resolved. All leak rates now are in terms of ref-cm^3/s. As defined in ANSI 14.5, this is understood as an air leakage rate at reference conditions of 25°C and 1 atm abs.	4-13 8-2a
Fuel End Drop Effect on Criticality	A new Section 6.4.2-D is included to determine the effect based on structural analysis results in Appendix 2.10.7.2.3	6-18A 6-27 6-28 6-29 6-32 6-33 6-34 6-84 6-95 6-95E 6-119

1.0 GENERAL INFORMATION

1-1 Place the fuel assembly data back in Chapter 1, and provide copies of the references, or the NRC Agency Document and Management System (ADAMS) accession number, for the four references cited in the footnote to the table in the response to the first round RAI 1-7.

Three of these documents had been produced in the 70's and early 80's. The staff could only locate one of them (i.e., XN-NF-78-34). Even then, the guide tube inside diameter and outside diameter for Exxon STD fuel assembly, indicated by TN to be in Table 2.1 of XN-NF-78-34 report, are not there. Generally, when an applicant references a document, it should be readily available to the staff. Otherwise, the applicant should provide the references. The staff asked in the first RAI for TN to verify some of the fuel assembly data and provide copies of the references. Instead, TN deleted all the fuel assembly data from Chapter 1 and provided only the titles of the references. TN needs to retain the fuel assembly data table, which reflects the assembly parameter values used in the structural, thermal, containment, shielding, and criticality calculations, in Chapter 1, and provide copies of the references.

This information is required for compliance with 10 CFR 71.33(a)(5).

Response to 1-1

The table containing fuel assembly data that was removed from Chapter 1 of the SAR has been returned to Section 1.2.3. In addition, a summary table has been added that provides summary fuel assembly physical data regarding the contents of packaging. The physical data is not fuel-type dependent. Material added to Section 1.2.3 in Rev. 2 of the SAR has been retained.

As explained below, three of the requested references are included as enclosures herein and an accession number is provided for the fourth.

The following items are included as Enclosure 10:

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

The following item is included as Enclosure 11:

- *Six pages from XN-NF-79-67 (NP), "Prairie Island Unit 2 Nuclear Plant Cycle 5 Safety Analysis Report," August 1979.*

For the following item, please use ADAMS Accession Number ML053390121:

- *WCAP-16517-NP, "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis," November 2005.*

For informational purposes, the table from the first round RAI 1-7 response is provided here, with values added to the table along with the references.

	Exxon STD	Exxon High Burnup	Exxon TOPROD	West STD	WEST OFA
MTU/Assembly	*0.380*	*0.380*	*0.380*	*0.410*	*0.380*
	These values originate from the NRC-741 Forms associated with the shipment of the fuel to Prairie Island.				
Rod Pitch	*0.556*	*0.556*	*0.556*	*0.556*	*0.556*
	Table 2.1 of Report XN-NF-78-34	*Table 2.1 of Report XN-NF-78-34, (see Sections 4.0 and 7.2 of Report XN-NF-79-67)*	*Table 4.1 of Report XN-NF-83-87*	*Table 3-1 of WCAP 16517-NP*	*Table 3-1 of WCAP 16517-NP*
Pellet OD	*0.3565*	*0.3565*	*0.3505*	*0.3659*	*0.3444*
	Not Requested	*Not Requested*	*Table 4.1 of XN-NF-83-87*	*Not Requested*	*Not Requested*
Guide tube ID	*16@0.507*	*16@0.507*	*16@0.507*	*16@0.505*	*16@0.490*
	Table 2.1 of XN-NF-78-34	*Table 2.1 of Report XN-NF-78-34, (see Sections 4.0 and 7.2 of Report XN-NF-79-67)*	*Table 4.1 of XN-NF-83-87*	*Table 3-1 of WCAP 16517-NP*	*Table 3-1 of WCAP 16517-NP(2)*
Guide Tube OD	*16@0.541*	*16@0.541*	*16@0.541*	*16@0.539*	*16@0.528*
	Table 2.1 of XN-NF-78-34	*Table 2.1 of Report XN-NF-78-34, (see Sections 4.0 and 7.2 of Report XN-NF-79-67)*	*Table 4.1 of XN-NF-83-87*	*Table 3-1 of WCAP 16517-NP*	*Table 3-1 of WCAP 16517-NP(2)*
Clad Thickness	*0.0300*	*0.0310*	*0.02950*	*0.0243*	*0.0243*
	Not Requested	*Not Requested*	*Not Requested*	*Table 3-1 of WCAP 16517-NP*	*Not Requested*

XN-NF-78-34 (P)(3) "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.

XN-NF-79-67 (NP)(3) "Prairie Island Unit 2 Nuclear Plant Cycle 5 Safety Analysis Report, August 1979.

XN-NF-83-87(3) "Mechanical Design Report Supplement for Margin Upgrade of Prairie Island Units 1 and 2 TOPROD Fuel" October 1983

WCAP-16517-NP(1) "Prairie Island Units 1 & 2 Spent Fuel Pool Criticality Analysis November 2005"

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)"*
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). Prairie Island Nuclear Generating Plant 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. *The Exxon Reports contain a "Nuclear Regulatory Commission Disclaimer" that reads in part: "...It is being submitted by Exxon Nuclear to the USNRC as part of a technical contribution to facilitate safety analyses by licenses of the USNRC which utilize Exxon Nuclear for light water power reactors ...". Therefore the NRC should already have copies of these reports.*

1-2 Analyze for the criticality, radiological safety, and normal handling effects of specific fuel assembly defects that are considered as undamaged. The location of this analysis in the SAR should be stated. The definition of damaged fuel, along with a statement that the approval is only for transportation of undamaged Prairie Island fuel at BU <45 GWd/MTU should be put on the proposed CoC.

Without stating such, bullets 3 and 4 of the SAR, Section 1.2.3 definition of undamaged fuel, are invoking Interim Staff Guidance (ISG) -1, Rev. 2, "CLASSIFYING THE CONDITION OF SPENT NUCLEAR FUEL FOR INTERIM STORAGE AND TRANSPORTATION BASED ON FUNCTION," but the definition is incomplete. The definition of undamaged fuel should be in the CoC so only approved contents are transported.

This information is required for compliance with 10 CFR 71.55(d)(1) and (2).

Response to 1-2

It is not the intention to license the TN-40 cask to transport damaged fuel assemblies and thus Section 1.2.3 was written such that the contents of the TN-40 packaging would be limited to unconsolidated UNDAMAGED FUEL ASSEMBLIES. Rather than perform the assessments outlined in Interim Staff Guidance (ISG) -1 Revision 2, it was the intent to utilize the "default" definition of damaged Spent Nuclear Fuel contained in ISG-1 Revision 2 as the basis for developing the definition of UNDAMAGED FUEL ASSEMBLIES. However, based on the RAIs, it appears that development of the definition of UNDAMAGED FUEL ASSEMBLIES was not successful in meeting the guidance contained in ISG-1 Revision 2.

The intention of the Safety Analysis Report is to allow transport of fuel assemblies that are currently stored within the TN-40 cask under License SNM-2506. Thus Section 1.2.3 of the Safety Analysis

Report has been revised to limit the contents of the TN-40 packaging to fuel that is not DAMAGED and to replace the definition of UNDAMAGED FUEL ASSEMBLIES with the definition of DAMAGED Spent Nuclear Fuel submitted to the NRC as part of the SNM-2506 License Amendment Request to Modify the TN-40 Cask Design, i.e., the TN-40HT LAR supplement dated June 26, 2009. To avoid unintended consequences, the wording of this definition is as consistent as possible with the wording in the original Technical Specifications appended to License SNM-2506.

1-3 Add a statement to the proposed CoC requiring the evaluation of storage records for indications of air ingress.

Determination of air ingress during storage is necessary to determine the condition of the fuel and its acceptability as content for transport.

This information is required for compliance with 10 CFR 71.33(b)(3).

Response to 1-3

A requirement will be included in the CoC for the package to be prepared for shipment in accordance with the Operating Procedures in Chapter 7 of the application (Safety Analysis Report). Since Step 7.4.1.2 in Chapter 7 already includes a review of maintenance records for situations where air may have leaked into the cask while it was in its storage configuration, this action would be incorporated into the CoC by reference.

1-4 Specify the reasons for the restrictions on Unit 1 Region 4 fuel assemblies.

It's not clear why assemblies D-01 through D-40 are not allowed to be loaded into TN-40.

This information is required for compliance with 10 CFR 71.33(b)((2).

Response to 1-4

During the Prairie Island Unit 1 cycle 4-5 refueling outage, abnormal fuel rod bowing was observed on several of the region 4 fuel assemblies (see Licensee Event Report P-RO-79-12). The cause of the abnormal bowing was attributed to high values of clad Wall Thickness Variation (WTV) in four tubing lots from a single ingot of cladding material. Since the time the Prairie Island Unit 1 region 4 tubing was fabricated, a further tightening of the WTV specification has been implemented.

The abnormal rod bowing is severe enough such that these fuel assemblies from this entire region of fuel would not be bounded by the fuel rod drop structural analyses and the modeling assumptions employed in the criticality analyses. Therefore, it was decided to explicitly eliminate them from the approved contents for transport in a TN-40 package.

2.0 STRUCTURAL

Section 1.2.3 Undamaged Fuel Assemblies

2-1 Perform a structural integrity evaluation of the fuel rod cladding subject to the 30-ft end-drop accidents by considering the "undamaged" fuel assembly configurations characterized with: (1) uniform rod bowing and (2) missing, displaced, or damaged structural components that can still be handled with normal means.

The application 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 application to demonstrate its acceptability.

This information is needed to determine compliance with 10 CFR 71.35(a), 71.55(d)(1), and 71.55(d)(2).

Response to 2-1

Please see the response to RAI 1-2.

2-2 Provide a copy of the Reference 1 cited in response to first round RAI 2-3, "TN Technical Report No. E-25768, Rev. 0, "Evaluation of Creep of NUHOMS Basket Aluminum Components under Long Term Storage Conditions," November 2007."

This information is required for compliance with 10 CFR 71.33.

Response to 2-2

A copy of the requested report is included in this submittal as Enclosure 9.

Section 2.7 Reporting Method for Cask Body Stresses

2-3 Clarify if the ASME Code was followed in calculating stress intensity as the total stress?

With respect to the statements, "Two or more individual load cases must be combined to determine the total stresses at *any* stress reporting locations for the various load combinations. This is accomplished using the ANSYS post-processor." Ascertain and confirm to the staff that the American Society of Mechanical Engineers (ASME) NB-3215, provisions is followed for the stress intensity derivation. If not followed, provide justification for conservative implementation of the method and identify it also as a code alternative in Section 2.11 of the application.

The above information is needed to meet the requirements of 10 CFR 71.7(a) and 71.35(a).

Response to 2-3

The ANSYS post-processor follows the same procedure for combining the normal and shear stress components due to individual loads into a load combination case (in a global or a defined *coordinate system) and computing the stress intensities as required in ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Para NB-3215.*

As described in ANSYS User's Manual Volume II Commands (command PRSECT), the combined stress components are linearized along a section path in separate membrane, bending, membrane plus bending and peak stress categories at the beginning, mid-length, and end of the section. The section is defined by a path consisting of two end points (nodes) and 47 intermediate points (automatically determined by linear interpolation in active coordinate system). The values of the component stresses to be linearized are interpolated at the path points within each path element from the element's average corner nodal values. Stress components through the section are linearized by a line integral method and separated into constant membrane stresses, bending stresses varying linearly between end points, and peak stresses (defined as the difference between the actual (total) stress and the membrane plus bending combination).

For each category of linearized stress components, principal stresses, stress differences and stress intensities are computed as described in NB-3215.

The above procedure has been followed in combining the nodal stress components of individual loads in all load combination cases and obtaining the linearized stress intensities reported in the SAR.

Section 2.7.1 30-Foot Free Drop

The following information is needed to meet the requirements of 10 CFR 71.35(a) or 71.73(c) unless otherwise stated.

2-4 Provide the Acceleration Due to drop On Covers (ADOC) code validation and verification documents, as appropriate, to demonstrate that ADOC can be implemented adequately for determining cask rigid-body decelerations in a slapdown drop event.

The two tables on page 2-28, which summarize the baseline cask decelerations, indicate that the primary impact is more severe than the secondary impact with the transverse decelerations of 39 g and 27 g, respectively. The decelerations appear to be calculated from the reported maximum impact limiter reaction forces of 12,209 kips and 7,526 kips, respectively. These calculated responses defy the common observation that the secondary impact tends to be more severe than the primary because of the cask rotation added terminal velocity of the tail end impact limiter upon its landing on the target.

Response to 2-4

Copies of the original Acceleration Due to Drop on Covers (ADOC) validation and verification reports are included herein, as described below:

- *Enclosure 12, E-10004, ADOC Test Report, 8/21/1997.*
- *Enclosure 14, TN calculation 10421-038, Verification of ADOC Program for Windows XP Operating System, 3/29/2006.*

2-5 In the 4th paragraph from the top of page 2-29, revise the statement, "It can be concluded that the slapdown transverse g-loads (both first and second impact) are less than the transverse g-loads from the side drop case, therefore the side drop transverse acceleration is used for basket and fuel rod side drop analysis," by recognizing that: (1)

the measured slapdown transverse deceleration of 61 g is higher than the measured side drop deceleration of 57 g, and (2) the "periodic basket" finite element analysis model, which is different from the "complete cask body" model, must consider the at-section vector sum of both the transverse and rotational deceleration components of the slapdown event.

The ADOC results appear to be inconsistent in that the basket transverse-impact g-loads would have been calculated as 90 g (39 + 51 = 90 and 76 g (27 + 49 = 76) for the primary and secondary impacts, respectively.

Response to 2-5

The statements and tables on pages 2-29, 2-30, and 2-31 have been revised to recognize that for the slap down drop case, the second impact (combined the transverse g load and rotational g load) will have a more severe impact on the components than the first impact. Therefore the reported g load for the slap down is based on the second impact. These combined g loads are higher than the side drop g loads; therefore, for the basket and fuel rod side drop analyses, the applied g load must bound both side drop and slap down g loads.

2-6 Considering the two questions above, revise the listed slap-down baseline g-loads in page 2-30 by recognizing that (1) the slapdown event must be based on the secondary impact, which would be seen as more severe than the primary impact, and (2) the g-loads summarized in the table at the page bottom must include those of the slapdown event.

The staff notes that the slapdown test and ADOC analysis results may not have been adequately and consistently correlated for developing the baseline g loads for analyzing the cask body and fuel basket.

Response to 2-6

Please see the response to RAI 2-5.

Section 2.11 ASME Code Alternatives

2-7 As a code alternative, add the fuel basket cell wall load limit tests as a supplement to the ASME, Section III, Appendix F-1341.4, plastic instability load analysis provisions for demonstrating structural stability of the basket under the 30-ft free drop accidents.

The accuracy of the plastic instability load analysis can be sensitive to modeling assumptions, including boundary and interface conditions between the stainless steel and aluminum plates and their respective strain-hardening rates. In recognizing potential uncertainties in the analysis, Appendix 4C to the Prairie Island ISFSI (SAR) performed load limit tests to support the TN-40 basket evaluation (Docket 72-10). This was done by testing representative TN-40 basket cell wall panels to supplement the plastic instability load analysis of the basket subject to side-impact g-loads. As previously reviewed, since the baseline slapdown g-loads can be markedly higher than that considered in the present analysis, the calculated structural stability margins would appear to be diminishing. Thus, load limit tests such as those for licensing the Prairie Island TN-40 storage cask must be used in conjunction with the analysis to demonstrate the basket structural acceptability.

Response to 2-7

The description and demonstration that the basket structural acceptability based on the fuel cell wall load limit test report provided as Appendix 4C to the TN-40 Storage SAR and as Enclosure 20 of this submittal [1] has been added to the Transport SAR Appendix 2.10.5, Section 2.10.5.5.3. It concludes that the actual basket panel can take up to 130g (90° drop at 529°F material property) before the panel reaches the buckling load limit. Use of this compression test is included in Chapter 2, Section 2.11, as ASME code alternatives.

[1] Scavuzzo, R. J., Lam, P. C., Gau, J. S., "Buckling Tests of Fusion Welded Composite Stainless Steel Aluminum Plates," Dept. of Mechanical Engineering, The University of Akron, May, 1990

Table 2-19 Linearized Stress Evaluation for Accident Condition

2-8 Verify that the cross section is appropriately selected for evaluating the maximum stress intensity for the containment boundary shell flange for the load combination Case A12, which includes the cold, 30-ft slapdown drop event.

The cross section defined by nodes No. 3920 and No. 5434 is comprised of an unusually large surface area free of stress or traction. As such, it does not appear to be the most critical cross section being screened for stress evaluation purpose.

Response to 2-8

The cross section for evaluating the maximum stress intensity for the containment boundary shell flange for load combination Case A12 (SAR Table 2-19) is appropriately selected. This cross section is defined by Nodes No. 3920 and No. 5434. The justification for this selection is given below.

The nodal stress intensity distribution in the shell flange for load combination Case A12 is shown in Figure 2-8.1 below. This load combination includes the cold, 30-ft slapdown drop on lid event. The figure shows that the maximum nodal stress intensity of 51.35 ksi occurs at node number 3920 located at the flange inner diameter corner. Flange node numbers at a section through 3920 are shown in Figure 2-8.2. The two cross sections from the highest stress intensity node 3920 shown in Figure 2-8.2 are considered bounding. One cross section is defined by nodes No. 3920 and No. 5434 and other cross section is defined by nodes No. 3920 and No. 3899. Stresses are linearized at both of the cross sections. The stress linearizing details at these two cross sections is given in Tables 2-8.1 and 2-8.2 below. A summary of stress intensities is given below:

Cross Section No.	Max. Nodal Stress Intensity (ksi)	Node Numbers	Linearized Stress Intensity	
			Type	Magnitude (ksi)
1	51.35	3920-5434	P_M	22.55
			$P_L + P_B$	39.77
2	51.35	3920-3899	P_M	14.65
			$P_L + P_B$	26.31

It is seen from above that cross section 1, defined by nodes No. 3920 and No. 5434, has the higher stresses and was therefore appropriately selected for reporting shell flange stresses in SAR.

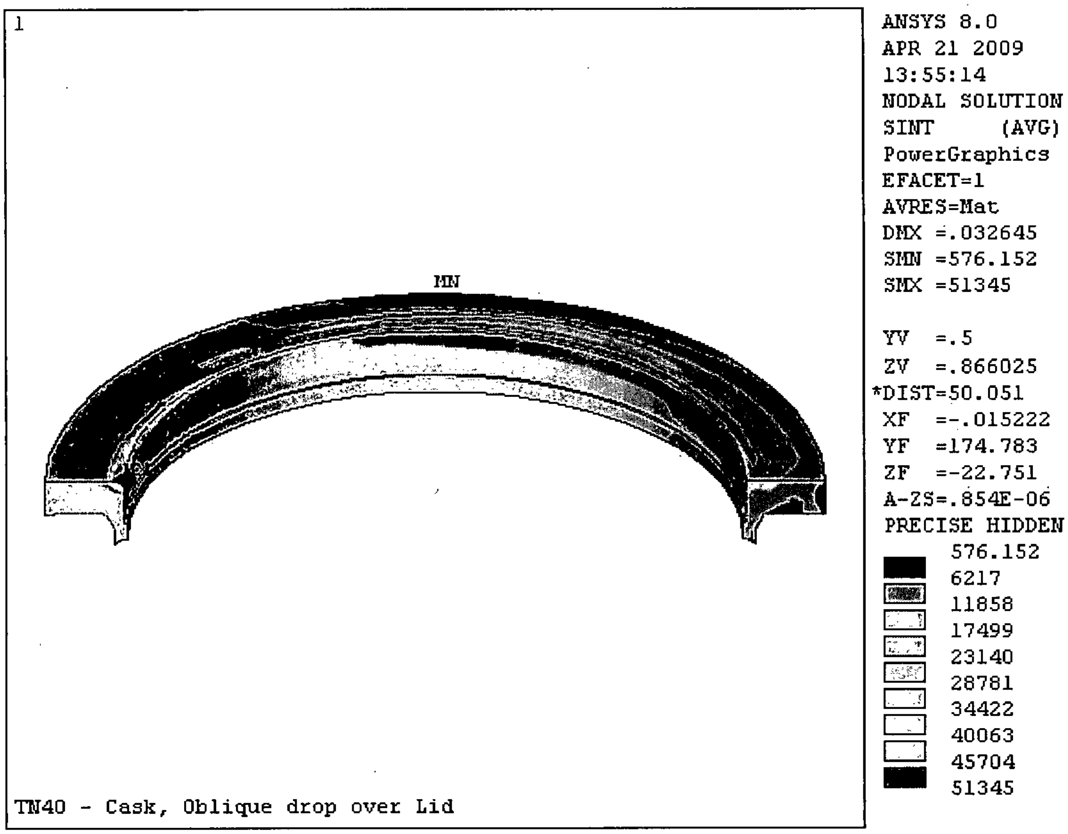


Figure 2-8.1 – TN 40 Shell Flange, Nodal Stress Intensity Distribution for A12 (Cold)

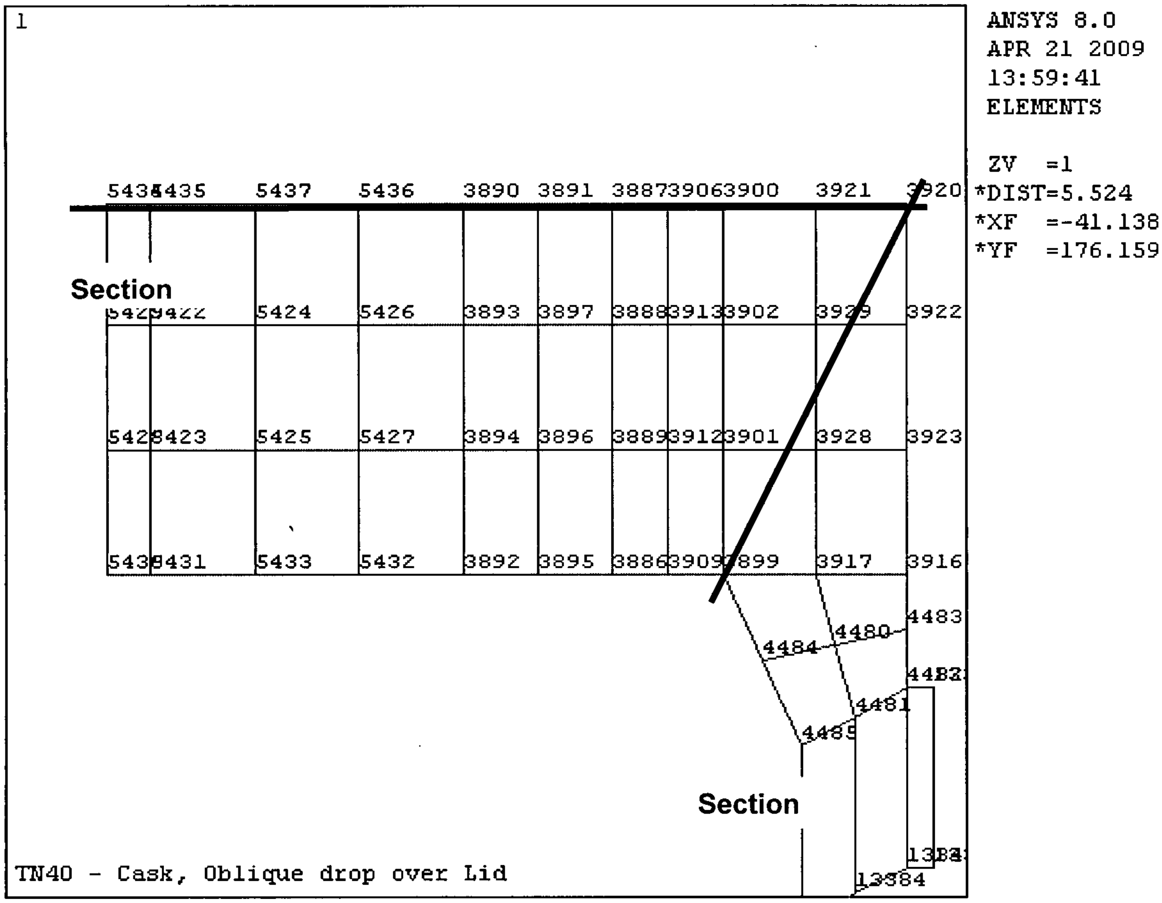


Figure 2-8.2 – TN 40 Shell Flange, Cross Section Locations

Table 2-8.1 – Linearized Stresses Shell Flange, Section 1

```
         ***** POST1 LINEARIZED STRESS LISTING *****
        INSIDE NODE =   3920     OUTSIDE NODE =   5434

 LOAD STEP     0  SUBSTEP=     0
 TIME=   0.0000          LOAD CASE=  8

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

             ** MEMBRANE **
     SX          SY          SZ          SXY          SYZ          SXZ
-0.2558E+05  -8119.     -0.2954E+05  -2761.      -522.2       1751.
     S1          S2          S3          SINT         SEQV
 -7665.     -0.2536E+05 -0.3021E+05  0.2255E+05  0.2055E+05

             ** BENDING **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY          SZ          SXY          SYZ          SXZ
I -0.2521E+05  -6216.      -6562.      -4938.      -1052.      -375.7
C   0.000       0.000       0.000       0.000       0.000       0.000
```

```
O  0.2521E+05   6216.        6562.        4938.        1052.        375.7
     S1          S2           S3           SINT         SEQV
I  -4569.       -6983.      -0.2643E+05  0.2186E+05   0.2076E+05
C   0.000        0.000        0.000        0.000        0.000
O  0.2643E+05   6983.        4569.       0.2186E+05   0.2076E+05

            ** MEMBRANE PLUS BENDING **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY           SZ           SXY          SYZ          SXZ
I -0.5078E+05 -0.1434E+05 -0.3610E+05  -7699.       -1574.        1376.
C -0.2558E+05  -8119.     -0.2954E+05  -2761.       -522.2        1751.
O  -370.7      -1903.     -0.2298E+05   2177.        529.4        2127.
     S1          S2           S3           SINT         SEQV
I -0.1264E+05 -0.3617E+05 -0.5241E+05  0.3977E+05   0.3464E+05
C  -7665.     -0.2536E+05 -0.3021E+05  0.2255E+05   0.2055E+05
O   1343.      -3413.     -0.2318E+05  0.2452E+05   0.2252E+05

            ** PEAK **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY           SZ           SXY          SYZ          SXZ
I -0.1689E+05  -9445.       -7380.       -5573.        789.9       -2168.
C  0.1308E+05   4415.        3373.        941.2        536.4        964.1
O -0.1356E+05  -3679.       -4165.       -2289.       -303.9       -1014.
     S1          S2           S3           SINT         SEQV
I  -5084.      -8573.     -0.2006E+05  0.1498E+05   0.1358E+05
C  0.1328E+05   4467.        3116.      0.1016E+05   9561.
O  -3168.      -4061.     -0.1418E+05  0.1101E+05   0.1059E+05

            ** TOTAL **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY           SZ           SXY          SYZ          SXZ
I -0.6768E+05 -0.2378E+05 -0.4348E+05 -0.1327E+05  -783.9       -792.1
C -0.1250E+05  -3704.     -0.2616E+05  -1820.       14.18        2715.
O -0.1393E+05  -5582.     -0.2714E+05  -112.7       225.5        1113.
     S1          S2           S3           SINT         SEQV         TEMP
I -0.2007E+05 -0.4346E+05 -0.7141E+05  0.5134E+05   0.4452E+05   0.000
C  -3330.     -0.1235E+05 -0.2669E+05  0.2336E+05   0.2040E+05
O  -5579.     -0.1384E+05 -0.2724E+05  0.2166E+05   0.1893E+05   0.000
```

Table 2-8.2 – Linearized Stresses Shell Flange, Section 2

```
        ***** POST1 LINEARIZED STRESS LISTING *****
       INSIDE NODE =    3920      OUTSIDE NODE =    3899

 LOAD STEP      0   SUBSTEP=      0
 TIME=     0.0000            LOAD CASE=  8

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

            ** MEMBRANE **
     SX          SY           SZ           SXY          SYZ          SXZ
-0.1117E+05 -0.1194E+05 -0.2251E+05  -3318.       -1029.       1233.
     S1          S2           S3           SINT         SEQV
 -8041.     -0.1490E+05 -0.2269E+05  0.1465E+05   0.1269E+05

            ** BENDING **  I=INSIDE C=CENTER O=OUTSIDE
     SX          SY           SZ           SXY          SYZ          SXZ
I -0.2131E+05  -5605.       -9306.       -7465.       -35.62       -519.7
C   0.000       0.000        0.000        0.000        0.000        0.000
O  0.2131E+05   5605.        9306.        7465.        35.62        519.7
     S1          S2           S3           SINT         SEQV
I  -2618.      -9293.     -0.2430E+05  0.2169E+05   0.1924E+05
C   0.000       0.000        0.000        0.000        0.000
```

```
O  0.2430E+05   9293.       2618.       0.2169E+05  0.1924E+05

             ** MEMBRANE PLUS BENDING **  I=INSIDE C=CENTER O=OUTSIDE
      SX          SY          SZ          SXY         SYZ         SXZ
I -0.3248E+05 -0.1755E+05 -0.3182E+05 -0.1078E+05  -1064.       712.8
C -0.1117E+05 -0.1194E+05 -0.2251E+05  -3318.      -1029.       1233.
O  0.1013E+05  -6340.     -0.1321E+05   4147.      -993.2       1752.
      S1          S2          S3          SINT        SEQV
I -0.1182E+05 -0.3190E+05 -0.3813E+05  0.2631E+05  0.2382E+05
C  -8041.     -0.1490E+05 -0.2269E+05  0.1465E+05  0.1269E+05
O  0.1121E+05  -7027.     -0.1359E+05  0.2480E+05  0.2226E+05

             ** PEAK **  I=INSIDE C=CENTER O=OUTSIDE
      SX          SY          SZ          SXY         SYZ         SXZ
I -0.3520E+05  -6231.     -0.1166E+05  -2490.       280.6      -1505.
C  0.1021E+05   410.6       3014.       1250.      -120.7       457.2
O -0.1430E+05  -971.5      -4523.      -2117.       215.8      -620.3
      S1          S2          S3          SINT        SEQV
I  -5989.     -0.1160E+05 -0.3550E+05  0.2951E+05  0.2714E+05
C  0.1039E+05   3000.       242.2      0.1015E+05   9088.
O  -619.0      -4514.     -0.1466E+05  0.1405E+05  0.1256E+05

             ** TOTAL **  I=INSIDE C=CENTER O=OUTSIDE
      SX          SY          SZ          SXY         SYZ         SXZ
I -0.6768E+05 -0.2378E+05 -0.4348E+05 -0.1327E+05  -783.9      -792.1
C  -967.4     -0.1153E+05 -0.1950E+05  -2068.      -1150.       1690.
O  -4171.      -7311.     -0.1773E+05   2030.      -777.5       1132.
      S1          S2          S3          SINT        SEQV        TEMP
I -0.2007E+05 -0.4346E+05 -0.7141E+05  0.5134E+05  0.4452E+05   0.000
C  -392.9     -0.1184E+05 -0.1977E+05  0.1937E+05  0.1687E+05
O  -3143.      -8160.     -0.1791E+05  0.1477E+05  0.1300E+05   0.000
```

Appendix 2.10.4 Fracture Toughness Evaluation of the TN-40 Cask

The following information is needed to determine compliance with 10 CFR 71.73(c)(1),(2), and (3), and 10 CFR 71.33 unless otherwise stated.

2-9 Provide justification that the testing, using a limited combination of potential base metals, filler materials, and weld techniques (2 tests), bounds the worst case fracture toughness expected from all potential combinations of these three parameters. Explain why TN fabricators chose the combination of weld processes, electrodes, and base material presented in the table to demonstrate the toughness of the weld and heat affected zone (HAZ). Defend why the table data are representative of all other possible combinations which can be used, or are these data the best case scenario?

Response to 2-9

Twenty nine TN-40 casks have been fabricated. Twenty seven have been completed and the two currently in production have progressed to a point where the Shield Shell to Bottom Shield welds have been completed. There are no unknown or yet to be determined material combinations.

Although the design provides for the option of fabricating Shield Shells from either SA-266 Class 4 forgings or SA-516 Grade 70 plate, all TN-40 Shield Shells are SA-266 Class 4 material. Similarly, although the design allows the option of fabricating Bottom Shields from either SA-105 forgings or SA-516 Grade 70 plate, all TN-40 Bottom Shields are SA-516 Grade 70 material.

Table 2-9.1 below presents the matrix of fabricators, material suppliers and materials used to construct the twenty nine TN-40 casks. For the purposes of later discussion the corresponding components on the current TN-68 cask project are also included.

Cask Numbers	**Mfr**	**Shield Shells**		**Bottom Shields**	
TN-40-1 thru 7	PX	Lavalin	SA-266 Cl 4	Lukens	SA-516 Gr 70
TN-40-8 thru 12	PCC	Hanjung	SA-266 Cl 4	Lukens	SA-516 Gr 70
TN-40-13 thru 17	PCC	Sheffield	SA-266 Cl 4	Creusot - Loire	SA-516 Gr 70
TN-40-18 thru 20	KSL	Forgiatura Vienna	SA-266 Cl 4	ISG (Lukens)	SA-516 Gr 70
TN-40-21 thru 24	KSL	Forgiatura a Vienna	SA-266 Cl 4	Kobe Steel	SA-516 Gr 70
TN-40-25 thru 29	KSL	Forgiatura a Vienna	SA-266 Cl 4	Kobe Steel	SA-516 Gr 70
TN-68-45 thru 54	ENSA	Forgiatura a Vienna	SA-266 Cl 2	Dillinger Hutte	SA-516 Gr 70
(1) The TN-68 Shield Shells listed were procured as SA-266 Class 2 material; however, the actual chemical compositions also meet SA-266 Class 4 (TN-40) requirements.					

Table 2-9.1

The need for preheat for welding and susceptibility to weld cracking is typically measured by the carbon equivalent (Ceq), or preferably, by the crack susceptibility parameter (Pcm). The Pcm value in particular is closely related to fracture toughness. The values for the twenty nine TN-40 Shield Shells and Bottom Shields are shown in Table 2-9.2 along with those for the TN-68 project.

Cask Fabricator	Cask Number	Material Supplier	Component	Ceq Min	Ceq Max	Pcm Min	Pcm Max
PX	1-7	Lavalin	Shield Shell	0.445	0.487	0.299	0.338
PCC	8-12	Hanjung	Shield Shell	0.39	0.418	0.287	0.316
PCC	13-17	Sheffield	Shield Shell	0.548	0.606	0.378	0.396
KSL	18-20	Forgiatura a Vienna	Shield Shell	0.382	0.453	0.254	0.295
KSL	21-24	Forgiatura a Vienna	Shield Shell	0.458	0.465	0.295	0.299
KSL	25-29	Forgiatura a Vienna	Shield Shell	0.419	0.459	0.267	0.279
ENSA	TN-68	Forgiatura a Vienna	Shield Shell	0.361	0.455	0.235	0.313
PX	1-7	Lukens	Bottom Shield	0.45	0.46	0.316	0.325
PCC	8-12	Lukens	Bottom Shield	0.489	0.517	0.335	0.344
PCC	13-17	Creusot-Loire	Bottom Shield	0.447	0.472	0.272	0.288
KSL	18-20	ISG (Lukens)	Bottom Shield	0.466	0.472	0.33	0.33
KSL	21-24	Kobe Steel	Bottom Shield	0.477	0.477	0.316	0.316
KSL	25-29	Kobe Steel	Bottom Shield	0.467	0.469	0.313	0.315
ENSA	TN-68	Dillinger Hutte	Bottom Shield	0.406	0.448	0.265	0.287

Ceq = C + Mn/6 + (Cr + Mo + V)/5 + (Ni +Cu)/15
Pcm = C + Si/30 + (Mn + Cu + Cr)/20 + Ni/60 + Mo/15 + V/10 + 5B

Table 2-9.2

Fracture toughness of a metal is influenced by chemical composition and various physical factors. For steels, carbon content, alloying elements, gas content and impurities are chemical factors that affect this property. The physical factors include microstructure, grain size, section size, hot working temperature and method of fabrication. Surface conditions such as carburization and decarburization are important also.

The chemical composition of the TN-40 Shield Shells and Bottom Shields is fixed within the limits of the material specifications. Since all fabricators exercised the same options (i.e., all Shield Shells are SA-266 Grade 4 and all Bottom Shields are SA-516 Grade 70) there is little difference between casks. This observation is further supported by the consistency in Ceq and Pcm values as shown in Table 2-9.2.

There is little difference between units with regard to the physical factors affecting fracture toughness as all units were hot worked and fabricated by the same methods and are identical with regard to section size.

No fracture toughness testing was performed on TN-40 Shield Shells and Bottom Shields. However, corresponding TN-68 components were tested during the latest procurement. Since there is little difference between the two designs with regard to size and configuration, the data gathered on the TN-68 project is directly usable for predicting the fracture toughness of the TN-40 components.

Table 2-9.3 presents a matrix of available fracture toughness data, including that for welding procedures and electrodes used in TN-40 fabrication.

		Shield Shells		**Bottom Shields**		**Welding**			
						Electrode		**PQR**	
Cask Numbers	**Mfr**	**-20F CVN**	**DW**	**-20F CVN**	**DW**	**-20F CVN**	**DW**	**-20F CVN**	**DW**
TN-40-1 thru 7	PX	No	No	No	No	No	No	No	No
TN-40-8 thru 12	PCC	No	No	No	No	**Yes**	**Yes**	**Yes**	No
TN-40-13 thru 17	PCC	No	No	No	No	**Yes**	**Yes**	**Yes**	No
TN-40-18 thru 20	KSL	No	No	No	No	**Yes** (1)	No	No	No
TN-40-21 thru 24	KSL	No	No	No	No	**Yes** (1)	No	No	No
TN-40-25 thru 29	KSL	No	No	No	No	**Yes** (1)	No	No	No
TN-68-45 thru 54	ENSA	**Yes**	No	**Yes**	No				

(1) The Charpy V-notch testing for these electrodes was conducted at 32°F.

Table 2-9.3

For current procurement, each TN-68 Shield Shell is fabricated from two forgings. Fracture toughness test results (-20F Charpy V-notch testing) for forty forgings is available and shown in Figure 2-9.1.

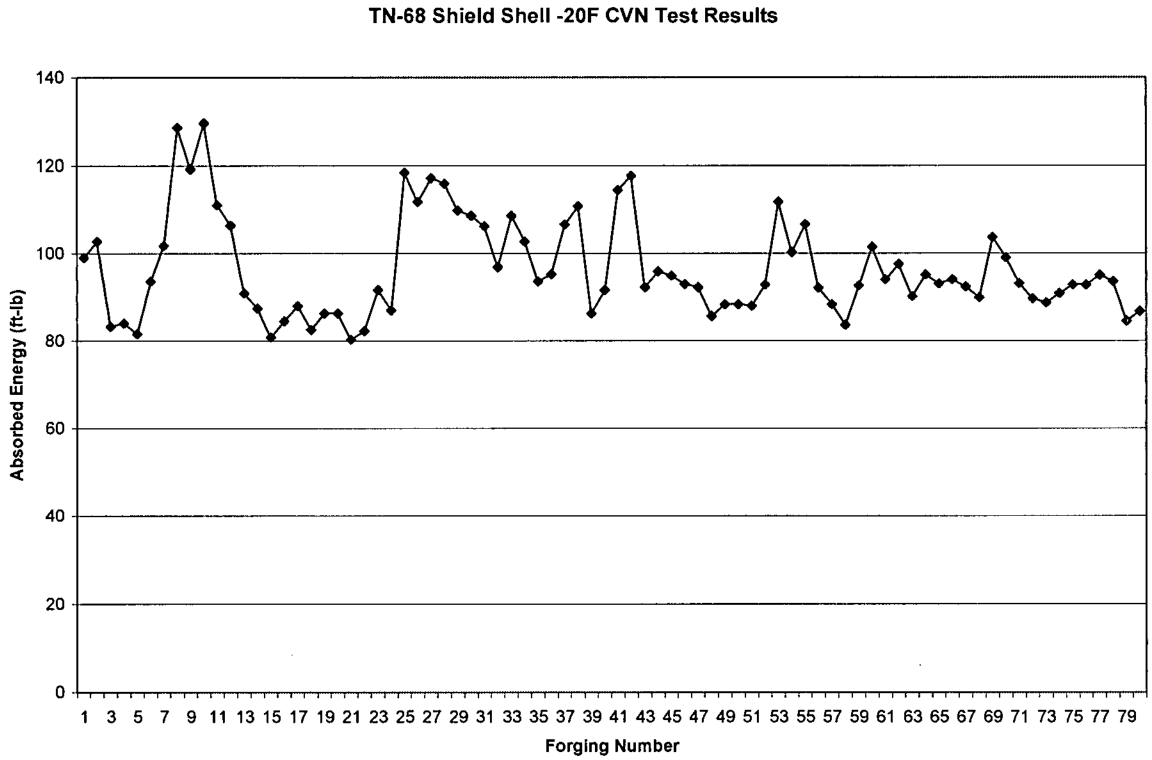


Figure 2-9.1

The average test result for the fortyTN-68 Shield Shells at -20°F was 96.6 ft-lb with a standard deviation of 11.3 ft-lb. The TN-68 results are both much higher (over five times) than the TN-40 requirement of 18 ft-lb and very consistent. 18 ft-lb equals the average TN-68 result minus nearly seven standard deviations. These results strongly support the conclusion that the fracture toughness properties of the TN-40 Shield Shells exceed the minimum required by a margin which can safely accommodate any realistic variations in chemical composition and physical processing as well as the effects of welding.

Fracture toughness test results (-20F Charpy V-notch testing) for nine plates used for Bottom Shield are available and shown in Figure 2-9.2.

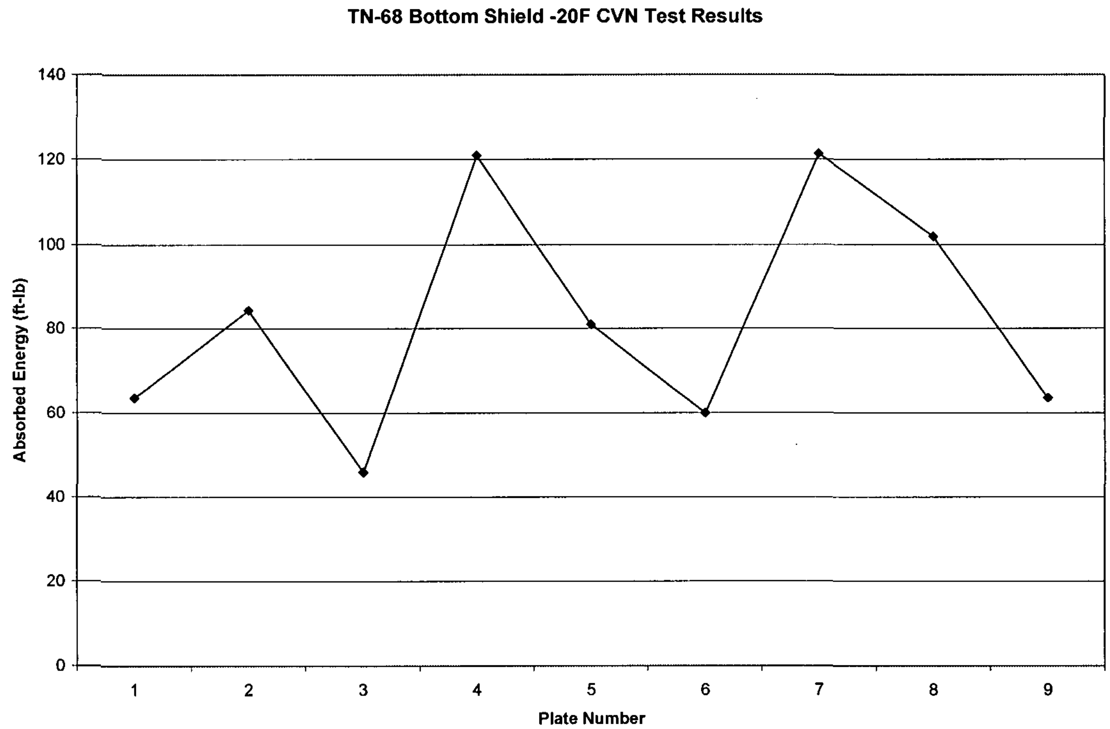


Figure 2-9.2

The average test result for the TN-68 Bottom Shields at -20°F was 82.6 ft-lb with a standard deviation of 27.2 ft-lb. The TN-68 results are much higher (4.6 times) than the TN-40 requirement of 18 ft-lb. 18 ft-lb equals the average TN-68 result minus 2.4 standard deviations. These results strongly support the conclusion that the fracture toughness properties of the TN-40 Bottom Shields exceed the minimum required by a margin which can safely accommodate any realistic variations in chemical composition and physical processing, as well as the effects of welding.

Limited fracture toughness test data is available for the lots of welding electrodes used in welding TN-40 Shield Shells and Bottom Shields. Although the number of tests is limited, the results for those lots tested far exceed that required as shown in Table 2-9.4.

Application	Electrode Supplier	Heat/Lot	NDTT by Drop Weight[1]	-20°F CVN (ft-lb)	32°F CVN (ft-lb)
FCAW (root)	Tri-Mark	5060A	≤-40F	65	
GTAW (repair)	ESAB	065535	≤-30F	213	
SAW	Lincoln	480A/480B	≤-80F	107	
SAW	Lincoln	534H/534J	≤-80F	169	
GTAW (repair)	Kobelco	E6B8391			134
SMAW (root)	Kobelco	A561			124
SMAW (root)	Kobelco	A781S			108

[1] The value provided is the lowest temperature at which the testing was conducted. The suppliers test at these temperatures; if they get no breaks they report the NDTT as equal to or less than the test temperatures. Thus, the NDTT is at some temperature below these listed temperatures.

Table 2-9.4

The nil-ductility transformation temperature of weld metal can be predicted based on the chemical composition of the electrode. Table 2-9.5 presents the predicted NDT temperatures for all welding electrode lots used to weld TN-40 Shield Shells to Bottom Shields. The method used is that described in the technical paper as Enclosure 13: H. Fujii and K. Ichikawa: "Estimation of Weld Properties by Bayesian Neural Network," Welding in the World, vol. 70 (2001), No.3, p.335-339. The predictions were made by a neural network analysis of a database of low alloy weld metal from the University of Cambridge. The predicted NDT temperature is that for a Charpy impact value of 28 J (21.4 ft-lb).The relationship between experimental and predicted values is shown in Figure 2-9.3.

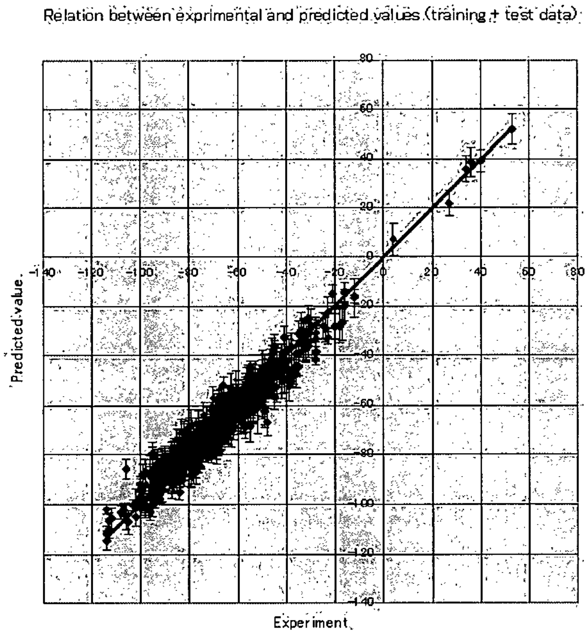


Figure 2-9.3

Predicted Weld Metal Nil-Ductility Transformation Temperatures

Electrode Heat/Lot	Specification	Welding Process	Actual NDTT by Drop Weight[1]	Actual -20F Charpy (ft-lb)	Predicted 28J (21 ft-lb) Transition (°F)
480A/480B	SFA-5.23 ENi1K	SAW	≤-80F	107	-86
534H/534J	SFA-5.23 EG-Ni1	SAW	≤-80F	169	-102
4FNR061/EZ40275	SFA-5.17 F7A6-EH14	SAW			-76
5FNR061/E250127	SFA-5.17 F7A6-EH14	SAW			-77
8HNR062/GZ80678	SFA-5.17 F7A6-EH14	SAW			-79
5060A	SFA-5.29 E81T1-Ni1	FCAW (root)	≤-40F	65	-101
A561	SFA-5.1 E7016	SMAW (root)			-73
A781S	SFA-5.1 E7016	SMAW (root)			-87
65535	SFA-5.18 ER70S-2	GTAW (repair)	≤-30F	213	-94
E6B8391	SFA-5.18 ER70S-2	GTAW (repair)			-78

[1] The NDTT values listed represent the lowest temperature at which testing was conducted. The suppliers test at these temperatures; if they get no breaks they report the NDTT as equal to or less than the test temperatures. Thus, the NDTT is at some temperature below these listed temperatures.

Table 2-9.5

The results presented in Table 2-9.5 show that the chemical composition of the electrodes actually used to weld the TN-40 Shield Shells and Bottom Shields is such that the resultant welds will retain fracture toughness properties well in excess of the required 18 ft-lb to temperatures well below -20°F.

One TN-40 fabricator performed -20°F Charpy V-notch testing in qualifying their TN-40 Shield Shell to Bottom Shield SAW procedure using 1-1/2" thick SA-516 Grade 70 base material. The results are shown in Table 2-9.6.

PQR Test Results for SAW on SA-516 Grade 70

Specimen Number	Absorbed Energy (ft-lb)	Lateral Expansion	% Ductility
1-weld	90	.051	70
2-weld	264	.085	100
3-weld	264	.080	100
Average	**206**	**.072**	**90**
1-HAZ	114	.080	60
2-HAZ	97	.077	55
3-HAZ	123	.080	65
Average	**111**	**.079**	**60**

Table 2-9.6

Another TN-40 fabricator performed -20°F Charpy V-notch testing in qualifying their TN-68 Shield Shell to Bottom Shield SAW procedure using 10" thick SA-266 Grade 4 material. The results are shown in Table 2-9.7.

PQR Test Results for SAW on SA-266 Class 4

Specimen Number	Location	Absorbed Energy (ft-lb)
TG-1-AD	Weld 1/16" depth	84
TG-1-BD	Weld 1/2T	62
TG-1-CD	Weld 3/4T	68
TG-1-DD	Weld 1/16" depth	92
TG-1-AH	HAZ 1/16" depth	106
TG-1-BH	HAZ 1/2T	140
TG-1-CH	HAZ 3/4T	78
TG-1-DH	HAZ 1/16" depth	89

Table 2-9.7

The second TN-40 fabricator also performed -20°F Charpy V-notch testing in qualifying their TN-68 GTAW and SMAW procedures using 10" thick SA-266 Class 4 material. The results are shown in Figure 2.9-8.

PQR Test Results for GTAW (root) and SMAW on SA-266 Class 4

Specimen Number	Location	Absorbed Energy (ft-lb)
TG-2-AD	Weld 1/16" depth	135
TG-2-BD	Weld 1/2T	137
TG-2-CD	Weld 3/4T	250
TG-2-DD	Weld 1/16" depth	166
TG-2-AH	HAZ 1/16" depth	132
TG-2-BH	HAZ 1/2T	149
TG-2-CH	HAZ 3/4T	93
TG-2-DH	HAZ 1/16" depth	91

Table 2-9.8

Therefore, quantitative test results are available for assessing the effect of all welding processes used to join Shield Shell and Bottom Shield materials. From Tables 2-9.6, 2-9.7 and 2-9.8 it can be seen that both SA-266 Class 4 and SA-516 Grade 70 materials retain notch toughness significantly in excess of 18 ft-lb to temperatures below -20°F.

Summary:

- *All TN-40 Casks have been fabricated to a point where the actual base materials, welding materials and welding processes have been determined.*
- *The design allows alternative materials for construction; however, all Shield Shells are SA-266 Class 4 material and all Bottom Shields are SA-516 Grade 70 material.*
- *Although fracture toughness testing was not required for the Shield Shell and Bottom Shield materials or for the welding electrodes, significant data does exist. All such data shows fracture toughness properties significantly in excess of that required by design.*
- *Weld metal fracture toughness predictions based on chemical composition are consistent with actual Charpy V-notch and Drop Weight testing results, and show all materials exhibit fracture toughness values significantly exceeding the required 18 ft-lb to temperatures well below -20°F.*

2-10 State the weld parameters utilized in the weld procedure during testing that resulted in weldments having fracture toughness >>18 ft-lbs.

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.

Response to 2-10

The primary parameters affecting fracture toughness for TN-40 Shield Shell to Bottom Shield welds are shown in Table 2-10.1.

Order	Units	WPS	Process	Filler Metal Size	Welding Current	Travel Speed & Bead Type	Minimum Preheat	Maximum Interpass	Mean Heat Input
1st	1-7	W192	SAW	5/32"	375-850 amps 28-36 volts DCEN	20-40 ipm Stringer Bead	55°F	400°F	54 kJ/in
2nd & 3rd	8-17	SW-010	FCAW (root)	0.035	100-300 amps 20-30 volts DCEP	5-20 ipm Stringer Bead	60°F	500°F	57 kJ/in
		SW-009	SAW	5/32"	400-700 amps 21-36 volts DCEN	8-25 ipm Stringer Bead	100°F	500°F	105 kJ/in
4th & 5th	18-29	366507-04 467017-04	SMAW (root)	5mm (~3/32")	180-240 amps 20-27 volts AC	15- 20cm/min (6-8 ipm) Stringer Bead	50°F	482°F	18 kJ/in
			SAW	4mm (~5/32")	550-650 amps 28-33 volts AC	65-5cm/min (25-30 ipm) Stringer Bead	50°F	482°F	41 kJ/in

Table 2-10.1

Although there are differences between the fabricators with regard to the manner in which root passes were accomplished, little difference exists with regard to the primary welding process. All fabricators used a high amperage, high heat input SAW procedure with 5/32" diameter electrode.

The SAW process was chosen not only for its consistent high quality clean weld quality and automation efficiency but also for its high heat input making preheat optional.

2-11 Explain what the term "Junction" means or refers to as used in the additional set of test results provided by TN in the response to the first round RAI 2-15.

An additional set of test results from another fabricator was provided in TN's response to show relative toughness of SA-516-70 welds and the term "Junction" is used as part of the areas evaluated.

Response to 2-11

RAI 2-11 requests information concerning a response to the first set of RAIs. The information provided in the first response has been replaced with the information contained in Responses 2-9 and 2-10 above. Therefore, RAI 2-11 is no longer relevant.

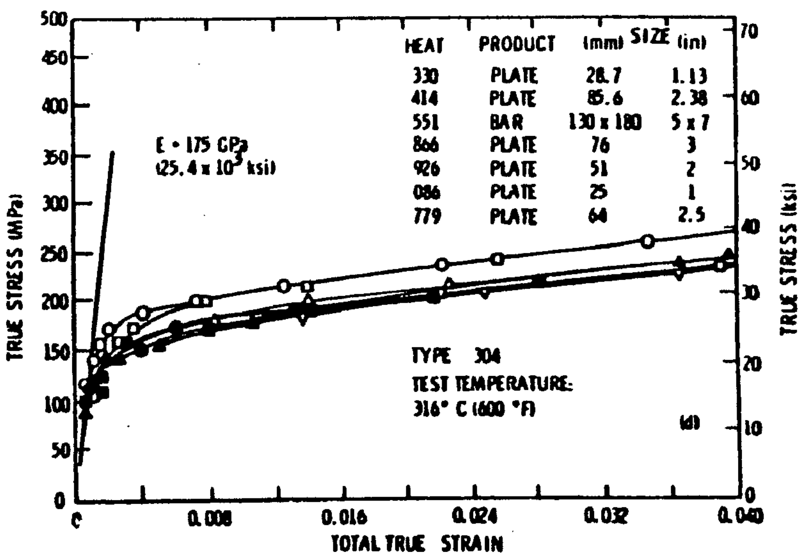
Appendix 2.10.5 Structural Analysis of the TN-40 Basket

The following information is needed to meet the requirements of 10 CFR 71.35(a) unless otherwise stated.

2-12 On page 2.10.5-4, provide data or reference to show that a 5% strain-hardening rate (Ep/E = 0.05) is conservative for the bilinear stress-strain curve implemented for both the SA-240 Type 304 SS steel and SB-209 Type 6061-T651 aluminum alloy plates in the basket non-linear finite element structural analysis.

Response to 2-12

A figure with representative stress-strain curves for SA-240 Type 304 is shown below. The figure is taken from NUREG/CR 0481 [1]. This figure is used as a basis for the 5% strain-hardening rate shown in the bilinear stress-strain curve.



Heat-to-heat variation in stress-strain diagram for 304 stainless steel tested at (a) 24°C, (b) 93°C, (c) 204°C, and (d) 316°C.

Using a 5% strain-hardening rate, which is greater than those shown in the figure above, is conservative because it will yield higher stresses. The maximum stress in the SS boxes is less than 30 ksi (~201 MPa); thus the material remains in the low plastic strain region where the effect of the strain-hardening rate is minimal.

For the aluminum SB-209 Type 6061-T651, Kaufman [2] gives elongations of 17 – 70% with associated strain hardening rates less than 5%. In the case of aluminum, also, using a 5% strain-hardening rate is conservative because it will yield higher stresses.

Note that all P_m limits ($0.7S_u$) are below S_y; therefore the strain-hardening rate has no effect on P_m stresses.

References for RAI 2-12:

1. *"An assessment of Stress-Strain Data Suitable for Finite Element Elastic-Plastic Analysis of Shipping Containers," NUREG/CR-0481, SAND77-1982.*

2. *Kaufman, J. Gilbert, Properties of aluminum alloys: tensile, creep, and fatigue data at high and Low Temperatures, 1999.*

2-13 On page 2.10.5-11B, supplement the fuel basket sensitivity analysis with structural stability load limit tests, such as those presented in Appendix 4C to the Prairie Island ISFSI SAR for the TN-40 basket evaluation (Docket 72-10).

The testing parameters, including specimen temperature, and plate thicknesses, of the Appendix 4C load limit tests should be evaluated by TN to ensure their applicability to the TN-40 shipping package configuration.

Response to 2-13

The fuel cell wall load limit test report provided as Appendix 4C to the TN-40 Storage SAR and as Enclosure 20 of this submittal [1] has been added to the Transport SAR Appendix 2.10.5. In addition, a 75g side drop evaluation of the basket was conducted. The resulting fuel cell wall buckling load was compared to the buckling load determined in Appendix 4C. The comparison shows a factor of safety of approximately 1.73. The description and results of the calculation have also been added to Transport SAR Section 2.10.5.5.3.

[1] Scavuzzo, R. J., Lam, P. C., Gau, J. S., "Buckling Tests of Fusion Welded Composite Stainless Steel Aluminum Plates," Dept. of Mechanical Engineering, The University of Akron, May, 1990

2-14 Ascertain and confirm that the stainless steel fuel compartment wall is not allowed to lose contact with the spacer plugs at three locations (pages 2.10.5-54, Figure 2.10.5-33).

The plug weld joints must be properly modeled to reflect actual interface conditions for the finite element analysis of the fuel basket. The model displayed in the figure shows that all five locations are treated as being connected with spacer plugs, which is misleading.

Response to 2-14

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

Transport SAR Figure 2.10.5-33 has been modified such that it shows both the contact elements that represent the plate-to-plate interface and the pipe elements that represent the attachment between adjacent fuel compartment walls to one another through the plug and fusion welds.

2-15 On page 2.10.5-43, Figure 2.10.5-22, revise the sketch to properly reflect the steel plate-to-aluminum plate contact and the plate-to-plug joint interface conditions, that is, the compartment wall remains in contact with the spacer plugs in these locations.

Response to 2-15

The fuel compartment tubes, aluminum plates and periphery plates are modeled with shell elements; 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 are modeled by unidirectional couples along the through thickness of the plates.

SAR Figure 2.10.5-22 has been revised to depict element types, and interfaces.

2-16 On page 2.10.5-35, with respect to the imposed initial "imperfections" displayed in Figure 2.10.5-35, perform a fuel basket buckling sensitivity analysis, as appropriate, by considering conservative sets of wall panel imperfections, including imperfections associated only with the "vertical" wall panels.

The present analysis imposes convex imperfections on all "horizontal" wall panels subject to downward loads. This assumption may yield non-conservative results due potentially to the negative work done by the downward loads on the horizontal wall panels undergoing upward deflections. Conservative assumptions must be used for the subject buckling analysis.

Response to 2-16

Initial imperfections are imposed on horizontal panels of the fuel compartments. However, the gap between the contact elements separating adjoining panels is set to zero so that no added force is needed to close the contacts.

Appendix 2.10.7

The following information is needed to meet the requirements of 10 CFR 71.35(a) unless otherwise stated.

2-17 In Section 2.10.7.2.F., page 2.10.7-6, Analysis and Results, provide a re-evaluation of "gap" effects on the fuel clad structural integrity for the 30-ft end drop accident by recognizing that: (1) The forcing function, which was measured for the rigid-body response of a combined cask and content as an integral body may need to be modified to account for the cask-content gap effect, before its implementation for the fuel rod time-history impact response analysis.

Table 2.10.7-5 of the application reports that as the gap between the cask and the content increases, the maximum total axial strain decreases, which is not physically intuitive. The staff notes that, as gap increases, the relative velocity between the impacting bodies (cask and content) also increases. Depending on gap size, as the content moves to contact the cask, it may potentially result in higher than the nominal contact velocity of 527.4 in/sec as the cask body begins to rebound to result in an exacerbated secondary impact effect. As such, the forcing function introduced at the interface between the end fitting and fuel rod (cask and content) may have to be modified depending on the gap size considered for the evaluation.

Response to 2-17

Transient dynamic analysis of the fuel for the 30 ft end-drop using LS-DYNA is performed to determine the maximum strain. The new analysis replaces the current analysis in the SAR Appendix 2.10.7. The methodology is based on [1] and [2] and is similar to the analysis submitted to NRC for review as part of the RAI responses for the TN40HT storage application [3]. The model is validated against the results provided in [1] and the validation analysis is also included in SAR Appendix 2.10.7. The analysis internally calculates the force on the fuel cladding depending on the gap size.

1. *H. E. Adkins, Jr., B. J. Koeppel, and D. T. Tang, Spent Nuclear Fuel Structural Response when Subject to an End Impact Accident, PVP2004, San Diego, CA, July 25-29 2004.*
2. *NUREG 1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," USNRC, March 2007.*
3. *L-PI-09-071, "Supplement to License Amendment Request (LAR) to Modify TN-40 Cask Design (Designated as TN-40HT) (TAC No. L24203)," June 6, 2009.*

2-18 In Section 2.10.7.2.B., on page 2.10.7, "Assumptions," in order to demonstrate adequate performance of the fuel rod analytical model, provide a sensitivity analysis of key modeling parameters, including gap size and contact stiffness between the grid spacer support and the fuel tube wall, integration time step size, and fuel rod element discretization scheme, to ensure proper performance simulation for the fuel rod subject to 30-ft end-drop condition.

TN performed transient analyses of a highly non-linear fuel rod system. However, the results are reported without evidence to show that model parameters are properly selected to ensure adequate simulation of key boundary and interface conditions, which may significantly affect calculated results. A sensitivity analysis should be performed to demonstrate realistic and conservative selection of model parameters.

Response to 2-18

Please see the response for RAI 2-17.

2-19 On page 2.10.7-16, Table 2.10.7-5, explain why the maximum clad total axial strain of 0.72% is markedly below the 1.2% reported in the July 2, 2008, public meeting for the same no-gap condition.

Provide justifications for model parameter changes if they are different from those used in the analysis presented in the public meeting.

Response to 2-19

Please see the response for RAI 2-17.

2-20 On page 2.10.7-24, Figure 2.10.7-6 with respect to the apparent "kink" shown in the displacement time history for Node No. 1190, ascertain and confirm that a potentially large local strain, which is associated with large curvature of a kink, can be captured by the bending of the PIPE20 elements used to simulate the fuel clad behavior.

The time-history plots show that the fuel rod deflects laterally to the point where it contacts the fuel tube or an adjacent rod and continues to expand the contact area to result in large curvatures or kinks at or near the ends of the flattened part of the rod. The PIPE20 element used to model the fuel rod is essentially a beam element, and its applicability must be justified for evaluating the fuel rods with large localized stress and strain.

Response to 2-20

Please see the response for RAI 2-17.

3.0 THERMAL

3-1 Provide the pages from the 1989 American Society of Mechanical Engineer (ASME) Boiler and Pressure Vessel (B&PV) code showing the composition of the group used to determine the Thermal Conductivity.

The staff could not identify the alloy SA350 Grade LF3 steel in the list of alloys for which the thermal conductivity was provided in the 2004 edition of the code.

This information is required for compliance with 10 CFR 71.43(f) & (g).

Response to 3-1

The nominal composition of SA-350 Grade LF3 is 3½ Ni as shown in ASME code 1989, Section VIII, Division 2, Table ACS-1, Page 80 and ASME code 2004, Section II, Part D, Table 1A, Page 58, Line 20 and 21.

The thermal conductivity of this material is listed below for reference.

Temperature (°F)	*Thermal Conductivity*		*Thermal Conductivity*	
	(Btu/hr-ft-°F) (1)	*(Btu/hr-in-°F)*	*(Btu/hr-ft-°F)* (2)	*(Btu/hr-in-°F)*
70	*22.9*	*1.91*	*27.3*	*2.28*
100	*23.2*	*1.93*	*27.6*	*2.30*
200	*23.8*	*1.98*	*27.8*	*2.32*
400	*23.9*	*1.99*	*26.5*	*2.21*
600	*22.9*	*1.91*	*24.9*	*2.08*
800	*21.6*	*1.80*	*23.2*	*1.93*
1000	*20.1*	*1.68*	*21.1*	*1.76*
1200	*18.2*	*1.52*	*18.3*	*1.53*
1400	*15.5*	*1.29*	*15.7*	*1.31*

Notes:

(1) These values are based on ASME code 1989, Section VIII, Division 2, Table 1, page 51. The conversions of these values to Btu/hr-in-°F are identical to those used in SAR Section 3.2, Material # 5. These values are more conservative than those from ASME code 2004.

(2) These values are based on ASME code 2004, Table TCD, Group B [Note (2)], page 684.

3-2 Provide additional clarification as to how the transverse effective conductivity values for the "Based on UO_2 from NUREG/CR-0200 Rev. 6 (Scale)" and the "Based on UO_2 from NUREG/CR-0497 (Matpro)" lines are calculated in Figure 3-16 of the RAI response.

Justify the comparison of the "TN-40 SAR Rev. 1" line with the aforementioned lines in Figure 3-16 of the RAI response.

It is stated in the RAI response that both calculated values of transverse effective conductivity are at least 20% higher than those used in the ANSYS model and reported in the SAR. It is not clear if the line "TN-40 SAR Rev. 1" in Figure 3-16 of the RAI response can be compared to the other two calculated lines in Figure 3-16 because it is not clear how those two lines were calculated. Both the "TN-40 SAR Rev. 1" line and the "Based on UO_2 from NUREG/CR-0200 Rev. 6 (Scale)" line are a function of the SCALE UO_2 parameter. Because those two lines are not coincident, another parameter (or multiple parameters) has to be varied (i.e., decay heat, compartment width, active fuel length), therefore it is not clear if the comparison is appropriate for the TN-40 package.

This information is needed to determine compliance with 10 CFR 71.43(f) & (g).

Response to 3-2

The transverse effective conductivities for the fuel assembly and hence all the lines in Figure 3-16 of the RAI-1 response are calculated using the same methodology described in SAR Appendix 3.7.1. The difference is that the Stefan-Boltzmann constant in SAR Revision 1 was incorrectly considered to be 1.983E-13 Btu/hr-in^2-°R^4. This value is 60 times lower than the correct value of 1.190E-11 Btu/hr-in^2-°R^4. In addition, the UO_2 conductivity in SAR Revision 1 was approximately 10 times lower than its correct value. Due to these errors in the ANSYS input files, a lower transverse effective conductivity was calculated in SAR Revision 1. Since the transverse effective conductivities from SAR Revision 1 are lower than the corrected values assigned as "Based on UO_2 from NUREG/CR-0200 Rev. 6 (Scale)" shown in the RAI-1 response, Figure 3-16, the SAR results for NCT were not changed and were considered as conservative. A sensitivity analysis documented in Appendix 3.7.1 demonstrates that the fuel assembly transverse effective conductivities used in the transient analysis of HAC are also conservative.

4.0 CONTAINMENT

4-1 Include the detailed calculation package for Basket Volume (1.05E+05 in^3) and Fuel Assembly Volume (1.64E+05 in^3) in Section 4.2 of the SAR.

This item is used in the calculation of source activity released from fuel in transport and subsequently in the determination of permissible leakage rates for normal conditions of transport and for hypothetical accident conditions. A detailed calculation is not included in the SAR.

The applicant in its response to RAI 4-3 of the round 1 request for additional information has provided the detailed calculation of the basket volume, but failed to include the detailed calculations in the revised SAR.

This calculation package is required to verify the applicant's evaluation of package design under normal conditions of transport per as required by 10 CFR 71.71, and under hypothetical accident conditions per 10 CFR 71.73.

Response to 4-1

The detailed calculation of the basket volume has been added to SAR Section 4.2.

5.0 SHIELDING

5-1 Include the response to the first round RAI 5-1 in the SAR and include a requirement for a neutron dose rate measurement at the cask surface prior to shipment in the proposed CoC.

The response to the 1st round RAI provides laboratory testing supporting the conclusion that there has been no deterioration of the neutron absorber and shielding during the storage period and as such should be in the SAR. Since the casks have been loaded and sealed, the only conclusive way to confirm the viability of the absorber and shielding material integrity is by measurement of the dose rate, therefore this testing should be included in the proposed CoC (see also RAIs 8-3, 8-6, and 8-7).

This information is required for compliance with 10 CFR 71.47(a).

Response to 5-1

The response to RAI 5-1 from the first round of RAI has been added to SAR Section 1.2.1.2.

A requirement will be included in the CoC for the package to be prepared for shipment in accordance with the Operating Procedures in Chapter 7 of the application (Safety Analysis Report). Since Steps 7.1.3.16, 7.1.3.25, 7.4.1.25 and 7.4.1.34 in Chapter 7 require radiation surveys to satisfy the shield test requirements and to assure compliance with regulations, these actions would be incorporated into the CoC by reference.

5-2 Modify the shielding evaluation to include spent fuel assemblies with natural or low-enriched uranium axial blankets.

Staff noticed that the criticality evaluation includes assemblies with natural or low-enriched uranium axial blankets (axial blankets); however, the staff was unable to identify any information in the shielding evaluation concerning these proposed contents. The shielding evaluation should account for all proposed contents, providing the necessary analyses (e.g., source term and dose rate calculations).

The use of axial blankets in the fuel assemblies will affect the neutron flux distribution along the axial direction of the assembly. The power density and burnup profiles will hence be skewed in comparison with those without the axial blankets. Staff analyses indicate that using an average fuel enrichment to represent the actual axially-blanketed assembly may significantly underestimate the source term of this type of spent fuel assembly.

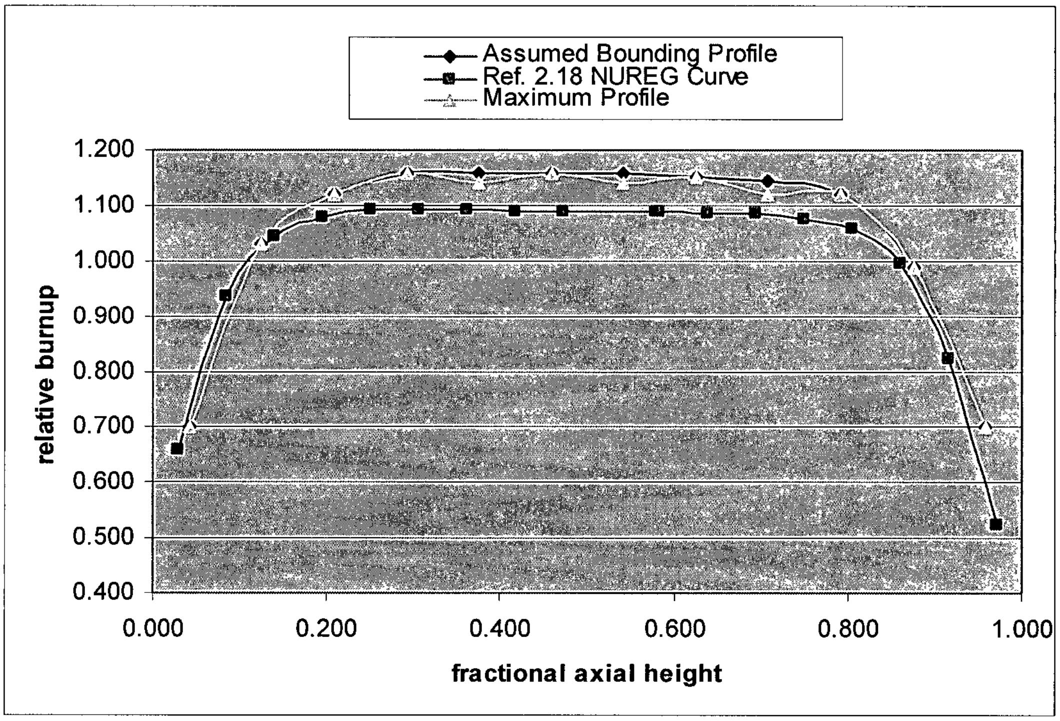
The applicant's evaluation should provide analyses for assemblies with axial blankets or provide justification (including quantitative as well as qualitative support) that the current analyses are applicable to and cover assemblies with axial blankets. The evaluation should consider the lengths of the blankets of the proposed assembly contents (e.g., 6-inch or 12-inch blankets). The evaluation should show the applicability of the burnup profiles selected, whether these profiles are new for additional analyses or are those in the current analyses. The applicability of the methods (to include the computer codes and assumptions) used to determine the source term for these assemblies should also be justified. For example, given the different regions of the assembly enrichment, a multi-dimensional code such as the TRITON sequence in SCALE may be necessary to determine the fuel neutron and gamma source terms accurately. Additionally, the applicant should clarify the determination of the enrichments used for the analyses, whether the minimum enrichments are assembly average enrichments and include or exclude the blankets. Note that in all shielding analyses, the minimum enrichment should be used as this enrichment results in maximum gamma and neutron source terms for the proposed contents. Clarification is also needed for the burnup values such as whether these are assembly average values that include or exclude the blankets. As stated above, the presence of blankets skews the power density and burnup profile such that an assembly with the same burnup as an un-blanketed assembly will have a higher burnup in the enriched part of the fuel than will an un-blanketed assembly.

This information is needed to confirm compliance with 10 CFR 71.47 and 71.51.

Response to 5-2

The population of Prairie Island fuel assemblies contain fuel assemblies with 6-inch and 6.2 inch blankets. All the fuel assemblies containing blankets are NOT the "design basis" type, WE 14x14 standard fuel assemblies. The burn-up of the blanket regions is significantly lower. The axial burn-up profile employed in TN-40 transport calculations is based on bounding profiles from fuel irradiated at Prairie Island including fuel with blankets and is conservatively applied to the design basis fuel assembly. The enrichment used in the shielding analysis is "minimum" assembly average and accounts for blankets. The burn-up used in the shielding analysis is "maximum" assembly average and accounts for blankets. The burnup profile corresponding to the "Assumed Bounding Profile" shown in the figure below is employed in the shielding evaluation. The use of a bounding profile ensures that the most penalizing profile is utilized that accounts for blanketed versus un-blanketed fuel.

New discussion regarding this issue is added to SAR Section 5.2.1.



5-3 Provide additional clarification regarding the response functions for the neutron and gamma dose rate analyses.

The applicant uses a response function method to analyze dose rates for the proposed cask contents. Staff's finding regarding acceptability of the analysis for the proposed cask and contents hinges upon a clear understanding of the analysis method. Specifically, the applicant should clarify the following: 1) the gamma response functions for all four gamma sources (i.e., fuel, plenum and hardware regions), were added to the appropriate energy range/bin response function, and 2) how and why the MCNP model for the response functions is only "essentially the same" as the model for the design basis shielding evaluation and not exactly the same. Staff's acceptance of the shielding analysis relies upon the summation of the gamma response functions as described above as well as the sameness of the MCNP models involved.

This information is needed to confirm compliance with 10 CFR 71.47 and 71.51.

Response to 5-3

The response function methodology employed for the fuel qualification is described in Section 5.2.5 of the SAR. The response functions are employed to rank the radiological sources for their contribution to the dose rates at 2m from the package surface. A more thorough discussion of this methodology is added to Section 5.2.5 of the SAR to provide the required clarification.

The dose rates calculated from the response function methodology include the source term contribution from each of the four source regions for gammas. In order to ensure that the response function is employed appropriately, the SAS2H/ORIGEN-S module calculates a "combined" gamma *source term that includes the contribution from each of the source regions. A new Table 5-20 has been added to the SAR to show the calculated response functions for the TN-40 cask.*

Due to the fact that the dose rates from the side of the TN-40 transportation package are controlling, the axial ends of the package are not modeled in full detail. Therefore, the MCNP models for response function calculations are "essentially the same".

Note that the response function methodology is employed to determine the bounding combination of spent fuel parameters (Burnup, Enrichment and Cooling Time) for shielding calculations. This methodology is also employed to develop the fuel qualification tables to satisfy the Part 71 dose rate criteria. Separate shielding calculations are performed and are documented in Section 5.4 using the design basis source terms.

5-4 Modify Table 5-2 to indicate the proper regulatory dose rate limit for areas on the impact limiter surfaces is 200 mrem/hr (2 mSv/hr) for normal conditions and ensure correctness of the reported dose rate values.

The surface limit of 1000 mrem/hr applies to cask surfaces that are enclosed by the personnel barrier. Based upon the licensing drawings, it appears the impact limiters are outside of the personnel barrier; therefore, the limit of 200 mrem/hr applies to their surfaces. Also, the gamma dose rates for the top and bottom package surfaces have two different values reported. It appears that a conversion between mrem/hr and mSv/hr was done incorrectly. The total dose rate should also be updated, as necessary.

This information is needed to confirm compliance with 10 CFR 71.47.

Response to 5-4

The shielding evaluation has been revised to reflect the effect of more detailed modeling of the trunnion area and to consider material density tolerance limits for the neutron shielding. As a result, dose values have changed. Thus the values provided in Table 5-2 have been revised. More details on modifications are provided in response to RAI 5-8. The correct values and limits at the surface of the impact limiter have been included in the table. Also, titles of some columns in SAR Table 5-2 are modified. The proper regulatory dose rate limit for areas on the impact limiter surfaces is 200 mrem/hour.

5-5 Justify the difference in the neutron shield in the model versus the licensing drawings and include all neutron shield thickness dimensions in the licensing drawings.

The model makes the neutron shield of uniform thickness (4.5 inches). However, Drawing No. 10421-71-3 shows the neutron shield has thinner regions, though the thickness dimension is not given. The application should include discussion of the basis for modeling the neutron shield differently from its actual configuration as given in the licensing drawings. Also, the neutron shield thickness should also be specified in the licensing drawing for these thinner regions.

This information is needed to confirm compliance with 10 CFR 71.47.

Response to 5-5

The shielding model has been revised to include a detailed area around the upper trunnions. This area has a thinner layer of neutron shielding material than the uniform thickness modeled for the balance of the shielding. In addition, the SAR drawings have been revised to provide additional details of the area surrounding the trunnions.

Also, other modifications to the shielding analysis as described in the response to RAI 5-6 are implemented.

5-6 Modify the shielding evaluation to account for the package tolerances, including tolerances on the neutron shield material specifications.

Review of the sample input file indicated that the current shielding evaluation uses nominal dimensions. Understanding that a package may be manufactured to meet the minimum tolerances, these tolerances need to be included in the evaluation; dose rates for such a package should be shown to meet transportation dose rate limits. Additionally, it is not clear that the minimum material density and hydrogen and boron content of the neutron shield material were used in the evaluation. Depending upon margins to the limits, inclusion of minimum tolerances in dose rate analyses may affect whether a package can meet the limits. For example, while the applicant has indicated that deviations within tolerance from the nominal neutron shield properties results in a dose rate change of less than 10%, the dose rate reported in Table 5-2 of the application for the 2-meter location is less than 10% below the regulatory limit.

This information is needed to show compliance with 10 CFR 71.47 and 71.51.

Response to 5-6

The shielding evaluation has been modified to account for packaging tolerances. Information regarding analyzed tolerances is added to the end of SAR Section 5.1. The effect of material tolerances is considered only for the neutron shielding resin. Only hydrogen concentration is considered significant enough to affect the dose rates. The weight percent of hydrogen considered in the design basis shielding analysis represents the minimum guaranteed value following the resin qualification. The actual measured hydrogen weight percent in the TN-40 casks is 5.21 ± 0.14 while that employed in the calculations is 5.05. The boron content has an effect on the secondary gamma dose rate. However, a concentration of greater than 0.75% ensures that this concentration is saturated and is sufficient to reduce the contribution of the secondary gamma component. Therefore, a material tolerance calculation with boron is not performed.

The dose rate results presented in the shielding evaluation (Chapter 5 of the SAR) account for the effect due to tolerances. Note that the "2 meter from Vehicle" dose rates presented in the "Side" column of Table 5-2 prior to the update corresponded to 2 meters radial distance measured from the side of a hypothetical, 10 ft. wide transportation platform. Updated data represents the dose rate at 2 meters radial distance measured from the side of the impact limiters. However, dose rates at 2 meters from the side of a 10 ft. wide transportation platform are also mentioned in the discussion at the end of Section 5.5.

The shielding calculations provide dose rates that are bounding due to the following:

- *Design basis Westinghouse 14x14 Standard fuel assemblies with the bounding neutron and gamma source terms are utilized in the shielding evaluation.*

- *The fuel qualification methodology calls for conservatively adjusting the enrichment / burn-up and cooling time of the loaded fuel assemblies (Chapter 5, Table 5-8).*

- *Calculated dose rates are generally higher than measured dose rates demonstrating the conservatisms in the shielding analysis methodology.*

- The burnup-enrichment parameters of the design basis fuel assembly employed in the shielding evaluation is conservative compared to the burnup-enrichment distribution for the actual inventory of fuel assemblies.

5-7 Restore to Section 1.2.3 of the application the table providing the physical specifications of the assemblies.

In response to staff first round RAIs, the applicant revised the contents description given in Section 1.2.3 of the application. However, some important specifications that were included in the original application were not retained (e.g., cladding material, maximum length, number of fuel rods per assembly, etc.). These specifications should continue to be included in the description of the proposed contents.

This information is needed to confirm compliance with 10 CFR 71.33(b).

Response to 5-7

A table of fuel assembly physical data (cladding material, maximum length, number of fuel rods per assembly, etc.) was added to Section 1.2.3.

5-8 Verify the dose rate profiles in Figure 5-7 are correct for the TN-40 package.

Based upon the maximum side dose rate in Table 5-2 (239 mrem/hr) for normal conditions, it appears that the dose rate profile in Figure 5-7 is not correct; the maximum dose rate in the figure is less than 200 mrem/hr.

This information is needed to confirm compliance with 10 CFR 71.47.

Response to 5-8

Table 5-2 has been updated and Table 5-18 has been added. Figure 5-7 has been deleted. Dose rate distributions previously displayed in the deleted figure are presented in the new table. These changes are made based on the following considerations.

- *The spatial locations of maximum neutron and gamma radiation dose rates do not coincide. That means the maximum of the total dose is less than or equal to the sum of the maximum of neutron and gamma radiation dose rates. However, the maximum total dose rate was presented as the sum of the maximum neutron and the maximum gamma radiation dose rates. The updated Table 5-2 now displays the total dose maximum as the maximum of the neutron + gamma radiation dose rate.*

- *Figure 5-7 displayed dose rate distributions at various radial distances along side of the cask, at axial locations between ends of impact limiters. A portion of displayed dose rate distributions on the figure corresponding to "Package Surface" and "Vehicle Edge" are within a range of axial coordinates that are encompassed by the impact limiters. Therefore, some dose rate values in those distributions are inside of the impact limiters. As the figure demonstrated, maximums for these two distributions are located inside of the impact limiters and were used in Table 5-2. The "Package Surface" and "Vehicle Edge" dose rates in Table 5-2 are updated such that the maximum of these dose rate distributions at or outside the surfaces of the impact limiters are presented now.*

- *Figure 5-7 is deleted for the reason explained in the bullet above. However, dose rates that were displayed in the deleted figure are presented now in a tabular form in Table 5-18. Shaded cells of the table highlight dose rates inside of impact limiters.*

- *"2 meter from Vehicle" dose rates presented in "Side" column of Table 5-2 prior to this update corresponded to 2 meters radial distance measured from side of a hypothetical, 10' wide transportation platform. Updated data represents the dose rate at 2 meters radial distance measured from the side of impact limiters.*

- *Finally, conversion of dose rates from mrem/hr to mSv/hr units are corrected in the updated Table 5-2.*

5-9 Provide further clarification that the dose rates in Table 5-2 are bounding for their respective areas of the cask and are for the bounding burnup, cooling time, and minimum enrichment combination.

It is not clear from the information provided that the dose rates in Table 5-2 are bounding for the proposed contents and, therefore, that the Part 71 dose rate limits will be met. While some statements indicate that fuel qualification analyses were done to determine fuel assembly burnup, cooling time, and minimum enrichment parameter combinations that result in dose rates less than those for the selected "design basis" parameter combination (e.g., last paragraph of Section 5.1 and first paragraph of Section 5.2.5), other statements in the application discuss being below a dose rate limit of 9.8 mrem/hr (end of 4th paragraph in Section 5.2.5) or simply meeting regulatory limits (see last paragraph of Section 5.2.5 and title of Table 5-9). In addition, the results of the analyses indicate many of the contents are at or very near the regulatory limit (Table 5-10). Staff recognize that the cooling times for many contents are extended to a minimum of 15 years, which is significantly longer than the analyzed cooling times; however, others are only rounded up to the next full year, which for some cases is only a minimal increase in cooling time. The foregoing make it unclear that Table 5-2 is the bounding case and that the bounding dose rates meet all the pertinent limits of Part 71.

This information is needed to confirm compliance with 10 CFR 71.47 and 71.51.

Response to 5-9

Section 5.2.5 has been expanded to provide additional discussion on the use of the response function methodology to determine the design basis source terms and to establish the fuel qualification table. The dose rate results shown in Table 5-10 are based on the cooling times shown in Table 5-9 and are estimated using the response functions. The results in Table 5-10 only demonstrate that any burnup, enrichment and cooling time (BECT) combination from Table 5-9 is *expected to result in approximately the same dose rate and will meet all applicable regulatory limits. This is demonstrated in the shielding analysis in Section 5.4 using one set of source terms that required the longest cooling time and hence are bounding from a shielding analysis standpoint. Further, the difference in the resulting dose rates for the various BECT combinations will be within 0.1 mrem/hour implying that the set of source terms employed in the shielding analyses is representative and bounding.*

A new Section 5.5 has been added to the SAR. This section discusses the quantification and effect of uncertainties in source terms and dose rates that are presented in the TN-40 shielding calculation results. These results demonstrate that the controlling 2m radial dose rate remains within the regulatory limit of 10 mrem/hour when all uncertainties are considered. Above all, the results of the criticality analysis documented in Chapter 6 require a minimum cooling time of 30 years providing additional margin to the fuel qualification calculations. The shielding results with the 30 year cooling time are shown in Table 5-2 and the new Table 5-19 of the SAR and demonstrate that there is sufficient margin to meet the applicable regulatory limits.

5-10 Clarify that the calculated dose rates in the shielding analyses account for the uncertainties in the dose rates.

The dose rates in Table 5-10 indicate many dose rates are at or very near the regulatory limit. Dose rate (and source term) calculations have uncertainty. This uncertainty should be accounted for in determining whether the package meets the regulatory dose rate limit. While the cooling times for the calculations are different than those that are used in the qualification table, the differences for some are only minimal and thus may have only a minimal effect on dose rates. Uncertainties should also be addressed for the dose rates in Table 5-2. The calculations may include conservatisms that offset the uncertainties; however, there is currently no evaluation (quantitative or otherwise) of the conservatisms in the calculations that could demonstrate an adequate level of offset of the uncertainties.

This information is needed to show compliance with 10 CFR 71.47.

Response to 5-10

A new Section 5.5 has been added to the SAR. This section provides a discussion on the uncertainty associated with source term and shielding calculations. Also please see Response to 5-6 above. Above all, the dose rates calculated with a cooling time of 30 years (required by the criticality analysis) ensure that there is sufficient margin to meet the applicable regulatory limits.

6.0 CRITICALITY

The following information is needed pursuant to the requirements of 10 CFR 71.55, 71.59, and 71.73.

6-1 RAI 6-1 is proprietary. Please see Enclosure 5.

6-2 Correct the reference in Section 6.3.2 (last paragraph on page 6-12 and Table 6-10a on page 6-69) for the technical report that provides data for the isotopic composition bias and measured spent fuel compositions for the isotopic composition validation.

Reference [1] (SCALE computer code) was cited in the last paragraph on page 6-12 and Reference [16] (NUREG/CR-6951) was cited in Table 6-10a. In these instances, Reference [13] (NUREG/CR-6811) is the correct reference to use.

Response to 6-2

The two references are corrected.

6-3 RAI 6-3 is proprietary. Please see Enclosure 5.

6-4 RAI 6-4 is proprietary. Please see Enclosure 5.

6-5 RAI 6-5 is proprietary. Please see Enclosure 5.

6.5.1 Benchmark Experiments and Applicability

6-6 Provide justification for the applicability of the selected criticality benchmarks as listed in Table 6-20.

Table 6-20 of the revised SAR provides a list of the critical experiments that are used in the code criticality benchmark and USL calculation for the cask criticality safety analyses. Demonstrate the neutronic similarities of the critical experiments to the cask by comparing the range of material composition, geometric arrangement, moderator condition, reflector, and neutron spectra indices to that for the cask. In addition, identify which laboratory criticals experiments (LCE) benchmark the cross sections for Eu-151, Gd-155, and Am-141. These isotopes are mainly produced after discharge from the decay of Sm-151, Eu-155, and Pu-241 with half lives of 90, 5, and 14 years respectively. Therefore, the commercial reactor criticals (CRC) do not provide adequate benchmarking of the cross sections for these isotopes.

Furthermore, with the exception of the CRCs, the criticality validation set does not contain critical experiments with the uranium and plutonium compositions similar to what is typically found in commercial spent nuclear fuel. Examples of actual spent fuel isotopes may be obtained from Appendix A of Reference [13] listed in Section 6.6 of the TN-40 criticality evaluation. Note that the mixed-Pu/U oxide LCEs listed in Table 6-20 (see page 6-91 in the Rev. 2 document) do not have plutonium and uranium in the proper proportions to be similar to commercial spent nuclear fuel. In all 11 selected LCEs, the U-235 content appears to be too low, the ratio of Pu/(U + Pu) appears to be too high, and the Pu-240 content appears to be too low as well. Section 4.3.3 of ANSI/ANS-8.1-1998 states: "The uncertainty in the bias shall contain allowances for uncertainties in experimental conditions, for the lack of accuracy and precision in the calculational method, and for extension of the area (or areas) of applicability." Data has recently become available for critical experiments with mixed-Pu/U oxide similar to spent fuel (see NUREG/CR-6979).

For the purpose of cask criticality benchmarking, it appears that the recently published NURGE/CR-6979, "Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data," provides a new set of critical benchmark experiments that may be more appropriate for the purpose of spent fuel cask criticality benchmark. The applicant may want to consider using these data for the criticality benchmark of the TN-40 Prairie Island burnup-credit transportation casks.

Response to 6-6

The comparison among LCE, CRC and TN-40 Systems are given in SAR Section 6.5.1. Sensitivity calculations are performed to address issues regarding Eu-151, Am-241 and Gd-155 (Section 6.4.5 and 6.4.6).

6-7 Provide justification for the applicability of the selected Commercial Reactor Critical records as criticality benchmarks for the Prairie Island burnup-credit transportation casks.

The applicant utilizes Commercial Reactor Critical records to benchmark the SCALE 4.4 computer code for the TN-40 Prairie Island burnup-credit transportation cask design. Table 6-20 of the revised SAR provides a list of the Commercial Reactor Critical records that are used in benchmarking the code for cask criticality and USL calculations. However, no sensitivity and uncertainty analyses were discussed in the revised SAR, which are integral parts of the methodologies discussed in References [3] and [17]. The applicant needs to provide justifications for the applicability of the selected CRC records for the TN-40 design-specific criticality benchmark and USL calculations. For this purpose, the methodology illustrated in Reference [17] is probably more applicable for determining the applicability of some CRC data to the burnup-credit cask designs such as TN-40. It appears that Reference [3] is probably intended to demonstrate that the SCALE system can be used to perform the criticality calculation for systems with spent fuel materials. The staff's understanding is that the work presented in Reference [3] is probably rather a proof of concept than a practical method because this publication does not seem to provide many details on how to bridge reactor critical configurations with that of spent fuel casks. The applicant is requested to provide justification with discussion of the impact of the important parameters such as the number of isotopes included in the models, the library used, the impact of isotopic concentration uncertainties in the CRC records, for the applicability of the selected CRC records for the TN-40 Prairie Island transportation casks.

In addition, use of CRC k_{eff} values from NUREG/CR-6951 may cause additional unknown uncertainty in the USL values for the following reasons: (1) the computational methods used to calculate the published CRC k_{eff} values (e.g., computer codes, nuclear data libraries, and modeling practices) are significantly different from what were used for the safety analysis models; (2) the computing platform and operating system used to calculate the NUREG/CR-6951 k_{eff} values were not the same computing platform and operating system used to calculate the k_{eff} values for the safety analysis models; (3) nuclear-safety-grade quality assurance practices were not followed in the generation of these k_{eff} values; and (4) uncertainties in the isotopic compositions for the CRC state-points have not been evaluated. The application justifies the use of published CRC k_{eff} values based on conservative k_{eff} results with the SCALE 44-group cross-section library, as compared with the results obtained with the 27- and 238-group cross-section libraries. Table 8.1 of NUREG/CR-6686 provides a comparison of average results for groups, organized by type of fissionable material, of LCEs for various cross-section libraries. For the LWR lattices, homogeneous LEU, Pu-239, and MOX LCEs, the differences between results for differing libraries are smaller than the uncertainties stated in Table 8.1. Consequently, the assertion that the data from NUREG/CR-6686 indicates that the 44-group library consistently yields higher k_{eff} values is not supported by the data. Additional confirmation is necessary to support the calculated USL values.

The applicant also needs to develop bias and the uncertainties associated with the cross sections used in SAS2H and KENO V.a calculations for TN-40 by modeling the specific reactor cores using the same cross sections.

Response to 6-7

SAR Section 6.4.4 addresses the sensitivity to the number of isotopes selected in the TN-40 criticality analysis. The applicability of the CRC benchmark is described in SAR Section 6.5.1.

7.0 PACKAGE OPERATION

7-1 Modify SAR Section 7.2.2 in order to describe the implementation of the operational restrictions which are necessary to prevent oxidation of the fuel during dry cell loading. Any limitations to prevent oxidation during unloading should be included by reference in the Operating Procedures.

Section 7.2.2 of the SAR suggests the possibility of unloading operations outside of a spent fuel pool (i.e., in a dry cell). The proposed operations descriptions are the same as for wet unloading in a spent fuel pool except for the removal of operations involving filling and draining the MPC with water. However, the operations overlook the prevention of fuel oxidation, a critical issue when spent fuel is exposed to an oxidizing gaseous atmosphere. The concerns expressed for fuel oxidation during loading in Interim Staff Guidance (ISG) No. 22, "Potential Rod Splitting Due To Exposure To An Oxidizing Atmosphere During Short-Term Loading Operations In LWR Or Other Uranium Oxide Based Fuel," also hold for unloading. ISG-22 discusses fuel oxidation, the conditions for which it can occur and means for its prevention. As stated in ISG-22, fuel oxidation can result in gross cladding breaches and create shielding, criticality and fuel dispersal concerns. The ISG further indicates that the oxidation concern extends to intact fuel as well, since intact fuel may have pinhole leaks and hairline cracks, which provide a path for the loading atmosphere to reach the fuel.

The applicant should provide a description of the essential operations and condition limitations through which fuel oxidation is prevented in Section 7.2.2. of the SAR. One way to prevent fuel oxidation is to limit dry cell unloading to only that fuel which is known to have no breaches (including pinhole leaks and hairline cracks). This limitation will necessitate the use of an appropriate method to ensure, to a high level of confidence, that a fuel assembly does not have any cladding breaches. As stated numerous times, the staff does not consider 4-sided visual inspections of an assembly to be sufficient for providing the necessary confirmation. Methods such as sipping, ultrasonic testing, and a review of reactor records can provide the necessary level of confidence.

For dry unloading of fuel containing cladding breaches, ISG-22 provides possible options to control and/or prevent fuel oxidation. One of these is to maintain the fuel rods in an inert environment. In developing the necessary operations and limitations, the applicant will need to consider impacts on other areas such as contamination control.

This information is needed to confirm compliance with 10 CFR 71.89.

Response to 7-1

The SAR is modified to only allow wet unloading. This is accomplished by deleting the note that appeared after step 7.2.2.4.

7-2 Clarify in the package operations that the cask trunnions (both the rear and the upper trunnions) are rendered inoperable for use as tie-down devices after the cask is placed onto the transport frame.

10 CFR 71.87(h) states that any structural part of the package that could be used to lift or tie down the package during transport must be made inoperable for that purpose unless it meets the criteria in 10 CFR 71.45 (including 71.45(a) for lifting attachments

and 71.45(b) for tie-down devices). It is not clear from the currently proposed package operations that the cask trunnions are handled in a manner that complies with 10 CFR 71.87(h).

This information is needed to confirm compliance with 10 CFR 71.87(h).

Response to 7-2

Two trunnion covers are welded to the front tiedown strap. This is depicted in drawing 10421-71-2 contained in SAR Appendix 1.4.1. When the tiedown strap is installed, the trunnion covers are located such that they prevent attachment to the trunnions. Thus the cover will prevent any attachment to the front trunnions while the tiedown straps are in place. Note that the rear cover is covered by the impact limiter during transport and thus does not require a separate cover.

7-3 Clarify the package operations for transporting an empty cask to ensure that the cask will meet the appropriate dose rate and contamination limits.

The currently proposed package operations for empty casks include steps 7.3 13 and 7.3.18, which appear to contradict each other. Hence, it is not clear how the applicant intends to ship an empty cask. For casks that are shipped as empty packages per DOT regulations, the cask must meet the regulations in 49 CFR 173.428, which set forth the dose rate and contamination limits for empty packages. However, empty casks may be transportable under 49 CFR 173.415(b), provided the applicable limits/criteria are met. The application's description of package operations should clearly indicate the designation under which the empty cask will be shipped and describe the operations necessary to ensure the DOT regulations are met that are applicable to packages under that designation.

This information is needed to ensure compliance with 10 CFR 71.35(c) and 71.87(f).

Response to 7-3

The procedure for preparing an empty cask for shipment has been modified. Step 7.3.1 now includes a check to verify that the cask is empty and requires decontamination of both inner and outer surfaces to a level that meets the limits given in 49CFR173.428, Empty Class 7 (radioactive) materials packaging. In addition, steps 7.3.13 and 7.3.18 have been revised to reference 49CFR173.428 limits rather than those of 10CFR71.47 or 10CFR71.87.

7-4 Modify the package operations descriptions to state the following:

a. Step 7.1.1.1 should verify the assembly to be loaded meets the specifications given in the Certificate of Compliance (and not Section 1.2.3 of the application). Step 7.4.1.1 should also be modified in the same manner.
b. Step 7.1.1.13 should be done with the upending/downending frame; the cask is already off of the transport frame per step 7.1.1.10.

This information is needed to confirm compliance with 10 CFR 71.87.

Response to 7-4

a) Steps 7.1.1.1 and 7.4.1.1 have been modified to reference the Certificate of Compliance 71-9313 as the basis for determining if the appropriate fuel is being loaded.

b) Step 7.1.1.13 has been revised so that it no longer refers to the shipping frame.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8-1 Modify Section 8.1.1 to state that dimensions and tolerances are confirmed to meet the drawings referenced in the Certificate of Compliance (CoC).

The CoC is the document that describes the requirements the package must meet and includes a listing of the licensing drawings (including revision numbers). The fabricated package must meet the specifications in these referenced licensing drawings.

This information is needed to confirm compliance with 10 CFR 71.85(c).

Response to 8-1

Section 8.1.1 has been expanded to include the statement that the dimensions and tolerances contained in the drawings referenced in the Certificate of Compliance have been met.

8-2 Incorporate also the following two structural tests into Section 8.1.2:

a. Fuel basket cell wall fusion weld qualification tests as described in Sections 2.1.2.2 and 2.10.5.4 of the SAR.

   The fusion weld qualification program, including its justifications as an ASME Code alternative, is being proposed for demonstrating adequate structural acceptance for the fuel basket design. As a structural strength test program, it must be captured in Section 8.1.2 of the SAR.

b. Trunnion tests performed to the American National Standard Institute (ANSI) N14.6 standard, as provided in NUREG-1617, Section 8.2.4.3 must be presented in Section 8.1.2 of the application. Specifically, this must include testing the front trunnions to 150% of its design capacity, as an exception to the standard approved previously in the NRR safety evaluations for the Prairie Island fuel.

   The NUREG-1617 trunnion test standard and the basis for its exception for the front trunnions must be captured in Section 8.1.2 of the application.

This information is needed to confirm compliance with 10 CFR 71.93(b).

Response to 8-2

a) *The fusion weld testing has been incorporated into Section 8.1.2. Reference is made to the TN-40 basket drawing that calls out the testing.*
b) *Specification of the trunnion testing has been expanded in Section 8.1.2 to reference the previous exception to ANSI N14.6 requirements.*

8-3 Modify Section 8.1.5, "Shielding Tests," to include a test of the as-fabricated neutron shield for each cask.

The currently proposed section describes chemical analysis to ensure composition and density and states qualification tests will be performed for personnel and procedures. However, there is no acceptance test to show the neutron shielding of the as-fabricated cask performs as designed. Such a test is necessary to ensure each cask meets the requirements described in 10 CFR 71.85(a) and (c). A test comprised of measurements with a check source or the loaded contents for the first use and a 6x6 inch test grid over the entire neutron shield surface compared to the calculated dose rates is one satisfactory method for acceptance testing. The proposed test method should be properly justified as satisfying the acceptance test requirement. Further guidance regarding shielding acceptance tests is contained in NUREG/CR-3854, "Fabrication Criteria for Shipping Containers." For those casks that are already fabricated and in use for storage, the applicant should propose a test that fulfills the acceptance test requirement, justifying how the test ensures 10 CFR 71.85(a) and (c) are met. Staff notes that measurements that are used to determine compliance with the dose limits for transportation (i.e., the limits in 10 CFR 71.47) do not demonstrate that the cask's neutron shield is fabricated and performing according to the approved design. However, these measurements may be found to be acceptable for this purpose for an already loaded (as of the date of the initial certificate) cask if the cask is used only for transporting the contents for which those measurements are taken (i.e., the cask is limited to a single use).

This information is needed to ensure compliance with 10 CFR 71.85(a) and 71.85(c), and 71.93(b).

Response to 8-3

The TN-40 casks are currently fabricated/loaded under a site specific Part 72 license. Cask surface dose rate measurements (both neutron and gamma dose rates) after loading of the cask are required by Part 72 Technical Specifications. Results of these measurements demonstrate the adequacy of the as-fabricated gamma and neutron shielding and can be used as test data. The shielding is not expected to lose its effectiveness under long term storage conditions based on prior experience with loaded storage casks. In addition, during storage of the spent fuel, the cask does not experience dynamic loads that could cause failure of the shielding. Thus, a periodic test during the storage usage is not performed.

In addition, prior to transport of the package, gamma and neutron dose rate measurements are to be taken over the complete cask surface to demonstrate the continued performance of the shielding. This documents both the shielding design and durability of the shielding materials. These measurements also demonstrate the loaded cask meets DOT shipping requirements.

Periodic testing of the shielding is not required because the Part 71 transport license application (Chapter 1) limits the use of the TN-40 for a one-time transportation use.

SAR Sections 1.1 and 7.4.1.24 are modified accordingly.

8-4 Justify why there is no need to perform any test after fabrication or prior to the use of each packaging to assure presence and functioning of the neutron absorbers in the basket.

Section 8.1.6 of the SAR does not address the testing of the neutron absorbers in the basket after fabrication or prior to the use of the packaging. Furthermore, no verification of the presence and proper condition of neutron absorbers are offered as part of maintenance program in Section 8.2 of the SAR.

This information is required to determine compliance with 10 CFR 71.85(a) and 71.87(g).

Response to 8-4

The fixed poison material (Neutron Absorber) in the TN-40 cask is BORAL®. The B-10 loading of this material is verified per Part 72 acceptance criteria during fabrication. The thermal neutron absorption and the total neutron fluence during storage conditions are insignificant compared to the number of B-10 atoms present. Therefore, no performance degradation is expected during storage.

For one time transportation, verification at the time of initial loading (under Part 72) is sufficient.

8-5 Provide procedures in Section 8.1.7 of the SAR to perform thermal acceptance tests prior to shipment on the TN-40.

Although the thermal analysis presented in Chapter 3 of the SAR is based on design configurations and thermal properties taken from industry recognized standards for the specified materials, Section 8.1.7 of the application did not address fabrication anomalies in critical heat-removing components, associated gaps between components, and uncertainties in the analytical models. Due to the decay heat load of the spent fuel, the insulative properties of the radial neutron shield, as well as uncertainties in calculations and fabrication, it is necessary to establish thermal acceptance tests. The staff needs to ensure the heat transfer capability of the package has been achieved during the fabrication process for each packaging prior to shipment.

Thermal acceptance tests should be performed on each unit after fabrication and during interim storage prior to shipment. For each of the TN-40 casks that have already been placed in interim storage, the periodic maintenance test may be used as the acceptance test prior to the first transport, with justification (see RAI 8-6). Section 8.1.7 of the SAR should include the method of testing, equipment used in the test, acceptance criteria, and the course of action if the acceptance criteria have not been met.

This information is needed to determine compliance with 10 CFR 71.85(a) and 71.93(b).

Response to 8-5

Regulation 10 CFR 71.85(a) requires assurance that no cracks, pinholes, uncontrolled voids, or other defects reducing significantly the effectiveness of the packaging exist prior to the shipment.

Since the TN-40 casks subject to this application are already fabricated and loaded, and will be used only for one time shipment, TN proposes the following to comply with the above regulation:

1) Perform a radiological survey over the cask outer surface as required in Chapter 7.0 per Operation Step 7.4.1.34 prior to transport. This survey assures compliance with 10 CFR 71.47 and 71.87 and will indicate the existence of any excessive defects, cracks, or void space through the cask shells.
2) Perform a thermal survey over the outer surface to determine the maximum cask outer shell temperature prior to transport and to show that there are no anomalies (hot or cold spots) in the surface temperature profile. Comparison of the measured temperature from the thermal survey to the calculated maximum cask outer shell temperature using the analytical model described in the SAR based on the ambient temperature indicates the sufficiency of the thermal performance prior to shipment.

3) Perform a leak test of the seals at final destination. This test is an indicator that the containment properties of the cask were not affected by the thermal performance of the cask.

In addition to the above proposal, it should be noted that TN-32 cask, which is similar in design and construction to TN-40 cask, was previously fabricated and tested by TN. The major dimensions of these casks are compared in the following table.

Table 8-5.1 Dimension of TN-32 and TN-40 Casks

	TN-32 SAR Drawing 1049-70-1, Rev. 6	TN-40 SAR Drawing 10421-71-3, Rev. 1
Cask ID (in)	68.75	72.00
Cask OD (in)	97.75	101.00
Cavity Length (in)	163.25	163
Basket Length (in)	160	160
Length of Radial Neutron Shield (in)	154.5	154
Inner Shell Thickness (in)	1.5	1.5
Gamma Shield Thickness (in)	8.0	8.0
Neutron Shield Thickness (in)	4.5	4.5
Outer Shell Thickness (in)	0.5	0.5
Thickness of Al boxes in NS (in)	0.12	0.12

The thermal tests of TN-32 casks provided to USNRC in TN Letter E-18578, "TN-32 Cask Thermal Testing", Docket 72-1021, dated December 1, 2000 have shown that the analytical model accounts conservatively for the uncertainties in the fabrication and tolerances and adequately models the insulation properties.

Since the fabrication methods, the shell thicknesses, and the analytical models for TN-32 and TN-40 casks are similar, the same conclusion would be expected from a thermal test of TN-40 cask.

In addition to the above thermal tests, an examination of the TN-40 analytical model shows that a margin of more than 250 °F exists between the maximum fuel cladding temperature (495 °F, SAR Table 3-1) and the allowable limit of 752 °F during normal conditions of transport (NCT). This margin is calculated based on a total heat load of 22 kW in the TN-40 thermal analytical model, which is 3 kW (~16%) higher than the allowable heat load. This adds to the conservatism in evaluation of the heat transfer capability of TN-40 transport cask.

All these measures ensure that the thermal analysis of TN-40 transport cask bounds the effects of fabrication tolerances, associated gaps, and adequately models the insulation properties of the radial neutron shield.

SAR Sections 3.4.7 and 8.1.7 are revised to reflect the above discussion.

8-6 Modify Section 8.2.4 to provide a periodic test of the neutron shield.

The application currently relies upon the measurements versus the 10 CFR 71.47 limits done prior to each shipment to demonstrate continued shield integrity and efficacy. However, these measurements only ensure the cask meets the limits for a particular shipment. Thus, if the cask is used only for the contents for which these measurements are done (i.e., the cask is limited to a single use), these measurements may serve as the periodic neutron shield test. However, if a cask is intended to be used for multiple shipments with different contents (as may be implied from the inclusion of package operations for transporting an empty cask), measurements made versus the 10 CFR 71.47 limits are insufficient to ensure the continued neutron shield integrity and efficacy, which is the purpose of the periodic test. Therefore, the applicant needs to propose a test and test frequency which fulfills the periodic test objective and provide justification as to how and why the proposed test and frequency are sufficient to meet this objective. For example, in the past, staff have found to be acceptable verifications that are based upon multiple measurements at multiple axial locations for either loaded contents or a check source which are compared to calculated values for the loaded contents/check source and that are performed within five years prior to each shipment.

This information is needed to confirm that the maintenance program is adequate to assure that packaging effectiveness is maintained throughout the packaging's service life to ensure continuing compliance with 10 CFR Part 71.87.

Response to 8-6

The TN-40 casks are currently fabricated/loaded under a site specific Part 72 license. Cask surface dose rate measurements (both neutron and gamma dose rates) after loading of the cask are required by Part 72 Technical Specifications. Results of these measurements demonstrate the adequacy of the as-fabricated gamma and neutron shielding and can be used as test data. The shielding is not expected to lose its effectiveness under long term storage conditions based on prior experience with loaded storage casks. In addition, during storage of the spent fuel, the cask does not experience dynamic loads that could cause failure of the shielding. Thus, a periodic test during the storage usage is not performed.

In addition, prior to transport of the package, gamma and neutron dose rate measurements are to be taken over the complete cask surface to demonstrate the continued performance of the shielding. This documents both the shielding design and durability of the shielding materials. These measurements also demonstrate the loaded cask meets DOT shipping requirements.

The Part 71 transport license application has been revised to limit the use of the TN-40 to a single transport of the spent fuel that was stored in the cask under Part 72. This makes periodic testing during the transport usage of the cask unnecessary.

SAR Sections 1.1 and 8.2.5 are modified accordingly.

8-7 Provide procedures in Section 8.2.5 of the SAR to perform thermal maintenance tests prior to shipment on the TN-40.

Thermal maintenance tests should be performed on each unit. A typical time interval for a thermal maintenance test is within five years prior to transport. Due to the physical changes of the package geometry and material properties during its service life, the staff needs to ensure the heat transfer capability for each packaging is adequate during its service life prior to shipment. Section 8.2.5 of the SAR should include the method of testing, equipment used in the test, acceptance criteria, and the course of action if the acceptance criteria have not been met.

This information is needed to determine compliance with 10 CFR 71.87(b) and 71.93(b).

Response to 8-7

The cask will see very small loads under storage conditions. Because of this and because of the materials used to fabricate the cask, the thermal properties of the cask are not expected to change significantly under "normal" long term storage conditions (Part 72).

For one-time transportation, surface temperature verification will be utilized as an acceptable test prior to transportation. See also Response 8-5 above.

SAR Sections 1.1 and 8.2. are modified accordingly.

Enclosure 8 to TN E-28034

Public Versions of Changed Drawings, Appendix 2.10.7 Pages, and Chapter 6 Pages, for the TN-40 Application Safety Analysis Report, Revision 3

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SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M-1994.

TOLERANCES

DECIMALS	.XX	±.12
	.XXX	±.030
ANGLES	±.5°	
MACHINED SURFACES	250	

BREAK AND DEBURR ALL SHARP EDGES

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

TN-40 TRANSPORT PACKAGING
PARTS LIST AND NOTES

DRAWING NO.	SCALE	SHEET	REVISION
10421-71-1	NONE	1 OF 1	2

SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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BREAK AND DEBURR ALL SHARP EDGES

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

TN-40 TRANSPORT PACKAGING
TRANSPORT CONFIGURATION

DRAWING NO.	SCALE	SHEET	REVISION
10421-71-2	NONE	1 OF 2	2

SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
10421-71-2	2 OF 2	2

SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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TOLERANCES

DECIMALS .XX ±.12 .XXX ±.030

ANGLES ±.5°

MACHINED SURFACES 250

BREAK AND DEBURR ALL SHARP EDGES

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

TN-40 TRANSPORT PACKAGING GENERAL ARRANGEMENT

DRAWING NO.	SCALE	SHEET	REVISION
10421-71-3	NONE	1 OF 1	1

Appendix 2.10.7
STRUCTURAL EVALUATION OF THE FUEL ROD CLADDING UNDER ACCIDENT IMPACT

PROPRIETARY INFORMATION WITHHELD UNDER 10CFR2.390

CHAPTER 6
CRITICALITY EVALUATION

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the isotopic composition of the burned fuel as a function of fuel design, initial enrichment, burnup, and cooling time using an assumed bounding reactor operating history. In addition, the benchmarking method to determine code biases is different. For the criticality code, additional benchmarks are required to account for the burned fuel composition. An additional bias (correction factors) for the depletion code, which determines the isotopic composition of the burned fuel, must also be addressed in the evaluation. Further, an additional bias for the criticality analysis code that accounts for the reactivity effect of the burned fuel must also be addressed in this evaluation.

The depletion calculations determine the isotopic composition of the burned fuel with the SAS2H control module of SCALE-4.4 [1]. The bias due to the *fission product absorber* isotopic composition of the fuel is accounted for by adjusting the calculated isotopic content based on comparison with available measure isotopic data of burned fuel from a variety of reactors and operating histories. The correction factors are based on the SAS2H benchmarks of measured data. *The bias due to the actinide fissile isotopic composition of the fuel is accounted for by a direct difference method [13]. The two biases are documented in Section 6.3.2. The criticality calculations determine k_{eff} with the CSAS25 control module of SCALE-4.4.* The bias due to the criticality code with the additional benchmark data to account for the composition of the burned fuel is also included in Section *6.5.2.*

A series of scoping calculations with the CSAS25 control module of SCALE-4.4 determines the most reactive configuration for the basket and assembly location. *It also* evaluates all of the eligible fuel assembly designs allowed for transport in the TN-40 package *and determines that* the Westinghouse 14x14 Standard fuel assembly *is the design basis fuel assembly* because it is the most reactive fuel assembly authorized for transport. *The results of those scoping calculations are documented in Section 6.4.2.*

With the CSAS25 control module of SCALE-4.4, a series of criticality calculations analyzes a set of bounding configurations *that are determined based on the most reactive configuration. Those bounding configurations are used to evaluate the horizontal bias, the TN-40 loading curve and fuel misload evaluations. They are characterized by* initial enrichment, assembly average burnup, and a minimum cooling time *in addition to the geometric arrangement in the most reactive configuration. The results of those criticality calculations are documented in Section 6.4.3. A series of sensitivity calculations has also been performed with the CSAS25 control module of SCALE 4-4. It addresses some specific issues associated with fission products and actinides. The results of those sensitivity calculations are documented in Section 6.4.4.*

The main results of the criticality calculations are shown in Table 6-1 and Figure 6-1. The *minimum assembly average burnup* as a function of *initial enrichment* required to ensure subcriticality is shown in Table 6-1 for fuel assemblies irradiated with and without BPRAs. A polynomial function is utilized to fit this data so that the required assembly burnup for all assembly enrichments that lie in between the minimum and maximum can be calculated. A separate curve is fit for fuel assemblies with and without BPRAs. A burnup curve based on the results in Table 6-1 is shown in Figure 6-1. The "acceptable" region in the curve pertains to those fuel assemblies with a burnup-enrichment combination AND a minimum cooling time greater than *30* years, that are eligible for transportation in the TN-40 cask.

*In conclusion, t*he results of the *criticality calculations and the sensitivity calculations* demonstrate that the maximum k_{eff}, including statistical uncertainty, is less than the USL determined from a statistical analysis of benchmark criticality experiments and includes an allowance for a burned fuel. The statistical analysis procedure includes a 95% confidence band with an administrative safety margin of 0.05.

6.2 Package Fuel Loading

The TN-40 Cask is capable of transporting 40 undamaged Westinghouse 14x14 (WE 14) class of PWR fuel assemblies irradiated at the Prairie Island Nuclear Generating Plant with or without Non Fuel Assembly Hardware (NFAH). Burnable Poison Rod Assemblies (BPRAs) and rod cluster control assemblies (RCCAs or Control Rods) are the only NFAH that are discussed in this evaluation since they bound all other NFAH. Each BPRA typically consists of 4, 8, 12, or 16 burnable poison (BP or discrete BP) rods. Each RCCA consists of 16 control rods or fingers. The fuel assemblies considered as authorized contents (as discussed in Chapter 1) include those listed in Table 6-2. Table 6-2 also lists the fuel parameters for the PWR fuel assemblies. The design basis fuel assembly for the TN-40 cask criticality analysis was determined to be the WE 14 Standard fuel assembly.

Proprietary Information Withheld Pursuant to 10 CFR 2.390

6.6 References

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4. U.S. Nuclear Regulatory Commission, "Criticality Benchmark Guide for Light-Water-Reactor fuel in Transportation and Storage Packages," NUREG/CR-6361, Published March 1997, ORNL/TM-13211.
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10. U.S. Nuclear Regulatory Commission, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," NUREG/CR-6801, Published March 2003, ORNL/TM-2001/273.
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14. Oak Ridge National Laboratory, "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analysis," ORNL/TM-12667, Published March 1995.

15. CAL-UDC-NU-000011 Rev A, , "Three Mile Island Unit 1 Radiochemical Assay Comparisons to SAS2H Calculations", Office of Civilian Radioactive Waste Management, U.S. Department of Energy, April 2002.

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17. Radulescu G, Mueller D. E. and J. C. Wagner, "Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit," Oak Ridge National Laboratory, January 2008, ORNL/TM-2006-87, NUREG/CR-6951.

18. Criticality Model, CAL-DS0-NU-0000003 REV 00A, Bechtel SAIC Company, Las Vegas, Nevada, 2004.

19. S. M. Bowman, W.C. Jordan, J. F. Mincey, C.V. Parks, and L. M. Petrie, "Experience with the SCALE Criticality Safety Cross-Section Libraries," Oak Ridge National Laboratory, NUREG/CR-6686, Published October 2000, ORNL/TM-1999/322.

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Table 6-1
Minimum Burnup as a Function of Enrichment

Maximum Assembly Average Initial Enrichment (wt. % U-235)	Minimum Assembly Average Burnup (GWD/MTU)
Assemblies with BPRAs	
2.00	*11.5*
2.30	*17.0*
2.55	*19.5*
2.85	*23.5*
3.00	*26.0*
3.30	*30.0*
3.55	*31.5*
3.85	*36.0*
Assemblies without BPRAs	
2.00	10.5
2.30	15.5
2.85	21.5
3.30	28.0
3.55	*30.0*
3.85	34.5

A minimum cooling time of *30* years is required for loading fuel assemblies prior to transport with burnup.

The mathematical formula to determine the minimum burnup as a function of initial enrichment for fuel assemblies with BPRAs is shown below:

$$B = -1259.8 * E^2 + 20242 * E - 23617$$

where, B = Minimum Assembly Average Burnup in MWD/MTU, and
E = Maximum Assembly Average Enrichment in wt. % U-235, ($E \geq 2.00$)

The mathematical formula to determine the minimum burnup as a function of initial enrichment for fuel assemblies without BPRAs is shown below:

$$B = -366.95 * E^2 + 14770 * E - 17200$$

where, B = Minimum Assembly Average Burnup in MWD/MTU, and
E = Maximum Assembly Average Enrichment in wt. % U-235 ($E \geq 2.00$)

Table 6-2
Parameters For PWR Assemblies For Shipment

Manufacturer	Array	Version	Active Fuel Length (in)	Number of Fuel Rods per Assembly	Fuel Rod Pitch (in.)	Fuel Pellet OD (in)
Exxon/ANF	14x14	Standard	144	179	0.556	0.3565
Exxon/ANF	14x14	High BU	144	179	0.556	0.3565
Exxon/ANF	14x14	Top Rod	144	179	0.556	0.3505
WE	14x14	Standard	144	179	0.556	0.3659
WE	14x14	OFA	144	179	0.556	0.3444

Manufacturer	Array	Version	Clad Thickness (in)	Clad OD (in)	Guide/ Instrument Tube OD (in)	Guide/ Instrument Tube ID (in)
Exxon/ANF	14x14	Standard	0.0300	0.424	16@0.541 1@0.424	16@0.507 1@0.374
Exxon/ANF	14x14	High BU	0.0310	0.426	16@0.541 1@0.424	16@0.507 1@0.374
Exxon/ANF	14x14	Top Rod	0.02950	0.417	16@0.541 1@0.424	16@0.507 1@0.374
WE	14x14	Standard	0.0243	0.422	16@0.539 1@0.422	16@0.505 1@0.3734
WE	14x14	OFA	0.0243	0.400	16@0.528 1@0.4015	16@0.490 1@0.3499

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Table 6-21
USL-1 Results

Parameter	Range of applicability	Formula for USL-1 (0.05 Δk_{eff} Margin)
U Enrichment (wt% U-235) (fresh fuel experiments)	2.35 – 5.74	0.9406 + (9.7267E-04)*X (X < 3.7186) 0.9442 (X >= 3.7186)
U Enrichment (wt% U-235) (all experiments)	2.35 – 5.74	*0.9389 + (7.3145E-04)*X (X < 4.5697)* *0.9422 (X >= 4.5697)*
Pu Enrichment (wt% Pu)	2.0 – 6.6	0.9424
Fuel Rod Pitch (cm) (fresh fuel experiments)	1.10 – 2.64	0.9351 + (5.2323E-03)*X (X < 1.7753) 0.9443 (X >= 1.7753)
Fuel Rod Pitch (cm) (all experiments)	1.10 – 2.64	*0.9330 + (5.2634E-03)*X (X < 1.8164)* *0.9425 (X >= 1.8164)*
Water/Fuel Volume Ratio (fresh fuel experiments)	0.38 - 10.8	0.9416 + (7.3797E-04)*X (X < 2.1094) 0.9431 (X >= 2.1094)
Water/Fuel Volume Ratio (all experiments)	0.38 - 10.8	*0.9387 + (9.2625E-04)*X (X < 2.6949)* *0.9412 (X >= 2.6949)*
Assembly Separation (cm) (fresh fuel experiments)	1.64 – 20.78	0.9410 + (4.9375E-04)*X (X < 6.9867) 0.9444 (X >= 6.9867)
Energy of the Average Lethargy for Fission (EALF) (fresh fuel experiments)	0.0083 – 1.39	0.9434 (Entire Range)
Energy of the Average Lethargy for Fission (EALF) (all experiments)	0.0083 – 1.39	*0.9421 (X <= 0.2069)* *0.9427 + (-3.0241E-03)*X (X > 0.2069)*

Table 6-22
USL Determination for Criticality Analysis

Parameter	Value from Limiting WE 14x14 Analysis	Bounding USL
Enrichment (wt. % U-235) (all benchmarks)	2.25 (minimum)(1)	0.9406
Enrichment (wt. % U-235) (fresh fuel benchmarks)	2.00 (minimum)(1)	0.9425
Enrichment (wt. % Pu) (fresh fuel benchmarks)	Not relevant since there is no variation in the USL	0.9424
Pin Pitch (cm) (all benchmarks)	1.412	0.9403
Pin Pitch (cm) (fresh fuel benchmarks)	1.412	0.9424
Water to Fuel Volume Ratio (all benchmarks)	1.610 (2)	0.9402
Water to Fuel Volume Ratio (fresh fuel benchmarks)	1.610 (2)	0.9427
Assembly Separation (cm) (fresh fuel benchmarks)	1.92 (3)	0.9419
Energy of the Average Lethargy for Fission (EALF) (all benchmarks)	0.35 (4)	0.9416
Energy of the Average Lethargy for Fission (EALF) (fresh fuel benchmarks)	0.35 (4)	0.9434

1) Extrapolation of the USL-1 formula is performed at this enrichment to determine the minimum USL since the k_{eff} data showed no trending with enrichment.
2) The water to fuel volume ratio is calculated using 179 rods.
3) Separation Distance = 2*(0.09") + 2*(0.250) + 0.075" = 0.755" ~ 1.92 cm, calculated with nominal dimensions for the stainless steel in the fuel compartment, nominal boral and aluminum plate width and inward fuel assembly positioning.
4) Examination of the results shows that the value is between 0.25 and 0.35 and hence a conservative value that produces the minimum USL was chosen even though there is no variation in the USL with fresh fuel experiments.

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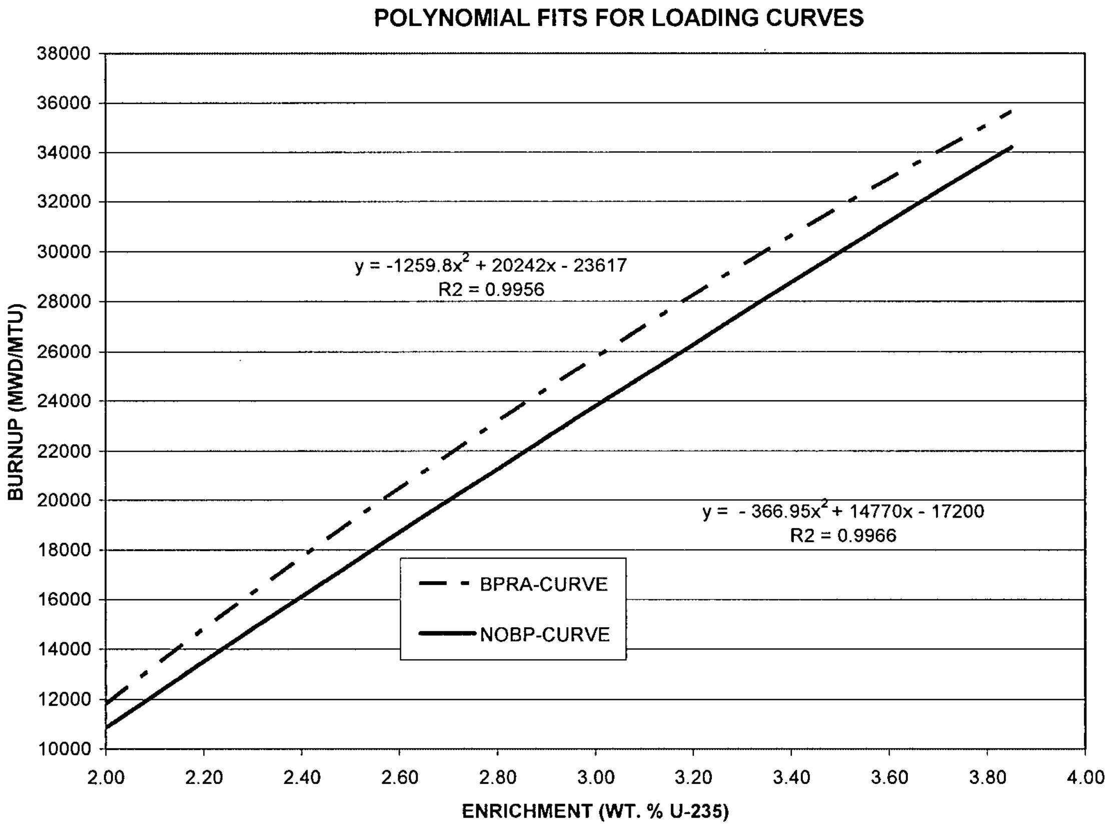

Note: The "ACCEPTABLE" region for the BPRA-CURVE always lies above the curve and corresponds to a burnup that is greater than or equal to the burnup loading curve. The "ACCEPTABLE" region for the NOBP-CURVE always lies above the curve and corresponds to a burnup that is greater than or equal to the burnup loading curve.

Figure 6-1
TN-40 Loading Curve
(with and without BPRAs)

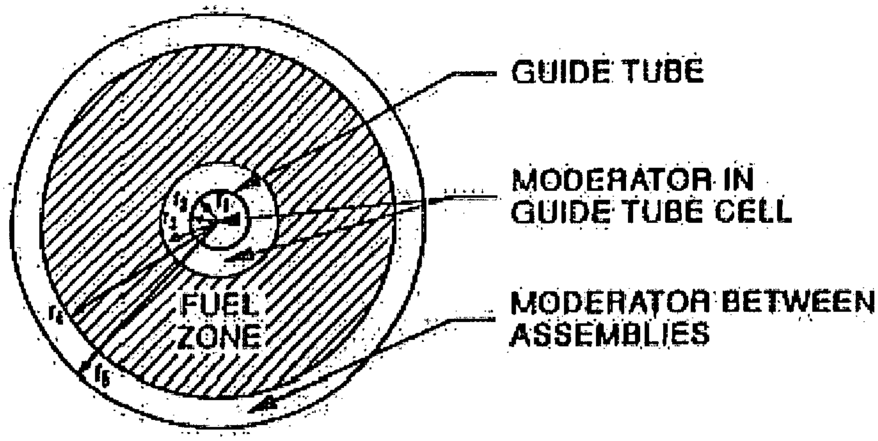

Figure 6-2
Example SAS2H Model

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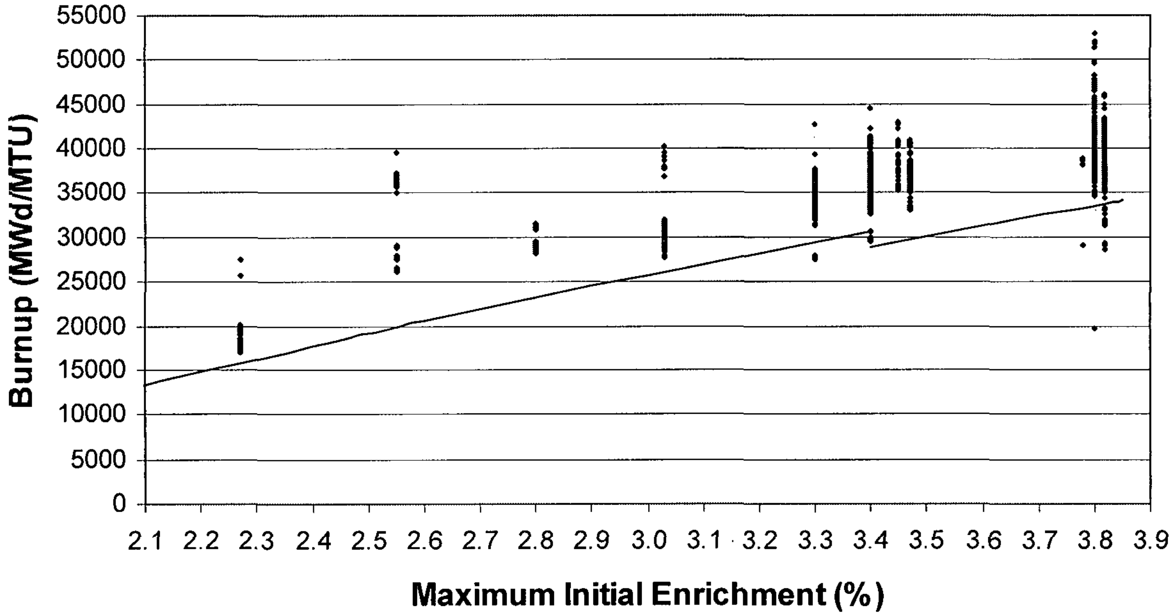

Note: The maximum enrichment for the Westinghouse 14x14 Standard fuel type is 3.4 wt. % U-235

Figure 6-17
Fuel Assembly Inventory at Prairie Island

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Enclosure 18 to TN E-28034

Listing of Disk Numbering and Contents for Computer Files
(all files are Proprietary)

Disk ID No. (size)	Discipline	File Series (topics)	Number of Files
Disk 1 (DVD) (2.21 GB) (2 zip files)	Structural (Fuel)	001-benchmark.zip (Appendix 2.10.7– Input and output file for fuel End drop- LS-Dyna analysis)	43
		002_TN40_transport_0.25gap.zip (Appendix 2.10.7– Input and output file for fuel End drop- LS-Dyna analysis)	38
Disk 2 (DVD) (3.42 GB) (1 zip file)	Criticality	Folder: 001 - 28-nuclide-sensitivity-KENO (The SCALE 4.4 sensitivity calculation on the CRC KENO input by reducing the actinides and fission products from 68-71 to 28)	10
		Folder: 002 - actinide-bias-KENO (The calculation of Δk between the measured-actinide-concentration cases and the calculated-actinide concentration cases, and the Δk values are used to determine the actinide bias)	586
		Folder: 003 - assay-SAS2H (The input and output of the SCALE 4.4 SAS2H 44-group depletion runs for benchmarking the Calvert Cliff, Obregheim, Robinson, Takahama, TMI, Trino and Turkey radiochemical assay measurements)	148
		Folder: 004 - crystal-river-benchmark-SAS2H-KENO-44G (The input and output files of the SCALE SAS2H 44-group depletion runs for Crystal River State Point 2 and 3, and the input and output files of the SCALE 44-group KENO runs)	2,174
		Folder: 005 - crystal-river-benchmark-SAS2H-KENO-238G (The input and output files of the SCALE SAS2H 238-group depletion runs for Crystal River State Point 2 and 3, and the input and output files of the SCALE 238-group KENO runs)	1,088
		Folder: 006 - tn40-criticality-model-SAS2H-KENO (TN-40 Criticality Analysis SAS2H Model, and TN-40 Criticality Analysis KENO Model)	26
		Folder: 007 - tn40-USL-Functions (The input and output files of the USL function generation runs, and the input and output of the SCALE runs for LCE and CRC benchmarks)	470