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June 6, 2023
E-62022

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

Subject: Response to Request for Additional Information for the Application for Revision 4 to Certificate of Compliance No. 9313 for the Model No. TN-40 Packaging, Docket No. 71-9313

References: (1) TN Letter E-62022. "Application for Revision 4 to Certificate of Compliance No. 9313 for the Model No. TN-40 Packaging, Docket No. 71-9313" dated December 16, 2021 (Agencywide Documents Access and Management System [ADAMS] Package Accession No. ML21350A282)

(2) Letter to Don Shaw (TN) from Pierre Saverot (NRC), "Request for Additional Information for the Review of the Model No. TN-40 Package," dated February 9, 2023, Docket Number 71-9313, Enterprise Project Identifier (EPID) No. EPID L-2021-LLA-0086 (ADAMS Package Accession No ML23037A102) with attachment "Request for Additional Information for the review of the Model No. TN-LC Package Docket No. 71-9313" (ADAMS Package Accession No ML23037A103)

(3) Revision 3 to Certificate of Compliance No. 9313 for the Model No. TN-40

In accordance with 10 CFR 71.31, TN Americas LLC (TN) made a submission of an application to revise Certificate of Compliance (CoC) No. 9313 for the TN-40 packaging [1]. The NRC submitted a request for additional information (RAI) needed to continue the review of the application [2].

Enclosure 1 provides a proprietary version of the TN responses to the RAIs. Enclosure 9 provides a public version of the responses to the RAIs.

Changes to the Safety Analysis Report (SAR) are provided as Revision 17B in Enclosure 2, and Enclosure 3 provides changed pages that may be made available to the public with proprietary information withheld. SAR Revision 17B changes are indicated by italicized text and revision bars and, additionally, are gray shaded to distinguish them from Revision 17A changes. A consolidated SAR Revision 17 will be submitted upon completion of the NRC review.

Enclosure 4 provides a list of changes that are not associated with any of the responses to RAIs.

The NRC Electronic Information Exchange (EIE) system is used for submission of this application. A set of computer calculation files is included as Enclosure 6. Enclosure 5 provides a listing of these computer files. Because the computer calculation files exceed the size limit allowed by the NRC EIE application process, they are provided separately on a portable hard drive.

Proposed changes to the NRC Certificate of Compliance [3] are annotated and provided as Enclosure 7.

Certain portions of this submittal include proprietary information. In accordance with 10 CFR 2.390, TN Americas is providing an affidavit (Enclosure 8) requesting that this proprietary information be withheld from public disclosure.

Should the NRC staff have any questions or require additional information regarding this submittal, please contact Mr. Peter Vescovi by telephone at (336) 420-8325, or by e-mail at peter.vescovi@orano.goup.

Sincerely,

SHAW Donis Digitally signed by SHAW Donis
Date: 2023.06.06 05:04:30 -04'00'

Don Shaw
Licensing Manager
TN Americas LLC

cc: Pierre Saverot, Senior Project Manager, U.S. Nuclear Regulatory Commission
Scott Bomar, Senior Project Manager, Orano TN Americas

Enclosures:

1. RAIs and Responses (Proprietary)
2. TN-40 Transportation Packaging SAR, Revision 17B, Changed Pages and Drawings (Proprietary Version)
3. TN-40 Transportation Packaging SAR, Revision 17B, Changed Pages and Drawings (Public Version)
4. Additional Changes Not Associated with Any RAI
5. Listing of Computer Files Contained in Enclosure 6
6. Computer Files (Proprietary)
7. Proposed Certificate of Compliance No. 9313, Revision 4 Markup
8. Affidavit Pursuant to 10 CFR 2.390
9. RAIs and Responses (Public)

__Enclosure 1 to E-62022__

RAIs and Responses
(Proprietary)

Withheld Pursuant to 10 CFR 2.390

<u>Enclosure 2 to E-62022</u>

TN-40 Transportation Packaging SAR, Revision 17B,
Changed Pages and Drawings
(Proprietary Version)

Withheld Pursuant to 10 CFR 2.390

<u>Enclosure 3 to E-62022</u>

**TN-40 Transportation Packaging SAR, Revision 17B,
Changed Pages and Drawings
(Public Version)**

PUBLIC

TN-40

Transportation Package

Safety Analysis Report

Prepared by:

TN Americas LLC

Columbia, Maryland

Revision 17B

June 2023

Proprietary and Security Related Information
for Drawing 10421-71-1, Rev. 6B
Withheld Pursuant to 10 CFR 2.390

Proprietary and Security Related Information
for Drawing 10421-71-2, Rev. 3A
Withheld Pursuant to 10 CFR 2.390

8 7 6 5 4 3 2 1

H G F E D C B A

Proprietary and Security Related Information for Drawing 10421-71-7, Rev. 3C Withheld Pursuant to 10 CFR 2.390

Proprietary and Security Related Information
for Drawing 10421-71-40, Rev. 2A
Withheld Pursuant to 10 CFR 2.390

Proprietary and Security Related Information
for Drawing 10421-71-41, Rev. 2B
Withheld Pursuant to 10 CFR 2.390

LIST OF FIGURES

All indicated changes on this page are for Enclosure 4 Item 2

Figure 2-3
DELETED

Figure 2-4
DELETED

Figure 2-5
DELETED

Figure 2-6
DELETED

All indicated changes on this page are for Enclosure 4 Item 2

Material Properties

The materials used for TN-40 transport cask and their properties as a function of temperature are listed below [4].

Cask Component	Material	Temperature (°F)	Ultimate Strength (ksi)	Yield Strength (ksi)	Allowable S_m (ksi)	Young's Modulus E (psi)	Thermal Expansion α (in/in/°F)
Containment Inner Shell & Bottom Inner Plate	SA-203 Gr. E or Gr. D	70	65	37	21.7	27.8×10^6	6.27×10^{-6}
		200	65	33.9	19.6	27.1×10^6	6.54×10^{-6}
		300	65	32.7	19.6	26.7×10^6	6.78×10^{-6}
Flange and Lid Outer Plate	SA350 LF3 or SA-203 Gr. E	70	70	37.5	23.3	27.8×10^6	6.27×10^{-6}
		200	70	34.3	22.8	27.1×10^6	6.54×10^{-6}
		300	70	33.2	22.2	26.7×10^6	6.78×10^{-6}
Lid Shield	SA105 or SA516, Gr.70	70	70	36	23.3	29.5×10^6	5.73×10^{-6}
		200	70	33	21.9	28.8×10^6	6.09×10^{-6}
		300	70	31.8	21.3	28.3×10^6	6.43×10^{-6}
Gamma Shield Shell Cylinder & Bottom Shield	SA266 CL 4 or SA516, Gr.70 or SA-105	70	70	36	23.3	29.5×10^6	5.73×10^{-6}
		200	70	33	21.9	28.8×10^6	6.09×10^{-6}
		300	70	31.8	21.3	28.3×10^6	6.43×10^{-6}
Lid Closure Bolt	SA320 Gr. L43*	70	125	105	35	27.8×10^6	6.27×10^{-6}
		200	125	99	33	27.1×10^6	6.54×10^{-6}
		300	125	95.7	31.9	26.7×10^6	6.78×10^{-6}

Note: Lower strengths of alternate materials are listed in the above table.
* The lid bolt material specified in SAR drawing 10421-71-1 is SA540 Gr. B23*/B24* CL1. The only SA320 Gr. L43 material property used in this appendix is Young's modulus which is identical to that of SA540 Gr. B23*/B24* CL1. Thus the results provided remain valid. The lid bolt evaluation concerning delayed impact presented in Appendix 2.10.11 is based on SA540 Gr. B23*/B24* CL1.

All indicated changes on this page are for Enclosure 4 Item 6

2.10.2 LID BOLT ANALYSIS

2.10.2.1 Introduction

This Appendix evaluates the ability of the cask closure bolt to maintain a leak tight seal under events defined by Normal Conditions Transport (NCT) and the Hypothetical Accident Conditions (HAC). Also evaluated in this section are the bolt thread and internal thread stresses, and lid bolt fatigue. The stress analysis is performed in accordance with NUREG/CR-6007 [1].

The TN-40 cask lid closure arrangement is shown in Figure 2.10.2-1. The 4.5 in. thick lid with a 6.0 in. radiation shield is bolted directly to the shell flange by 48 high strength alloy steel 1.375 in. diameter bolts (with 1 ½ -8UN threaded portion). Close fitting alignment pins ensure that the lid is centered in the vessel. The bolt material is SA-540 Gr. B23*/B24* CL1.

The lid bolt analysis presented in this appendix is done in accordance with NUREG/CR-6007 and conservatively uses lid bolt material of SA-320 Grade L43 with yield strength of 105 ksi and tensile strength of 125 ksi at 70 °F. One exception to this is the minimum engagement length determination in Section 2.10.2.9 where the higher strength bolt material is used. The actual lid bolt material used is SA-540 Grade B23*/B24* CL1 with yield strength of 150 ksi and tensile strength of 165 ksi at 70 °F. The lid bolt evaluation due to delayed impact is presented in Appendix 2.10.11 and is based on the lid bolt material of SA-540 Grade B23*/B24* CL1.

The following ways to minimize bolt forces and bolt failures for shipping casks are taken directly from Reference [1], page xiii. All of the following design methods are employed in the TN-40 closure system.

- Protect closure lid from direct impact to minimize bolt forces generated by free drops (use impact limiters).
- Use materials with similar thermal properties for the closure bolts, the lid, and the cask wall to minimize the bolt forces generated by fire accident.
- Apply sufficiently large bolt preload to minimize fatigue and loosening of the bolts by vibration.
- Lubricate bolt threads to reduce required preload torque and to increase the predictability of the achieved preload.
- Use closure lid design which minimizes the prying actions of applied loads.
- When choosing a bolt preload, pay special attention to the interactions between the preload and thermal load and between the preload and the prying action.

The following lid bolt evaluations are presented in this section:

- Lid bolt torque
- Bolt preload

All indicated changes on this page are for Enclosure 4 Item 6

2.10.2.8.1 Assumptions

- CG over corner lid impact with internal pressure is the worst case condition.
- The force to seat the seals is 1399 lbs./in [2] and 2198 lbs/in [6]. The total load to seat the seal is 660,142 lbs or 1,037,164 lbs, but 700,000 lbs and 1,040,000 lbs will be used conservatively.
- The maximum allowable decompression of the seal is 0.040” [2].

2.10.2.8.2 Analysis

The finite element model from Appendix 2.10.1 is modified to include contact elements (CONTAC52) at the lid/cask axial interface, internal pressure, bolt preload, seal load and 30 foot drop conditions.

Gap elements (CONTAC52) were used to model the lid/cask axial interface. To get an accurate contact representation a 60 mil axial gap was included radially outwards of Ø77.25” (closest node at Ø78.10”) between the lid/cask axial interface. Figure 2.10.2-1 shows the lid/cask axial interface.

A pressure of 100 psi was applied to all internal surfaces. Bolt shank prestrain was calculated based on $\varepsilon = \sigma/E$, where σ is the bolt prestress (50 ksi per Section 2.10.2.2 above). The seal loads of 700,000 lbs and 1,040,000 lbs were applied via CONTAC52 elements. The stiffness for the gap element was calculated based on F=kx. The accident drop conditions were kept consistent with Appendix 2.10.1.

2.10.2.8.3 Results

Figure 2.10.2-2 plots the decompression of the seal as a function of circumferential location. The maximum decompression is 0.003 in. which is less then the allowable seal decompression of 0.040 in.

From the analysis results presented in the Figures and discussion, it can be concluded that during the CG over corner drop lid impact loading with internal pressure, the metal-to-metal contact exists at the Helicoflex seal. Since a seal exists around the circumference of the TN-40 vessel, the internal contents will not leak during a worst case loading condition.

2.10.2.9 Minimum Engagement Length for Bolt and Flange

For a 1½ – 8UN bolt, the material is SA-540 GR B23*/B24* CL1, with

Su = 165 ksi, and

Sy = 150 ksi (at room temperature)

All indicated changes on this page are for Enclosure 4 Item 6

2.10.4 FRACTURE TOUGHNESS EVALUATION OF THE TN-40 CASK

2.10.4.1 Introduction

This appendix documents the fracture toughness evaluation of the TN-40 cask.

2.10.4.2 Fracture Toughness Requirements of The Cask

The TN-40 cask material is a ferritic steel (penetration covers are stainless steel) and is therefore subject to fracture toughness requirements in order to assure ductile behavior at the lowest service temperature (LST) of –20 °F.

The inner shell and bottom inner plate are fabricated from SA-203 Gr. D or E plate material, 1.5 inches thick. The shell flange is 4.6 inches thick, fabricated from SA-350 Gr. LF3 forging material and the lid outer plate is 4.5 inches thick, fabricated from either SA-350 Gr. LF3 or SA-203 Gr. E material. The 1.5-inch lid closure bolts are fabricated from SA540 Grade B23/*B24*, Class1 material.

By interpolating between values provided in NUREG/CR-3826 [1] and NUREG/CR-1815 [2], the nil ductility transition temperatures (T_{NDT}) of the containment boundary materials are:

- Inner Shell and bottom inner plates (1.5 in.): -80 °F
- Shell Flange (4.6 in.): -137 °F
- Lid Outer Plate (4.5 in.): -125 °F

The fracture toughness requirements of the lid closure bolts meet the criteria of ASME Code, Section III, Subsection NB (Para. NB-2333) [3]. Charpy v-notch testing is performed at –20 °F. The acceptance criterion is that the material exhibits at least 25 mils lateral expansion (Table NB-2333-1). All the lid closure bolt materials meet the NB-2300 criteria.

The 1.5 in. plate material which forms the inner shell and inner bottom plate meets the NUREG fracture arrest criteria.

Drop weight and Charpy test measurements of the shell flange and lid outer plate from 24 TN-40 casks are shown in the table below.

2.10.5.2 TN-40 Fuel Basket Stress Analysis

2.10.5.2.1 Approach

Bounding inertial loads of 20g and 75g are applied for the NCT and HAC transport cask free drop cases respectively. 0°, 45° and 90° azimuth orientations are analyzed in order to bound all possible drop orientations.

Nonlinear analyses with bilinear material properties and small deflections were performed in ANSYS [1] for the critical azimuth side drop orientations. The membrane and membrane plus bending stresses were compared against S_m and 1.5 S_m stress criteria values [2] for the NCT cask drop. The membrane and membrane plus bending stresses were compared against 0.7 S_u and 0.9 S_u elastic-plastic analysis stress criteria values [2] for the HAC cask drop.

The TN-40 transport cask geometry is described in Section 2.10.5.1.1 and depicted in detail in the design drawings provided in Chapter 1, Appendix 1.4.1 (Drawings 10421-71-8 and -9). Nominal dimensions are used in the analyses that follow.

Side drop impact analyses using finite element methods are provided in Section 2.10.5.2.2 and the analytical analysis for the end drop impact is provided in Section 2.10.5.2.3. The thermal stress analysis of the fuel basket is provided in Section 2.10.5.2.4.

2.10.5.2.2 Basket Finite Element Analysis for Side Impact Loads

A) Finite Element Model Description

A three-dimensional finite element model of the fuel basket is constructed using shell elements. The overall finite element model of the fuel basket is shown in Figure 2.10.5-2. The fuel tubes, aluminum structural plates, aluminum outer plates and periphery plates are included in the model. For conservatism, the strength of Boral® plates in the basket is neglected by excluding these from the finite element model. However, their weight is accounted for by increasing the structural aluminum plate material densities.

Because of the large number of plates in the basket and large size of the basket, certain modeling approximations were necessary. In view of continuous support of fuel compartment tubes by the peripheral rails along the entire basket length during a side drop, only an 8.0 inch long slice of the basket and rail is modeled. At the two cut faces of the model, symmetry boundary conditions are applied (UY = ROTX = ROTZ = 0). The displacement constraints for the 0°, 45°, and 90° side drop angles are shown in Figure 2.10.5-3, Figure 2.10.5-5, and Figure 2.10.5-7 respectively. For clarity, symmetry displacement constraints are not shown.

All indicated changes on this page are for Enclosure 4 Item 1

Proprietary Information on This Page
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Since most of the TN-40 casks to be transported are already loaded and since each cask is limited to a one-time shipment, the following measurements taken prior to shipment ensure the adequacy of the cask thermal performance and compliance with 10 CFR 71.85(a) in lieu of fabrication tests:

Based on the thermal evaluation of the cask, the margins of the fuel cladding and seal temperatures when compared to the allowable limits are:
752 - 495 = 257 °F for fuel cladding, and 536 -195 = 341 °F for seals.

16. Mitchell S. Sweet, "Fire Performance of Wood: Test Methods and Fire Retardant Treatments," Fire Safety of Wood Products, USDA Forest Service. http://www.fpl.fs.fed.us/documents/pdf1993/sweet93a.pdf.

17. NUREG/CR-0322, "Effects of Temperature on the Energy Absorbing Characteristics of Redwood," Von Riesemann, Gues, SAND77-1589, Sandia Lab.

18. Transnuclear, Inc., Letter to USNRC, "TN-32 Cask Thermal Testing," Docket No. 72-1021, December 1, 2000, Transnuclear Document No. E-18578, Project 1066.

19. *EPRI, "High-Burnup Used Fuel Dry Storage System Thermal Benchmark Modeling Results, Round Robin Results," Final Report, April 2020*.

20. *North Anna Power Station ISFSI - Amendment No.5 to Materials License No.2507 for the North Anna Power Station Independent Spent Fuel Storage Installation (CAC No. L25047), NRC Accession Number: ML17234A534.*

21. *High Burnup Dry Storage Research Project Cask Loading and Initial Results, EPRI 3002015076, October 2019.*

22. *North Anna Power Station ISFSI - TN-32 DLBD, Revision 8, "HBU Demonstration Cask Design/Licensing Basic Document," NRC Accession Number: ML17109A457.*

All indicated changes on this page are for Enclosure 4 Item 3

Additional Changes Not Associated with Any RAI

Item 1

SAR Appendix 2.10.5, Section 2.10.5.2:

- This change is to delete the second sentence in the first paragraph of Section 2.10.5.2.1 of Chapter 2, Appendix 2.10.5 of the TN-40 SAR. This is an editorial correction for clarification regarding the 20g and 75g loads used in basket analysis.

Item 2

SAR Chapter 2:

List of Figures and Figures 2-3 through 2-6

- Editorial change to move the word “DELETED” to replace the title name of figures that were deleted during Revision 4 of the SAR. The List of Figures was also updated to reflect this change.

Item 3

SAR Chapter 3, Section 3.6:

- The version of the report date for Reference 19, EPRI, “High-Burnup Used Fuel Dry Storage System Thermal Benchmark Modeling Results,” was changed from “Draft, 2019” to “Final Report, April 2020” to reference the current version.

Item 4

Drawings:

10421-71-1 Revision 6B
10421-71-2 Revision 3A
10421-71-7 Revision 3C
10421-71-40 Revision 2A
10421-71-41 Revision 2B

- Changes are made to clarify bill of materials (BOM) items for the impact limiter, in particular the item numbers on drawings for the shared drawings (TN-40 and TN-40HT impact limiters).
- Changes made to material specifications of items that are not important to safety to a more general specification.
- Changes made to the description of features that are not important to safety to a more general description.
- Changes made to some dimensions that are not important to safety to reference dimensions.

Additional Changes Not Associated with Any RAI

Item 5

SAR Chapter 3, Section 3.4.7:

- Removes the requirements for a test to ensure adequacy of the cask thermal performance for the same reasons the thermal acceptance test was removed.

Item 6

SAR Appendix 2.10.1, Section 2.10.1.2

SAR Appendix 2.10.2, Sections 2.10.2.1 and 2.10.2.9

SAR Appendix 2.10.4, Section 2.10.4.2

SAR Appendix 2.10.11, Sections 2.10.11-3 and 2.10.11.4

SAR Drawing 10421-71-1 Revision 6B

- SAR Appendix 2.10 and Drawing 10421-71-1 have been revised to add Grade B24 material for lid bolt material. This adds flexibility to allow B23 or B24 for the grade of material allowed for lid bolts.

Listing of Computer Files Contained in Enclosure 6

Disk ID No. (size)	**Discipline**	**System/Component**	**File Series (topics)**	**Number of Files**
Enclosure 6 One Computer Hard Drive Total (52.2 GB) Structural (52.2 GB)	Structural	TN-40 Fuel End Drop	Appendix 2.12.9 Folder: \Structural \ Fuel End Drop Subfolders: Case 1 (645 files) Case 2 (646 files) Case 3 (645 files) Case 4 (645 files) Case 5 (652 files)	3,233 files

<u>Enclosure 7 to E-62022</u>

Proposed Certificate of Compliance No. 9313, Revision 4 Markup

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9313	~~3~~ 4	71-9313	USA/9313/B(U)F-96	1 OF	9

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

TN Americas LLC
~~7135 Minstrel Way, Suite 300~~
~~Columbia, Maryland 21045~~
7160 Riverwood Drive, Suite 200
Columbia, Maryland 21046

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

~~Transnuclear, Inc., application dated~~
~~August 7, 2006, as supplemented~~
TN Americas, LLC application dated

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5. (a) Packaging

(1) Model No.: TN-40

(2) Description: For descriptive purposes, all dimensions are approximated nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

The TN-40 is designed to transport up to 40 Pressurized Water Reactor (PWR) spent nuclear fuel assemblies discharged from the Prairie Island Nuclear Generating Plant (PINGP). These assemblies have been stored prior to shipment in the TN-40 package used as a dry storage cask at PINGP under SNM-2506. These 29 loaded packages at the PINGP are authorized for single use. The TN-40 packaging consists of a basket assembly, a containment vessel, a package body which also functions as the gamma shield and neutron shield, and impact limiters. A transport frame, which is not part of the packaging, is used for tie-down purposes.

The containment vessel components consist of the inner shell and bottom inner plate, shell flange, lid outer plate, lid bolts, penetration cover plates and bolts (vent and drain), and the inner metallic seals of the lid seal and the vent and drain seals. The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. The overall containment vessel length is approximately 170.5 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a nominal diameter of 72.0 in. and a length of 163 in.

Double metallic seals are used for the lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium above atmospheric pressure. The cask cavity is accessed via draining and venting ports. Double metallic seals are utilized to seal these two lid penetrations. The over-pressure (OP) port provides access to the volumes between the double seals in the lid and cover plates for leak testing purposes. The OP port cover is not part of the containment boundary.

5.(a)(2) Description (Continued)

The carbon steel packaging body, which also functions as the gamma shielding, is around the inner shell and the bottom inner plate of the containment vessel. The 8.0 in. and 8.75 in. gamma shield completely surround the containment vessel shell and bottom plate, respectively. A 6.0 in. thick shield plate is also welded to the inside of the 4.5 in. thick lid outer plate.

Radial neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield shell. The total radial thickness of the resin and aluminum is 4.50 in. The array of resin-filled containers is enclosed within a 0.50 in. thick outer steel shell. The aluminum container walls also provide a path for heat transfer from the gamma shield shell to the outer shell. A pressure relief valve is mounted on top of the resin enclosure to limit the possible internal pressure increase under hypothetical accident conditions.

The basket structure consists of an assembly of stainless steel cells joined by a fusion welding process and separated by aluminum and poison plates which form a sandwich panel. The panel consists of two aluminum plates separated by a poison plate. The aluminum plates provide the heat conduction paths from the fuel assemblies to the cask inner plate. The poison material provides the necessary criticality control. The opening of the cells is 8.05 in. x 8.05 in. which provides a minimum of 1/8 in. clearance around the fuel assemblies. The overall basket length (160.0 in.) is less than the cask cavity length to allow for thermal expansion and fuel assembly handling.

The impact limiters consist of balsa wood and redwood blocks encased in stainless steel plates. The impact limiters have an outside diameter of 144 in., and an inside diameter of 92 in. to accommodate the cask ends. The bottom limiter is notched to fit over the lower trunnions. The impact limiters are attached to each other using tie rods. The impact limiters are also attached to the outer shell of the cask with bolts. Each impact limiter is provided with fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has lifting lugs for handling, and support angles for holding the impact limiter in a vertical position during storage. An aluminum spacer is placed on the cask lid prior to mounting the top impact limiter to provide a smooth contact surface between the lid and the top impact limiter.

The nominal external dimensions, with impact limiters, are 261 in. long by 144 in. wide. The total weight of the package is 271,500 pounds (lbs.).

5.(a)(3) Drawings

The packagings are fabricated and assembled in accordance with the Transnuclear, Inc., Drawing Nos.:

10421-71-1, Rev. ~~5.~~ 6
10421-71-2, Rev. ~~2, sheets 1 and 2.~~
10421-71-3, Rev. ~~2.~~ 3
10421-71-4, Rev. 0.
10421-71-5, Rev. 0.
10421-71-6, Rev. 0.
10421-71-7, Rev. ~~2.~~ 3

5.(a)(3) Drawings (Continued)

10421-71-8, Rev. 0.
10421-71-9, Rev. 0.
~~10421-71-10, Rev. 0.~~
10421-71-40, Rev. ~~1.~~ 2
10421-71-41, Rev. ~~1.~~ 2
10421-71-42, Rev. ~~0.~~ 1
10421-71-43, Rev. ~~0.~~ 1
10421-71-44, Rev. ~~0.~~ 1

5.(b) Contents

(1) Type, form, and quantity of material

The characteristics of the contents of the TN-40 packaging are limited to the following.

I. Fuel shall be unconsolidated.

II. Fuel shall be limited to the following fuel types with specifications depicted in Table 1-1 of this certificate:

i. Exxon 14X14 Standard,
ii. Exxon 14x14 High Burnup,
iii. Exxon 14X14 TOPROD,
iii. Westinghouse (WE) 14X14 Standard, and
iv. Westinghouse 14X14 OFA.

III. Fuel shall only have been irradiated at the PINGP Unit 1, cycles 1 through 16 or Unit 2, cycles 1 through 15.

IV. The fuel assemblies from Unit 1, Region 4, i.e., assemblies identified as D-01 through D-40, are not authorized contents.

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

i. the BPRAs have cooled for a minimum of 25 years, and
ii. the maximum exposure of the BPRA(s) shall be 30,000 Megawatt-Days per Metric Ton of Uranium (MWd/MTU).

VI. Fuel may include thimble plug assemblies (TPAs) provided:

i. the minimum cooling time of the TPAs is 25 years,
ii. the maximum exposure of the TPA(s) shall not exceed 125,000 MWd/MTU, and

5.(b)(1) Contents – Type, form, and quantity of material (Continued)

iii. only TPAs that do not have water displacement rods extending into the active fuel may be loaded into the cask.

VII. The combined weight of a fuel assembly and any BPRA or TPA shall not exceed 1330 lbs.

VIII. The combined weight of all fuel assemblies, BPRAs, and TPAs in a single cask shall not exceed 52,000 lbs.

IX. The fuel shall not be a Damaged or Oxidized Fuel Assembly; a Damaged or Oxidized Fuel Assembly is:

- a partial fuel assembly from which fuel pins are missing unless dummy fuel pins are used to displace an amount of water equal to or greater than that displaced by the original pins;
- has known or is suspected to have gross cladding failures (other than pinhole leaks) or have structural defects sufficiently severe to adversely affect fuel handling and transfer capability; or
- has been exposed to air oxidation during storage, as indicated by maintenance or operating records

X. The number of assemblies in the container shall not exceed 40.

XI. The assembly average burnup shall be greater than or equal to the burnup calculated according to the following equations:

$B = -1{,}259.8X^2 + 20{,}242X - 23{,}617$; for fuel assemblies with BPRA insertions during depletion
$B = -366.95X^2 + 14{,}770X - 17{,}200$; for fuel assemblies without BPRA insertions during depletion

Where:
B = Burnup (MWd/MTU),
X = Initial enrichment (weight percent (wt%) U-235)

XII. The minimum cooling time for the fuel assemblies is 30 years. Content may include BPRAs or TPAs, which have a minimum cooling time of 25 years. Various combinations of minimum assembly average enrichment and maximum assembly average burnup prior to transport shall be in accordance with Table 1-2 in this certificate.

XIII. The maximum decay heat per fuel assembly shall not be more than 0.475 kW and 19 kW per package including the BPRAs and TPAs.

XIV. The boron-10 (B-10) in the Boral neutron poison plates in the basket must be uniformly distributed in the plates with a minimum areal density of 10 mg/cm^2.

5.(b)(1) Contents – Type, form and quantity of material (Continued)

XV. Integral Fuel Burnable Absorber is not an authorized content.

XVI. Fuel assemblies with the following irradiation history shall be authorized for transport:

i. The minimum average specific power shall be 14 MW/Assembly,
ii. The minimum hot leg average moderator density shall be 0.705 g/cm^3,
iii. The maximum hot leg average moderator temperature shall be 584 K (592°F),
iv. The average fuel temperature shall not exceed 901 K (1,162°F), and
v. The maximum average soluble boron concentration shall not exceed 675 parts per million based on an average over the limiting non-linear boron letdown curve.

XVII. The nominal length of the assembly axial blankets shall not exceed 6.2 in.

XVIII. The maximum cooling time of the spent fuel shall not exceed 200 years.

Table 1-1 Fuel Assembly Specifications[1,2]

	Fuel Assembly Type				
Fuel Characteristics	Exxon 14x14 Standard	Exxon 14x14 High Burnup	Exxon 14x14 TOPROD	WE 14x14 Standard	WE 14x14 OFA
Max. Active Fuel Length (in.)	144	144	144	144	144
Max. Number of Fuel Rods per Assembly	179	179	179	179	179
Max. Fuel Rod Pitch (in.)	0.556	0.556	0.556	0.556	0.556
Min. Clad Thickness (in.)	0.0300	0.0310	0.0295	0.0243	0.0243
Min. Clad Outer Diameter (OD) (in.)	0.424	0.417	0.426	0.422	0.400
Clad Material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Max. Pellet OD (in.)	0.3565	0.3565	0.3505	0.3659	0.3444
Min. Guide/Instrument Tube OD (in.)	16@0.541 1@0.424	16@0.541 1@0.424	16@0.541 1@0.424	16@0.539 1@0.422	16@0.528 1@0.4015
Max. Guide/Instrument Tube Inner Diameter (in.)	16@0.507 1@0.374	16@0.507 1@0.374	16@0.507 1@0.374	16@0.505 1@0.3734	16@0.490 1@0.3499
Max. Assembly and BPRA Length (in.)	161.3	161.3	161.3	161.3	161.3
Max. Assembly Width (in.)	7.763	7.763	7.763	7.763	7.763

Table 1-1 Fuel Specifications (Continued)

	Fuel Assembly Type				
Fuel Characteristics	Exxon 14x14 Standard	Exxon 14x14 High Burnup	Exxon 14x14 TOPROD	WE 14x14 Standard	WE 14x14 OFA
Maximum MTU/Assembly	0.380	0.380	0.380	0.410	0.380
Maximum Initial Assembly Average Enrichment (wt% U-235)	3.85	3.85	3.85	3.85	3.85
Maximum Assembly Average Burnup (MWd/MTU)	45,000 (see Table 1-2[2])	45,000 (see Table 1-2)	45,000 (see Table 1-2)	45,000 (see Table 1-2)	45,000 (see Table 1-2)
Minimum Cooling Time (years)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)	30 (see Table 1-2)

Notes:
1. Pre-irradiated nominal dimensions used in the design analyses and may be verified against as-built records.
2. Table 1-2 is located in this certificate.

Table 1-2 Required Minimum Cooling Time for Spent Fuel Assemblies[1,2,3,4]

	Minimum Assembly Average Initial Enrichment (wt.% U-235)								
Maximum Assembly Average Burnup (GWd/MTU)	2	2.25	2.35	2.75	3	3.25	3.4	3.6	3.85
17	30	30	30	30	30	30	30	30	30
18	30	30	30	30	30	30	30	30	30
19	30	30	30	30	30	30	30	30	30
20	30	30	30	30	30	30	30	30	30
21	30	30	30	30	30	30	30	30	30
22	30	30	30	30	30	30	30	30	30
23	30	30	30	30	30	30	30	30	30
24	30	30	30	30	30	30	30	30	30
25	30	30	30	30	30	30	30	30	30
26	30	30	30	30	30	30	30	30	30
27	30	30	30	30	30	30	30	30	30
28	30	30	30	30	30	30	30	30	30

Table 1-2 Required Minimum Cooling Time for Spent Fuel Assemblies (Continued)

Maximum Assembly Average Burnup (GWd/MTU)	Minimum Assembly Average Enrichment (wt.% U-235)								
	2	2.25	2.35	2.75	3	3.25	3.4	3.6	3.85
29			30	30	30	30	30	30	30
30			30	30	30	30	30	30	30
31			30	30	30	30	30	30	30
32			30	30	30	30	30	30	30
33			30	30	30	30	30	30	30
34			30	30	30	30	30	30	30
35			30	30	30	30	30	30	30
36			30	30	30	30	30	30	30
37			30	30	30	30	30	30	30
38			30	30	30	30	30	30	30
39			30	30	30	30	30	30	30
40			30	30	30	30	30	30	30
41			30	30	30	30	30	30	30
42			30	30	30	30	30	30	30
43					30	30	30	30	30
44						30	30	30	30
45						30	30	30	30

Notes:

1. For fuel characteristics that fall between the assembly average enrichment values in Table 1-2 of this certificate, use the next lower enrichment, and next higher burnup to determine minimum fuel cooling time.
2. Fuel assemblies that were located in the Rod Cluster Control Assembly control bank D position during Unit 1 cycle 1 and Unit 2 cycle 1 shall have a minimum cooling time of greater than 35 years.
3. The assembly average enrichment and the assembly average burnup are the enrichment and burnup averaged over the fuel assembly, including the axial blankets.
4. Fuel assemblies with a maximum average burnup and a minimum average enrichment for which no cooling time is specified in the table are not authorized contents.

5.(c) Criticality Safety Index: 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the "Operating Procedures" in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the "Acceptance Tests and Maintenance Program" in Chapter 8 of the application, as supplemented.

(c) The package contents shall be limited to the contents that were in storage in the package under SNM License No. 2506 (10 CFR Part 72) as of May 2011. Any additional reuse of the packaging after post-shipment unloading of the original content is prohibited.

(d) This certificate applies to only the 29 TN-40 packages already fabricated and in use at the PINGP under SNM License No. 2506 (10 CFR Part 72).

(e) As part of the preparation for transport, the 48 as-installed 1.375-in. diameter SA-320 Grade LA43 closure lid bolts shall be replaced by the SA-540 Grade B23 or B24 Class 1 bolts of the same configuration.

~~(f) As part of the preparation for transport, a 0.75-in. thick by 71.75-in. diameter aluminum spacer shall be installed between the cask lid and the payload.~~

~~(g) As part of the preparation for transport, the metallic seals used in the package and the vent and drain ports shall be replaced and tested to a maximum allowable leak rate of 1.0x10^{-4} ref-cm^3/sec (at a sensitivity of 5.0x10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5.~~

(h) Within 12 months prior to shipment, the user shall perform a leak rate test of the entire containment boundary, with an acceptance criterion of 1.0x10^{-4} ref-cm^3/sec (at a sensitivity of 5.0x10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5. This test is necessary to meet the intent of the containment acceptance tests.

~~(i) A temperature survey shall be performed on each loaded package and the results compared to calculated outer shell temperatures from SAR thermal model analysis in Section 3.4.7 of the application, as supplemented, with appropriate adjustments for decay heat and ambient temperature. The temperature difference between calculated and measured values shall not exceed ±25°F~~.

(j) To comply with 10 CFR 71.85(a), a neutron and a gamma dose rate survey must be performed over the entire surface of the overpack. Total dose rates from these surveys must meet the regulatory limits in 10 CFR 71.47.

(k) For casks that are configured for storage, the operating procedures prescribed in Section 7.4 of the application, as supplemented, must be used to convert the storage configuration to transportation configuration of the package.

7. Transport by air is not authorized.

8. Packagings must be marked with Package Identification Number USA/9313/B(U)F-96.

NRC FORM 618 (8-2000) 10 CFR 71				U.S. NUCLEAR REGULATORY COMMISSION
CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES				
1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE OF PAGES
9313	3	71-9313	USA/9313/B(U)F-96	9 OF 9

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13. The personnel barrier shall be installed at all times while transporting a loaded overpack.

~~14. Revision No. 2 of this certificate may be used until January 31, 2018.~~

15. Expiration date: ~~June 30, 2021.~~

REFERENCES

~~Transnuclear, Inc., application dated: August 7, 2006.~~

~~As supplemented: June 29 and September 11, 2007; August 29, 2008; December 10, 2009; March 6, 15, and 30, April 23, May 7, June 18, July 30, August 26, September 15, and December 22, 2010; May 24, 27, and June 9, 2011; and January 27, 2014.~~

~~AREVA Inc., letter dated May 11, 2016.~~

~~TN Americas LLC letter dated November 18, 2016.~~

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John McKirgan

John McKirgan, Chief
Spent Fuel Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety
and Safeguards

Date: 1/30/17

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

TN Americas LLC)
State of Maryland) SS.
County of Howard)

I, Prakash Narayanan, depose and say that I am Chief Technical Officer of TN Americas LLC, 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 listed below:

- Enclosure 1 - Portions of Responses to RAIs
- Enclosure 2 - Portions of certain chapters of the Safety Analysis Report (SAR) for Certificate of Compliance No. 9313, Revision 17B, Docket 71-9313 (Proprietary Version)
- Enclosure 6 - Certain computer files associated with CoC 9313 (Proprietary)

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by TN Americas LLC 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 certain design details associated with the SAR analyses, SAR drawings, and analysis computer files for the TN-40 System, which are owned and have been held in confidence by TN Americas LLC.
2) The information is of a type customarily held in confidence by TN Americas LLC and not customarily disclosed to the public. TN Americas LLC has a rational basis for determining the types of information customarily held in confidence by it.
3) Public disclosure of the information is likely to cause substantial harm to the competitive position of TN Americas LLC because the information consists of descriptions of the design and analysis of a radioactive material transportation system, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with TN Americas LLC, take marketing or other actions to improve their product's position or impair the position of TN America LLC's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

A. Prakash
Prakash Narayanan
Chief Technical Officer, TN Americas LLC

Subscribed and sworn before me this 2 day of June, 2023.

Fangmin Freese
Notary Public
My Commission Expires 12/15/2025

FANGMIN FREESE
NOTARY PUBLIC
BALTIMORE COUNTY
MARYLAND
My Commission Expires 12-15-2025

Structural:

RAI 2.1:

Justify the use of a 25% flexural rigidity increase when stress and strain of the fuel rod were calculated during the 30-ft end drop under hypothetical accident conditions (HAC) and provide the maximum principal strain of the fuel rod without the use of the 25% flexural rigidity increase.

Subsection 2.10.7.2.2, "TN-40 Fuel End Drop Analysis," of the SAR, Revision 17A (Reference 2.1) indicates that a 25% flexural rigidity increase was used to calculate stress and strain of the fuel rod during the 30-ft end drop under HAC. The 25% flexural rigidity increase was based on a statement in NUREG-2224 (Reference 2.2) for the purpose of calculating lateral displacements. It would be important to consider flexural rigidity of the fuel rod under a side drop, where bending dominates the structural responses. However, flexural rigidity of the fuel rod may not be significant under an end drop, where axial compression and the associated buckling of the fuel rod dominate the structural responses. The staff requests a justification for the use of a 25% flexural rigidity increase to calculate stress and strain of the fuel rod during the 30-ft end drop under HAC and provide the maximum principal strain of the fuel rod without the use of the 25% flexural rigidity increase in the analysis.

This information is needed by the staff to determine compliance with 10 CFR 71.73(c)(1).

Response to RAI 2.1:

Proprietary Information on Pages 2 through 9
Withheld Pursuant to 10 CFR 2.390

References

1. NUREG-2224, “Dry Storage and Transportation of High Burnup Spent Nuclear Fuel,” November 2020.
2. NUREG/CR-1729, BMI-2066, “Evaluating Strength and Ductility of Irradiated Zircaloy Task 5,” November 1980.
3. NUREG/CR-7139, “Assessment of Current Test Methods for Post-LOCA Cladding Behavior,” August 2012.
4. M. Billion, et al., “Embrittlement and DBTT of High-Burnup PWR Fuel Cladding Alloys,” FCRD-UFD-2013-000401, ANL-13/16, September 2013.
5. U.S. NRC CNWRA 2004-08, “A Review Report of High Burnup Spend Nuclear Fuel—Disposal Issues,” Page 6-10.
6. U.S. Nuclear Waste Technical Review Board, “Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel,” December 2010.
7. D. Olander, “Nuclear Fuels Present and Future,” Journal of Nuclear Material, 389, 1-22, 2009.
8. K. Geelhood and C. Beyer, “Corrosion and Hydrogen Pickup Modeling in Zirconium Based Alloys,” In Water Reactor Fuel Performance Meeting, Seoul, Korea, Korean Nuclear Society, 2008.
9. I.E. Porter, et al., “FAST-1.0: A Computer Code for Thermal-Mechanical Nuclear Fuel Analysis Under Steady-State and Transients,” PNNL-29720, March 2020.
10. NUREG/CR-6999, “Technical Basis for a Proposed Expansion of Regulatory Guide 3.54 - Decay Heat Generation in an Independent Spent Fuel Storage Installation,” February 2010.
11. U.S. NRC, Technical Letter Report, “Assessment of Non-Gross Breaches in Spent Fuel Cladding during Drying Operations,” TLR-RES/DE/REB-2021-02, May 2021.
12. U. S. NRC, “NRC Information Notice 2018-01: Noble Fission Gas Releases During Spent Fuel Cask Loading Operations,” February 2018.
13. NUREG-1536 Rev-1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility,” July 2010.

Impact:

SAR Sections 2.10.7.2.2 and 2.10.7.3. been revised as described in the response to RAI 2.3.

RAI 2.2:

Provide a buckling analysis and buckling load of the fuel rod with an initial gap of 1.45 in. between the bottom of the fuel rod and the cask.

A stress analysis was performed to calculate the maximum principal strain of the fuel load during the 30-ft end drop under HAC in Subsection 2.10.7.2.2, “TN-40 Fuel End Drop Analysis,” of the SAR, Revision 17A. However, it appears that there is no buckling analysis performed. Since axial compression and the associated buckling of the fuel rod dominate the structural responses, the staff requests a demonstration of the structural adequacy of the fuel load with respect to buckling during the 30-ft end drop under HAC.

This information is needed by the staff to determine compliance with 10 CFR 71.73(c)(1).

Response to RAI 2.2:

Impact:

SAR Sections 2.10.7.2.2 and 2.10.7.5 have been revised as described in the response.

RAI 2.3:

Provide responses to the following questions related to Subsection 2.10.7.2.2, "TN-40 Fuel End Drop Analysis," of the SAR, Revisions 16 (Reference 2.3) and 17A:

a. Provide (i) the maximum principal stress, and (ii) the calculated maximum stress and strain in the three-dimensional (3-D) cylindrical coordinate system.

b. Provide a factor of safety (FS), which is defined as a ratio of the maximum stress with respect to the yield stress of the fuel rod. If the calculated factor of safety is less than 1.0, explain why the fuel rod is structurally adequate during the 30-ft end drop under HAC.

c. Explain how the initial gap size (i.e., 0.04 in., 0.5 in., 1.0 in., and 1.45 in.) influences and attributes to the different stress and strain in the fuel rod.

The table below presents the results of the stress analysis from Subsection 2.10.7.2.2, "TN-40 Fuel End Drop Analysis," of the SAR, Revisions 16 and 17A.

Case No.	Initial Gap between Pin and Cask (in.)	Internal Pressure (psi)	Maximum Principal Strain (%)
1	0.04	0	0.35
2	0.04	1,400	0.37
3	0.50	1,400	0.47
4	1.00	1,400	0.70
5	1.45	1,400	1.42

The table only presents the results of the maximum principal strain. It is requested that the applicant provide (i) the maximum principal stress, (ii) the calculated maximum stress and strain in the three-dimensional (3-D) cylindrical coordinate system (i.e., longitudinal, rotations and radial directions), and (iii) the calculated FS in a tabular form for all five (5) cases.

Additionally, explain how the initial gap size (i.e., 0.04 in., 0.5 in., 1.0 in., and 1.45 in.) influences and attributes to the different stress and strain in the fuel rod. It is staff's understanding that the methodology and assumptions used for all five cases are similar and the only main difference is the initial gap size. It is not clear how the gap size induces the different stress and strain, especially an elastic behavior to a plastic behavior of the fuel rod during the 30-ft end drop. It is requested that the applicant explain how the initial gap size (i.e., 0.04 in., 0.5 in., 1.0 in., and 1.45 in.) influences and attributes to the different stress and strain in the fuel rod during the 30-ft end drop under HAC.

This information is needed to determine compliance with 10 CFR 71.73(c)(1).

References:

2.1 TN Application for Revision 4 to Certificate of Compliance No. 9313 for the Model No. TN-40 Packaging, Docket No. 71-9313, December 16, 2021.

2.2 NUREG-2224, Dry Storage and Transportation of High Burnup Spent Nuclear Fuel – Final Report, November 2020.

2.3 TN-40 Transportation Packaging Safety Analysis Report, Revision 16, June 2011.

Response to RAI 2.3:

The fuel rod LS-DYNA HAC 30-foot end drop analyses are updated to include only the cladding properties without accounting for the added rigidity of the fuel pellets. Additionally, the weight of the fuel pellets is accounted for using an effective density for the cladding. The cladding material is modeled as elastic-plastic with a strain-hardening modulus that is assumed to be 1% of the elastic modulus. These properties correspond to Zircaloy-4 at 500 °F with an elastic modulus of 11.636×10^6 psi, a strain-hardening modulus of 11.636×10^4 psi, a yield strength of 87,670 psi, and a Poisson's ratio of 0.404.

The results of the updated LS-DYNA analyses are presented in the following table. In addition to the mean IPT maximum principal strain, the maximum principal stress is also reported in the table. The initial gap reported in the table refers to the gap between the bottom of the fuel rod and the cask body spring in the model. An initial velocity of 527.45 in/s is applied to the fuel rod model to simulate a drop from 30 feet. The initial gap between the fuel rod and cask body spring element must close before the cask spring interacts with the fuel rod. As the initial gap increases, the cask and impact-limiter springs compress more before the cask spring interacts with the fuel rod thus causing higher stresses and strains in the fuel rod.

The factor of safety presented in the table is calculated by dividing the yield strength of the cladding (87,670 psi) by the calculated maximum principal stress. The factor of safety for Cases 4 and 5 are below 1.00—indicating plastic deformation.

Table RAI 2.3-1

SAR Figure 2.10.7-20 shows the location at which the maximum principal strain occurs for Case 5 (1.45-inch initial gap). The peak maximum principal strain of 2.09% is very localized. The Westinghouse 14x14 Zircaloy-4 fuel cladding is expected to remain ductile well above this calculated maximum principal strain of 2.09% when considering the burn-up, hoop stresses, hydrides concentration, and temperatures experienced for transport conditions within the TN-40 package (see the Response to RAI 2.1). Therefore, although the fuel cladding will plastically deform at localized regions, it will remain ductile, without rupturing, during the HAC 30-foot end drop.

In addition to the maximum principal stresses and strains reported in the previous table for each case considered, Table RAI 2.3-2, below, provides data regarding additional post-processing performed to calculate the maximum stresses and strains in a local cylindrical coordinate system located at the base of the fuel rod.

Table RAI 2.3-2

Impact:

SAR Sections 2.10.7.2.2 and 2.10.7.3 and Figures 2.10.7-12, 2.10.7-19, and 2.10.7-20 have been revised as described in the response.

RAI 2.4:

Provide a description of the non-destructive evaluations (NDE) performed on the base materials for the containment boundary components and a justification for why the NDE performed is sufficient in lieu of leakage testing.

In Section 1.2.1.1, “Containment Vessel” of the SAR, the applicant states that the inner shell, bottom inner plate, lid outer plate will be examined in accordance with Subsection NB of the ASME Code to the maximum extent practicable. However, the actual NDE that was performed on these components is not described in the application.

NUREG-2216, Section 7.4.2.2 describes the ASME Code requirements for fabrication of components. A fabrication leakage test of the containment boundary seals is specified as a way of meeting the requirement for leak testing in compliance with ANSI N14.5.

However, leak tests are not performed on the base materials of the components that provide a containment function (inner shell, bottom inner plate, lid outer plate, and shell flange) in accordance with ANSI N14.5.

This information is needed to determine compliance with 10 CFR 71.31(c), 71.43(f), and 71.51.

Response to RAI 2.4:

Per section 4.3.1 of the Orano TN procurement specification [1], the non-destructive examinations (NDEs) described below were performed for the containment boundary components.

Plate material (lid outer plate, inner shell, and bottom inner plate) was ultrasonically examined. The requirements and acceptance standards were in accordance with paragraph NB-2530 of [2].

The closure flange was ultrasonically examined in accordance with paragraph NB-2542 of [2], with the acceptance criteria of paragraph NB-2542.2.

Additionally, the lid outer plate and closure flange were also examined by the liquid penetrant or magnetic particle method in accordance with paragraph NB-2546 or NB-2545, with the acceptance criteria of paragraph NB-2546.3 or NB-2545.3.

The welds used in the construction of the containment boundary were all full penetration welds, which were examined by either the liquid penetrant or magnetic particle method, and also radiographed.

Following assembly, the entire containment boundary was hydrostatically tested in accordance with paragraph NB-6200 of [2] at a pressure of 28 psig max for a period of 10 minutes.

Finally, a leakage test was performed on the lid, vent and drain port closure seals and sealing surfaces per the requirements of ANSI N14.5.

However, and as explained in Item 1 of Enclosure 1 of [3], it is not the intent to substitute these NDEs to the containment boundary leak test required by ANSI N14.5; the intent is to remove the specification of the method to be used to perform this leak test of the containment boundary, in order to provide some flexibility to the operator of the packaging regarding the choice of method used to perform this leak test, because the method previously specified in the SAR was deemed not only impractical but also inadequate.

References:

1. Orano TN Specification E-12253 Rev. 14, “Procurement Specification for the TN-40 Spent Fuel Storage Casks.”

2. ASME Boiler & Pressure Vessel Code, 1989 Edition with No Addenda, Section III, Division 1, Subsection NB for Class 1 Components.

3. Orano TN Letter E-59049 dated December 16, 2021, “Application for Revision 4 to Certificate of Compliance No. 9313 for the Model No. TN-40 Packaging, Docket No. 71-9313.”

Impact:

None.