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
7160 Riverwood Drive, Suite 200
Columbia, MD 21046

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

TN-40 Transportation Packaging Safety Analysis Report, Revision 17, dated November 2023

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 approximate 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 packaging is fabricated and assembled in accordance with TN Americas LLC Drawing Nos.:

Drawing No	**Title**
10421-71-1 Rev. 6	TN-40 Transport Packaging Parts List and Notes (1 sheet)
10421-71-2 Rev. 3	TN-40 Transport Packaging Transport Configuration (2 sheets)
10421-71-3 Rev. 3	TN-40 Transport Packaging General Arrangement (1 sheet)
10421-71-4 Rev. 0	TN-40 Transport Packaging Lid Assembly and Details (1 sheet)

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

10421-71-5 Rev. 0	TN-40 Transport Packaging Lid Details (1 sheet)
10421-71-6 Rev. 0	TN-40 Transport Packaging Trunnion, Basket Rail and Neutron Shield Details (1 sheet)
10421-71-7 Rev. 3	TN-40 and TN-40HT Transport Packaging Impact Limiter Spacer Details (1 sheet)
10421-71-8 Rev. 0	TN-40 Transport Packaging Basket Assembly (1 sheet)
10421-71-9 Rev. 0	TN-40 Transport Packaging Basket Details (1 sheet)
10421-71-40 Rev. 2	TN-40 Transport Packaging Impact Limiters General Arrangement (1 sheet)
10421-71-41 Rev. 2	TN-40 and TN-40HT Transport Packaging Impact Limiters Parts List and Notes (1 sheet)
10421-71-42 Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Assembly (1 sheet)
10421-71-43 Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Details (1 sheet)
10421-71-44 Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Parts (1 sheet)

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:

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

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
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

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

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.

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.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9313	4	71-9313	USA/9313/B(U)F-96	6 OF	9

Table 1-1 Fuel Assembly Specifications[1]

Fuel Characteristics	Exxon 14x14 Standard	Exxon 14x14 High Burnup	Exxon 14x14 TOPROD	WE 14x14 Standard	WE 14x14 OFA
	Fuel Assembly Type				
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
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)

1. Pre-irradiated nominal dimensions used in the design analyses and may be verified against as-built records.

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
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) Within 60 days of the first shipment of a shipping campaign involving any TN-40 package, the Certificate holder will notify the NRC of the leakage test method chosen to demonstrate compliance with the regulations in 10 CFR Part 71 related to leakage from the TN-40 package.

7. Transport by air is not authorized.

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

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

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

11. Expiration date: December 31, 2028.

REFERENCES

TN-40 Transportation Packaging Safety Analysis Report, Revision 17, November 2023

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Signed by Diaz-Sanabria, Yoira
on 12/18/23

Yoira K. Diaz-Sanabria, Chief
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety
and Safeguards

Date: December 18, 2023

SAFETY EVALUATION REPORT
Docket No. 71-9313
Model No. TN-40 Package
Certificate of Compliance No. 9313
Revision No. 4

SUMMARY

By letter dated December 16, 2021 (Agencywide Documents Access and Management System [ADAMS] Package Accession No. ML21350A282), TN Americas LLC (TN or the applicant) submitted an amendment request for Revision No. 4 to certificate of compliance (CoC) No. 9313 for the Model No. TN-40 package. The staff accepted the application, with observations, for a detailed technical review on June 15, 2022 (ML22164A321).

On June 6, 2023, the applicant provided its responses to staff's request for additional information (RAI) (ML23157A029, ML23157A030). On November 14, 2023, the applicant provided supplemental information that was requested by staff (ML23318A179) after the review of the RAI responses and the applicant finalized its supplemental responses by letter dated December 5, 2023 (ML23339A062, ML23339A063, ML23339A064). On December 6, 2023, the applicant provided the consolidated TN-40 safety analysis report (SAR), Revision 17 that is now referenced in the CoC No. 9313, Revision 4. (ML23341A013, ML23341A014).

By letter dated April 12, 2021, and in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 71.38, TN had previously submitted a timely renewal request for CoC No. 9313 which had an expiration date of June 30, 2021. The CoC renewal was processed as part of this amendment request.

The amendment request introduced six changes, as detailed below, along with some editorial corrections to the SAR.

1. Acceptance Criteria for Fabrication Leak Testing

The initial application for approval of the TN-40 for use as a transportation package was submitted in August 2006; CoC No. 9313, Revision No. 0, was issued on June 10, 2011. The application specified a fabrication leakage test of the containment boundary seals that was considered to be the requirement for leak testing in the leak testing standard that was in effect (American National Standards Institute [ANSI] N14.5-1987) when the TN-40 packages were fabricated for use as storage casks. A supplement to the original application added details of a test method for the containment boundary: such details were approved by the U.S. Nuclear Regulatory Commission (NRC) staff. However, TN identified some difficulties in the performance of the leakage test specifications and provided a report, as part of this amendment request, discussing alternative leakage test methods.

TN proposed to specify that, prior to the first use of the TN-40 for transportation, there will be a leak test of the package to verify that it does not exceed the allowable combined reference air leakage rate of 1.0 x 10^{-4} ref cm^3/s for the containment boundary. TN said that leak testing of the containment boundary in the ANSI N14.5-1987 standard, to which the TN-40 was designed, licensed, and constructed for use as a storage cask, required testing of the joints and seals only. Because of the thickness of the inner containment shell (1.5 inches), of the multiple weld layers, and the hydrostatic testing, TN believed that it was not necessary to perform a helium leak test of the entire inner shell and lid. Therefore, leak testing was limited to the lid and port cover seals. Leak-testing of the containment boundary base material only became mandatory after the last TN-40 cask was fabricated.

TN said that the removal of a specific method of the leakage test from the SAR will provide the flexibility needed for leakage test experts to develop and qualify a leakage test procedure that is practicable and will ensure that the containment boundary leakage rate meets the criteria for a transportation package. The final procedure may involve multiple leakage tests using different methods to determine the combined leakage rate of the components comprising the containment boundary.

The staff's approval of the applicant's request to remove the specification of leak testing method from their SAR does not provide relief from any commitment to conduct leakage testing of the TN-40 package in accordance with ANSI N14.5. The NRC staff remains committed to applying risk insights to future applications that propose alternative methods from leak testing to demonstrate that the containment performance of the package meets applicable regulatory requirements.

2. Acceptance Criteria for Thermal Testing

The thermal survey requirement was removed from the acceptance tests because the design basis cask thermal model considered the adequacy of the TN-40 cask thermal model and compliance with 10 CFR 71.85(a), in lieu of fabrication tests, to ensure the sufficiency of the thermal performance prior to shipment.

3. Specification for Packaging Marking

10 CFR 71.85(c) requires marking of the packaging with its model number, serial number, gross weight, and package identification number assigned by the NRC. A drawing in the SAR, which specified these markings on a "regulatory plate", has been removed to provide more flexibility for the method used to mark the package with the required information, because the method used for marking this information is in fact not specified in the regulations; therefore, the regulatory plate is not needed in the SAR, as it is not a regulatory requirement.

4. Lid Bolt Replacement Prior to Transportation

The operating procedures have been updated to be consistent with the preparation for transportation from use as a storage cask and allows for lid bolt replacement before the first use of the TN-40 for transportation.

TN performed an engineering study to determine the lid/cask seal status of the TN-40 package when the lid bolts are replaced one at a time. The analysis concluded that a seal is maintained during the replacement of each bolt, one at a time.

5. Removal of the Cavity Spacer

The TN-40 storage cask in use under site specific license special nuclear material (SNM)-2506 does not have a cavity spacer installed. Installation of a spacer, as requested initially, requires returning the storage cask to the spent fuel pool and removing the lid. The removal of the cavity spacer requirement in the CoC will allow the transport of the TN-40 without reopening the TN-40 package.

The staff concluded that, although the validity of the applicant's proposed approach could not be verified, the applicant's calculated maximum principal strain of 2.09% for the 1.45 in. initial gap case was acceptable based on the staff's independent evaluations of the minimum factor of safety of 1.05, thus indicating that cladding rupture can be excluded based on the experimental data for cladding failure. As a result, cladding integrity is reasonably assured.

The staff determined that the removal of the cavity spacer allowing a 1.45 in. gap is acceptable and that there is reasonable assurance of safety for a one-time use only.

6. SAR Drawings

Revisions were made to the drawings for the impact limiter to incorporate lessons learned from the fabrication of other transportation casks and to bring the drawings into conformity with latest TN formatting practices.

The package was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages and the performance standards specific to fissile material packages under normal conditions of transport and hypothetical accident conditions. The analyses performed by the applicant demonstrate that the package provides adequate structural and thermal protection to meet the containment, shielding, and criticality requirements after being subject to the tests for normal conditions of transport and hypothetical accident conditions.

Based on the statements and representations in the application, and the conditions listed in the CoC, the NRC staff (the staff) concludes that the package meets the requirements of 10 CFR Part 71.

EVALUATION

1.0 GENERAL INFORMATION

The packaging is fabricated and assembled in accordance with the following drawings:

Drawing No	Title
10421-71-1 Rev.6	TN-40 Transport Packaging Parts List and Notes (1 sheet)
10421-71-2 Rev.3	TN-40 Transport Packaging Transport Configuration (2 sheets)
10421-71-3 Rev.3	TN-40 Transport Packaging General Arrangement (1 sheet)
10421-71-4 Rev.0	TN-40 Transport Packaging Lid Assembly and Details (1 sheet)

10421-71-5 Rev.0	TN-40 Transport Packaging Lid Details (1 sheet)
10421-71-6 Rev.0	TN-40 Transport Packaging Trunnion, Basket Rail and Neutron Shield Details (1 sheet)
10421-71-7 Rev.3	TN-40 and TN-40HT Transport Packaging Impact Limiter Spacer Details (1 sheet)
10421-71-8 Rev.0	TN-40 Transport Packaging Basket Assembly (1 sheet)
10421-71-9 Rev.0	TN-40 Transport Packaging Basket Details (1 sheet)
10421-71-40 Rev 2	TN-40 Transport Packaging Impact Limiters General Arrangement (1 sheet)
10421-71-41 Rev. 2	TN-40 and TN-40HT Transport Packaging Impact Limiters Parts List and Notes (1 sheet)
10421-71-42 Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Assembly (1 sheet)
10421-71-43 Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Details (1 sheet)
10421-71-44 Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Parts (1 sheet)

Based on review of the statements and representations in the application, the staff concludes that the package design has been adequately described and evaluated, meeting the requirements of 10 CFR Part 71.

2.0 STRUCTURAL AND MATERIALS EVALUATION

The TN-40 is a dual-purpose package designed for use as a storage cask and licensed for transportation (reference 1). The applicant submitted an amendment request along with a SAR, Rev. 17A, (reference 2), including six proposed changes as summarized above; only the proposed removal of the cavity spacer is relevant to the structural performance of the TN-40 package and required structural evaluations.

This safety evaluation report (SER) evaluates the structural analyses of the TN-40 package performed by the applicant to verify that the structural performance of the package meets the regulatory requirements of 10 CFR Part 71 under hypothetical accident conditions (HAC). It is noted that the applicant initially submitted the SAR, Rev. 17A (reference 2), but the SAR, Rev. 17A was updated to the SAR, Rev. 17B during the request for additional information (RAI) process (reference 3). Therefore, the staff's review on the structural performance of the TN-40 package under HAC is mainly based on the applicant's structural analyses and evaluations provided in the SAR, Rev. 17B (reference 3).

2.1 Description of the TN-40 Package

The TN-40 package is a dual-purpose cask to be used as a Type B(U)-F spent fuel transportation packaging and storage cask. The TN-40 package is licensed for storage at the Prairie Island Nuclear Generating Plant (reference 4). The TN-40 package can transport up to 40 pressurized water reactor (PWR) undamaged fuel assemblies with or without fuel inserts.

The TN-40 package consists of: (i) a basket assembly to locate and support the fuel assemblies, transfer heat, and provide neutron absorption, (ii) a containment vessel with closure lid and metallic O-ring seals to provide containment and maintain an inert gas atmosphere, (iii) a thick-walled, forged steel shield shell, bottom shield, and lid shield plate to provide gamma shielding, (iv) a radial neutron shield enclosed within a steel outer shell, (v) a set of impact limiters consisting of balsa and redwood encased in stainless steel shells with an aluminum spacer to provide a smooth contact surface between the top impact limiter and cask lid, secured with tie-rods and attachment bolts, and (vi) upper and lower trunnions.

The overall dimensions of the TN-40 package are 260.87 in. long and 144 in. in diameter with the impact limiters installed. The cask body is 183.75 in. long (with the lid installed) and 91 in. in diameter. The lid is 82.75 in. in diameter. The cask outside diameter including the radial neutron shield is 101.0 in. The cask cavity is 163 in. long and 72.0 in. in diameter. The nominal gross weight of the package is 271.5 kips with a 52.0 kips maximum payload weight.

2.2 Evaluation of the Removal of the Cavity Spacer

The TN-40 package initially required a cavity spacer to fill in a gap between the bottom of the fuel rods and the cask in its transport configuration. However, the applicant stated that the TN-40 packages, used as storage under site specific license SNM-2506 (reference 4) do not have a cavity spacer installed; thereby, a cavity spacer would need to be installed in the TN-40 packages prior to transportation.

Installation of a cavity spacer would require returning the TN-40 package to the spent fuel pool in order to remove the cask lid and install the cavity spacer. Since installation of a cavity spacer would require time and effort, the applicant developed a plan to use the TN-40 package for transportation without a cavity spacer and submitted this amendment request to, in part, allow removal of the cavity spacer requirement in order to use the TN-40 package for transportation without reopening the cask to install the cavity spacer.

The applicant previously performed structural analyses to demonstrate the structural adequacy of the PWR high burnup fuel rod (Westinghouse 14x14 STD) in the TN-40 package with varying cavity spacer gaps (i.e., 0.04 in., 0.50 in., and 1.00 in.) under an HAC 30-feet (ft) end drop event, which the staff previously reviewed and accepted (reference 5). Appendix 2.10.7, “Structural Evaluation of the Fuel Rod Cladding Under Accident Impact,” of the SAR, Rev. 16 provides the structural analyses and evaluations to demonstrate the structural integrity of fuel rod cladding for both the HAC side and end drops.

Specifically, section 2.10.7.2, “End Drop Analysis,” in appendix 2.10.7 of the SAR, Rev. 16, contains the structural analyses and evaluations with varying initial spacer gaps between the fuel rod bottom and cask (i.e., 0.04 in., 0.50 in., and 1.00 in.) using the LS-DYNA finite element (FE) model to simulate the characteristics of the fuel rod in the TN40 cask under the 30-ft end drop.

2.2.1 Structural Analysis for Single Fuel Rod

In the SAR, Rev. 17B, the applicant considered one additional gap of 1.45 in. between the fuel rod bottom and the cask to demonstrate that the fuel cladding will not buckle or rupture under HAC. The applicant used the previously developed LS-DYNA FE model and performed an additional structural analysis with a new spacer gap of 1.45 in. under the HAC 30-ft end drop.

The applicant submitted the evaluations of the structural analysis with the LS-DYNA input files, model files, and results files for staff's review. The information discussed in the following paragraphs for the FE parameters is based on the review of the LS-DYNA input files (k-files), the LS-DYNA graphical output results files (d3plot), and LS-DYNA American Standard Code for Information Interchange output results files (MATSUM, GLSTAT, etc.) submitted by the applicant in the application.

Geometry: The LS-DYNA FE model geometry for the end drop consists of a single fuel rod and a highly simplified representation for the basket internals, cask, and impact limiter. The model assumes that all the fuel rods in an assembly would deform in the same manner as a single rod. The fuel rod is therefore supported by springs at the spacer grid locations, allowed to move axially, and allowed to deform laterally under end impact. Lateral deformation of the fuel rod is limited by a contact surface that represents the maximum deflection of the fuel assembly prior to contacting the basket wall.

The TN-40 model considers a Westinghouse PWR 14x14 STD fuel assembly. For conservatism in end drop loading, the model includes the weight of the fuel by adjusting the cladding density but does not directly include the complete stiffness of the fuel pellets. In the validation case performed by the applicant, springs were used to represent the compression of the fuel pellets to prevent possible collapse of the fuel cladding during bending. In the initial TN-40 evaluation in the SAR, Rev. 17A (reference 2), these fuel pellet springs were not used, and the fuel cladding stiffness was increased by a factor of 1.25 (i.e., 25 percent (%) increase in flexural rigidity) per the guidance of NUREG-2224 to account for fuel pellet effects (reference 6). In response to the staff's RAI on the 25% increase in flexural rigidity, the applicant performed updated structural analyses without the factor of 1.25 on the cladding stiffness for the different initial axial gap sizes. The model also used a minimum cladding thickness and reduced outer diameter due to oxidation to provide a minimum cross section (i.e., minimum flexible rigidity) of the fuel rod.

The cask is modeled as a point mass which interacts with the fuel rod and impact surface through springs. The spring between the cask and impact surface represents the impact limiter. The spring between the cask and fuel rod represents the stiff intermediate components such as the cask lid, canister, etc., and includes the ability to allow a prescribed gap where initial free travel of the fuel rod can occur prior to compressing the spring. The scaled cask mass is based on a fully loaded package weight minus the weight of the fuel assemblies.

Material Properties: The 30-foot end drop analysis using the LS-DYNA FE model is 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 degrees Fahrenheit (°F) with an elastic modulus of $11,636 \times 10^6$ pounds per square inch (psi), a strain-hardening modulus of $11,636 \times 10^4$ psi, a yield strength of 87,670 psi, and a Poisson's ratio of 0.404.

Elements: The LS-DYNA FE model for the fuel rod evaluation uses shell, mass, and discrete element types to define the fuel rod geometry using a total of 35,589 elements and 40,225 nodes. Fully integrated shell elements (Type 16) are used for the fuel cladding, fuel end caps, and basket compartment walls. The shells include 5 integration points through the thickness to adequately capture plastic deformations of the fuel rod. Discrete elements are used for the translational springs representing the spacer grids, rod-to-cask spring, and cask-to-ground springs with linear or nonlinear force-deflection curves. A single nodal mass element is used to represent the scaled cask mass.

Contacts: The fuel rod model uses surface-to-surface sliding contacts between the fuel rod cladding and the basket wall. The static and dynamic friction coefficients are zero.

Boundary Conditions: Symmetry boundary conditions for the half-symmetric fuel rod geometry are implemented correctly in the model. The displacements of the rigid shells representing the fuel basket compartment wall and the node representing the impact surface are fully constrained. The lateral displacements of the spacer grid spring ends are constrained, and the axial displacements of the spacer grid spring ends are free, such that only lateral forces are generated in support of the fuel rod cladding as it deforms by bending.

Initial Conditions: The initial velocity is applied at the beginning of the analysis to the fuel rod and cask node. The HAC velocity for 30-ft drop is 527.45 in/s. The fuel cladding model uses material properties for a maximum temperature of 500°F.

The fuel rod is allowed to freely travel a predefined initial gap of 0.04 in., 0.50 in., 1.00 in., or 1.45 in. between the fuel rod and cask prior to the initialization of the force-displacement curve.

Loads: No gravitational acceleration is included in the fuel rod model, which is acceptable since the impact loads are significantly greater than the gravitational load.

2.2.2 Results of the Structural Analysis for Single Fuel Rod

The applicant performed structural analyses using the LS-DYNA FE model with the material properties, boundary conditions and loadings described above, and provided the results of the LS-DYNA FE structural analyses for the different initial gap conditions.

Table 1 below presents the calculated maximum principal strain of the fuel rod under the 30-ft end drop for each gap condition from the SAR, Rev. 17B.

Table 1. Results of the LS-DYNA FE Structural Analyses

Case No.	Initial Gap (in.)	Internal Pressure (psi)	Max. Principal Stress (psi)	Mean IPT Max. Principal Strain (%)	Factor of Safety
1	0.04	0	47,910	0.35	1.83
2	0.04	1,400	49,368	0.34	1.78
3	0.50	1,400	67,182	0.51	1.30
4	1.00	1,400	98,407	0.71	0.89
5	1.45	1,400	98,921	2.09	0.89

The factor of safety presented in table 1 above 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 strain or permanent deformation.

Table 2 below lists the maximum effective plastic strains, where the effective plastic strain is defined as a monotonically increasing scalar value which is calculated incrementally as a function of the plastic component of the rate of deformation tensor indicating that the cladding is actively yielding.

Table 2. Maximum Effective Plastic Strain

Case No.	Initial Gap (in.)	Internal Pressure (psi)	Maximum Effective Plastic Strain (%)
1	0.04	0	0.00
2	0.04	1,400	0.00
3	0.50	1,400	0.00
4	1.00	1,400	0.39
5	1.45	1,400	3.02

Table 2 indicates that plastic strain or permanent deformation of the fuel begins with a gap of less than 1.00 in. (a gap of about 0.95 in.) and increases as gap size increases larger than 1.00 in. The maximum principal strain increases by a factor of 2.94 with an increased gap from 1.00 in. to 1.45 in. (0.45 in. gap increment), while the maximum principal strain increases by a factor of 1.39 with an increased gap from 0.50 in. to 1.00 in. (0.50 in. gap increment) indicating that a rate of increasing plastic strain accelerates after the yield point.

2.2.3 Evaluation of the Structural Analysis for Single Fuel Rod

<u>Strain Rupture Analysis</u>: The modeling approach taken by the applicant in the SAR, Rev. 17B to predict cladding strains is acceptable. This approach was previously reviewed and accepted by the staff (reference 5). The developed LS-DYNA FE model for the single rod adequately represents the deflection and buckling of the fuel rod under HAC end impacts. The results of the LS-DYNA FE analyses show that cladding plastic strain increases with an increase in gap between the fuel rod and the cask under HAC starting at a gap of about 0.95 in.

The applicant presented the calculated maximum principal strains in the fuel rod cladding under the HAC 30-ft end drops in table 1 above with the largest value of 2.09% observed with the 1.45 in. gap condition. The applicant stated that the result of the maximum principal strain of 2.09% is acceptable when considering the burn-up, hoop stresses, hydrides concentration, and temperatures experienced for transport conditions within the TN-40 package as discussed in NUREG-2224 (reference 6), where a fuel cladding strain limit (offset strain) of about 3% from a summary of the testing results was proposed. As a result, the applicant concluded that the Westinghouse 14x14 Zircaloy-4 fuel cladding will remain ductile, without rupturing, during the HAC 30-ft end drop, because the calculated maximum principal strain of 2.09% is less than the offset strain of 3.0%.

The staff reviewed the applicant's responses and technical justifications to the staff's RAI (reference 3) based on the information available in reference 6. The staff agrees that lower strains would be calculated when the additional flexural rigidity due to fuel pellets is accounted for in the LS-DYNA FE structural analyses.

However, the staff does not agree with a comparison between the calculated maximum principal strain of 2.09% for the 1.45 in. initial gap case and the offset strain of 3% from NUREG-2224. The offset strain in NUREG-2224 is determined from ring compression testing (RCT) and is defined as RCT offset displacement at the 12 o'clock position relative to static support at 6 o’clock divided by the outer diameter of the cladding material. The objective of the RCT analysis is to use multiple tests at different temperatures to determine the ductility transition temperature (DTT) for a cladding material. The offset strain is a metric that can be easily measured in the RCT test to determine the material’s DTT, but it is not the same as the actual material strain in the test specimen that can be compared to strain values predicted in the LS-DYNA FE structural analysis. These two quantities cannot be directly compared to each other to demonstrate adequacy of the cladding ductility. As a result, the applicant responded that it would submit technical justifications of why the calculated maximum principal strain of 2.09% is acceptable and the fuel rod remains ductile.

Additional calculations were then submitted to develop and justify a new displacement criterion to demonstrate that lateral deflections under end drop are acceptable through comparison to lateral deflections from the 4-point bend tests of actual fuel rods performed as part of the sibling pin testing campaign (reference 7). The approach proposed to obtain a deflection criterion by scaling the deflection from the bend test by two different factors that accounted for the (i) difference in unsupported fuel rod length between the test and transportation configuration, (ii) difference between the bend test loading configuration and the axial loading condition during an end drop, and (iii) difference between the bend test specimen, which consisted of fuel and cladding, and the end drop model, which consisted of cladding only. The staff reviewed the proposed approach and did not agree that it was an acceptable method for evaluation of cladding integrity.

The staff’s technical bases for not agreeing the applicant’s approach are:

(i) the approach did not discuss strains used in any of the models to demonstrate that the approach is valid,

(ii) (ii) the approach assumes linear beam bending response though plastic deformation is occurring,

(iii) the approach scales the results from only a single endpoint condition though the model does not fully represent the behavior of the test,

(iv) models such as distributed lateral loading were used to support the scaling relationship but are otherwise not relevant to the loading under end drop,

(v) the largest errors in the scaling factor were obtained for an end loading condition, which is closest to the condition under end drop,

(vi) the argument for different loading pathways to justify the scaling factors was not correct, and

(vii) the end loading models are using deflections from a post-buckled state, even though the rod did not buckle in the end drop simulation.

The staff concluded that the applicant’s approach had many uncertainties and did not provide a convincing argument that cladding integrity could be maintained.

However, the staff performed independent evaluations to show that cladding integrity can be reasonably assured by using data on high burnup Zircaloy cladding from the same test campaign cited by the applicant (reference 7). The 4-point bend test specimens demonstrated maximum strains of 1.6%-2.9% with an average value of 2.2% at 200°C (392°F) conditions (reference 7).

Additionally, the tension test of cladding-only specimens demonstrated maximum plastic strains of 2.0%-3.1% with an average of 2.5% at 200 degrees Celsius (°C) conditions (reference 8). The corresponding total strains for these axial tests ranged from 2.9% to 4.0% with an average of 3.4% at 200°C conditions.

The applicant's end drop results showed a maximum total strain in the cladding of 2.09% for the cladding-only LS-DYNA FE model with the initial gap of 1.45 in. and a value of 1.42% when a stiffness factor of 1.25 was applied to account for pellet interaction (per NUREG-2224). The TN-40 content has a burnup of 45 GWd/MTU, so (i) the test data for high burnups of 51-64 GWd/MTU are expected to be conservative, and (ii) the gap between the pellet and cladding is expected to be closed around 20 GWd/MTU (reference 9), which may cause some pellet interactions with the cladding.

Assuming pellet support increases the flexural rigidity, a maximum cladding strain of 1.36%-1.42% is estimated under the HAC 30-ft end drop. As result, a factor of safety of 1.13 would be obtained if the minimum bend test data is used or a factor of safety of 1.41 would be obtained if the minimum plastic strain from the axial test data is used.

Furthermore, if the effect of pellet support on the flexural rigidity is neglected, a factor of safety of 1.05 is obtained using the average bend test data or a factor of safety of 1.24 using the average plastic strain from the axial test data.

Therefore, the staff concludes that, although the validity of the applicant's proposed approach could not be verified and is not accepted, the applicant's calculated maximum principal strain of 2.09% for the 1.45 in. initial gap case is acceptable based on the staff's independent evaluation of the minimum factor of safety of 1.05 indicating that cladding rupture can be excluded based on the experimental data for cladding failure and, as a result, cladding integrity is reasonably assured.

<u>Buckling Analysis</u>: The applicant performed a buckling analysis for the fuel rod with an initial gap of 1.45 in. between the bottom of the fuel rod and the cask in the SAR, Rev. 17A. The applicant's approach was to have a sinusoidally-shaped initial deformation, with a bowing of 0.015 in., to account for inelastic bowing, in the LS-DYNA FE analysis, which was reviewed and accepted by the staff in the SAR, Rev. 16 (reference 5). Additionally, the applicant did perform a buckling analysis of the fuel rod using the guidance of UCID-21246 (reference 10) and ISG-12 and provided the results of the analysis in the updated subsection 2.10.7.2.2 of the SAR, Rev. 17B in response to RAI 2-2 (reference 3).

The buckling calculation accounts for the flexural rigidity of the fuel pellets and determines the critical buckling load of the fuel rod as 552 pounds (lb). This critical buckling load is divided by the weight of the fuel rod (6.77 lb.) to determine the required gravitational load of 81.5 grams (g) needed to buckle the fuel rod. Using the peak deceleration of 62g from the 30-ft end drop, at -20°F, and a dynamic load factor of 1.15 from NUREG/CR-3966 (reference 11), the fuel rods will experience a gravitational load of 71.3g.

The staff reviewed the buckling analysis and determined that the theoretical buckling approach by the applicant is acceptable. The fuel rod will not buckle since impact acceleration loads experienced by the fuel rod (71.3g) are less than that required to buckle the fuel rod (81.5g).

Conclusion: Based on the evaluations of the applicant's structural analyses and evaluations as discussed above, the staff determines that the approach taken by the applicant to predict the structural performance of the fuel rod under the HAC 30-ft end drop is acceptable with the fuel rod-to-cask gap size of 1.45 in. The approach using the LS-DYNA FE model is in line with what was done in the previous application for the SAR, Rev. 16 (reference 5), which was reviewed and accepted by the staff.

However, the staff does not agree with the applicant's demonstration that the calculated maximum principal strain of 2.09% for the fuel rod is shown to be acceptable, and that it remains ductile based on a comparison of the 2.09% maximum principal strain with the 3.0% offset strain from NUREG-2224.

It appears that the applicant incorrectly intermingles the use of offset strain from the RCT test and the maximum principal strain from the LS-DYNA FE analysis. The offset strain is not the same as the material strain in a test specimen and cannot be directly compared to the strains predicted by the LS-DYNA FE analysis to demonstrate a stress failure of the fuel rod is not a concern under HAC.

The applicant provided additional clarification and justification as to why the predicted strains from the LS-DYNA FE analyses are acceptable using a deflection-based criterion based on experimental test data from the literature, but this approach was incomplete and not accepted by the staff. Additional evaluation of the allowable strain limits based on the sibling pin cladding test campaign (reference 7) and application to the TN-40 content by the staff demonstrated that the predicted model strain of 2.09% is acceptable when accounting for pellet stiffness effects.

As a result, the staff determines that the removal of the cavity spacer allowing a 1.45 in. gap is acceptable and there is reasonable assurance of safety for a one-time use only.

2.3 Evaluation Finding

The staff reviewed the applicant's proposed change of the removal of the 1.45 in. cavity spacer with the structural analyses and evaluations performed by the applicant. Additionally, the staff also performed an independent evaluation of the experimental data from the sibling pin cladding test campaign, and determines that the proposed change, PC 5, to the TN-40 package is acceptable.

The structural performance of the TN-40 package with the change is in compliance with the requirements of 10 CFR 71.73. The applicant's evaluations of the structural analyses combined with the staff's independent evaluation show that the localized high plastic strain in the cladding has adequate safety margin when considering flexural rigidity from the pellets and they provide reasonable assurance that the TN-40 package will allow safe transportation of spent nuclear fuel for a one-time shipment.

This finding is reached on the basis of a review that considered the applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

2.4 References

1. U.S. Nuclear Regulatory Commission, Certificate of Compliance No. 9313 for the Model No. TN-40, Revision 3.

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

3. “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,” Orano TN Letter E-62022, June 6, 2023.

4. NSPM “Prairie Island Independent Spent Fuel Storage Installation Safety Analysis Report,” Revision 14, Materials License No. SNM-2506, ADAMS No. ML113040131, Minneapolis, MN, Northern States Power Company–Minnesota, October 2011.

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

6. U.S. Nuclear Regulatory Commission, “Dry Storage and Transportation of High Burnup Spent Nuclear Fuel,” NUREG-2224, November 2020.

7. “Sister Rod Destructive Examinations (FY21),” Montgomery, Rose, & Bevard, Bruce B., M2SF-22OR010201042 (ORNL/SPR-2021/2303), March 31, 2022.

8. “Sibling Pin Testing Results (PNNL FY2022),” Shimskey, R. W., Allred, J. R., Asmussen, S. E., Cooley, S. K., Daniel, R. C., Edwards, M. K., Geeting, J., Goulet, A. P., Holbrook, C. H., MacFarlan, P. J., Nickerson, E. K., Richmond, L. I., Roosendaal, T. J., Westesen, A. M., Westman, B. E., & Hanson, B. D., M2SF-23PN010201041 (PNNL-33781), December 19, 2022.

9. Geelhood KJ, “Fuel Performance Considerations and Data Needs for Burnup above 62 GWd/MTU,” PNNL-29368, Pacific Northwest National Laboratory, Richland, WA, 2019.

10. Chun, R., Witte, M., Schwartz, M., "Dynamic Impact Effects on Spent Fuel Assemblies," UCID-21246, Lawrence Livermore National Laboratory, Livermore, California, 1987.

11. Nelson, T.A., Chun, R.C., “Methods for Impact Analysis of Shipping Containers,” UCID-20639 (NUREG/CR-3966), Lawrence Livermore National Laboratory, Livermore, California, 1987.

2.5 MATERIALS EVALUATION

The staff reviewed and evaluated the information provided by the applicant requested in Revision 17A. The specific changes evaluated in this section include:

1. Acceptance Criteria for Fabrication Leak Testing

The SAR chapter 7, Operating Procedures, and chapter 8, Acceptance Tests and Maintenance Program have been revised to remove the specification of the method used for leakage testing. Chapter 7, Operating Procedures, has been revised and includes information regarding preparation for transport in lieu of its former use as a storage cask. The applicant proposes to only specify that, prior to the first use of the TN-40 for transportation, there will be a leakage test of the packaging to ensure that the containment of the contents does not exceed the allowable combined reference air leakage rate of 1.0×10^{-4} ref cm^3/s for the containment boundary.

2. Specification for Packaging Marking

   10 CFR 71.85(c) requires marking of the packaging with its model number, serial number, gross weight, and package identification number assigned by the NRC. A drawing in the current SAR specifies these markings on a “regulatory plate” that is attached to the package. That drawing has been removed.

3. Lid Bolt Replacement Prior to Transportation

   Chapter 7, Operating Procedures, has been revised and includes information regarding preparation for transport in lieu of its former use as a storage cask, and describes the process for lid bolt inspection and replacement before transportation.

4. Remove the Cavity Spacer

   The TN-40 storage cask in use under site specific license SNM-2506 does not have a cavity spacer installed. Installation of a spacer requires returning the storage cask to the spent fuel pool and removing the lid. The removal of the cavity spacer requirement will allow use of the TN-40 for transportation without reopening the TN-40 to install the cavity spacer.

   Chapter 7, Operating Procedures, has been revised and includes information regarding preparation for transport in lieu of its former use as a storage cask. This new version of the operating procedure recognizes that the TN-40 has already been loaded and will be used once for transportation.

5. SAR Drawings

   Change to update the following drawings for the impact limiter:

   - 10421-71-41 R1 has been updated by incorporating lessons learned from the fabrication of other transportation casks and by bringing the drawing into conformity with latest TN formatting practices.
   - 10421-71-42 R0 has been updated to make fabrication easier.
   - 10421-71-44 R0 has been updated by bringing the drawing into conformity with latest TN formatting practices.

The staff reviewed the changes to the SAR chapters 1, 2, 3, 6, 7, and 8, including figures and drawings. The staff review was conducted using the guidance in chapter 7 of NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material,” to determine whether there was adequate materials performance of the transportation package under normal conditions of transport and hypothetical accident conditions necessary to meet the requirements of 10 CFR Part 71.

The areas of review covered in this SER section are described in NUREG-2216, section 7.2, and included drawings, codes and standards, weld design and inspection, mechanical properties, thermal properties of materials, radiation shielding and criticality control materials, corrosion resistance, protecting coatings, content reactions, radiation effects, package contents, bolting material, and seals.

In addition to the guidance in NUREG-2216, the staff evaluated the engineered drawings and the description of the structures, systems, and components included in the application using the information provided in NUREG/CR-5502, “Engineering Drawings for 10 CFR Part 71 Package Approval,” and NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety.”

Acceptance Criteria for Fabrication Leak Testing

The applicant proposed revising SAR section 8.1.3, which states that leakage tests are performed to verify containment contents by the packaging prior to first use for transportation.

The applicant stated that the original application specified a fabrication leakage test of the containment boundary seals that was considered to be the requirement for leak testing in the leak testing standard that was in effect (ANSI N14.5-1987) when the TN-40 casks were fabricated for use as spent nuclear fuel storage casks. A supplement to the original application added details of a test method for the containment boundary of the package that were approved by the staff.

The applicant stated that the leakage test method that was previously approved by the staff has been reviewed by a certified leakage testing expert, and some difficulties with performing the leakage test specification as previously approved were identified. In addition to the difficulties identified, the certified leakage testing expert has provided alternative methods for leak testing the containment boundary.

Based on the alternative methods discussed above, the applicant proposed removing the specification of the method used for leakage testing in chapters 7 and 8. This resulted in the applicant also proposing changes to SAR chapter 7, section 8.1.3, and section 8.3.

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 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 the ASME Boiler and Pressure Vesel Code section III, subsection NB to the maximum extent practicable. The applicant provided the Orano TN procurement specification which described the nondestructive examination (NDE) performed on the base materials of the components that provide a containment function (inner shell, bottom inner plate, lid outer plate, and shell flange).

The staff reviewed the procurement specification and verified that the inner shell, bottom plate, and lid outer plate were ultrasonically examined in accordance with subsection NB-2530 of the ASME Code and the shell flange was ultrasonically examined in accordance with subsection NB-2542.2 acceptance criteria. The staff verified that the lid outer plate and shell flange were also examined by liquid penetrant or magnetic particle method in accordance with subsection NB-2546 or NB-2545.

The staff also verified that the welds used in the construction of the containment boundary were all full penetration welds and examined by either liquid penetrant or magnetic particle method and also volumetrically examined using radiographic testing per the requirements of NB-5200. Based on the NDE performed, the staff finds that there are no credible leakage paths through the welds or base materials.

The staff finds that the NDE performed on the base materials for the containment boundary components in accordance with ASME Code subsection NB is in compliance with 10 CFR 71.31(c), 71.43(f), and 71.51. However, the staff's approval of the applicant's request to remove the specification of leak testing method from the SAR does not provide relief from any commitments currently made by the applicant to conduct leakage testing of the TN-40 package in accordance with ANSI N14.5.

Specification for Packaging Marking

The applicant proposed removing the requirement of providing packaging marking on a "regulatory plate" on the basis that 10 CFR 71.85(c) does not require a specific method used in the regulation. This resulted in the applicant proposing changes to SAR sections 1.4.1 and 2.1.1 and Drawing 10421-71-10.

The staff reviewed Drawing 10421-71-10, which stated that the regulatory plate was attached to the cask in SAR Drawing 10421-71-2 Sheet 1 of 2, which illustrated where the regulatory plate was previously, and SAR section 1.4.1, where reference to Drawing 10421-71-10 was removed, and SAR section 2.1.1 which previously had a pointer to the regulatory plate drawing. The staff noted that although Drawing 10421-71-10 was removed, the applicant will continue to meet 10 CFR 71.85(c) because the regulatory plate and pad are specified in the Parts List and Notes in Drawing 10421071-1 and shown in the Transport Packaging Assembly in Drawing 10421-71-3, General Arrangement, will conspicuously and durably mark the packaging with its model number, serial number, gross weight, and a package identification number assigned by the NRC. The staff therefore finds the change to packaging marking acceptable and in accordance with 10 CFR 71.85(c).

Lid Bolt Replacement Prior to Transportation

The applicant proposed revising chapter 7, Operating Procedures, and includes information regarding preparation for transport in lieu of its prior use as a storage cask and allows for lid bolt replacement before the first use of TN-40 for transportation. The basis provided by the applicant was an engineering study performed which determined the lid/cask seal status when the lid bolts are replaced one at a time. The maximum allowable decompression of the seal is 0.0039 in., and the maximum decompression of one bolt being removed is 0.0013 in. The applicant stated that the analysis concludes that, per the decompression results above, the seal is maintained when replacing one bolt at a time.

This proposed change resulted in SAR chapter 7 updates to the entire section.

The staff reviewed SAR section 7.1.3, "Preparation for Transport," which states, "Verify that the lid bolts are the material specified on Drawing 10421-71-1. If lid bolts require replacement, then the bolts shall be replaced one at a time." Additionally, per step 7 of SAR section 7.1.3, "Preparation for Transport," if any bolts are replaced, the applicant will perform a pre-shipment leak test of the lid, vent port cover, and drain port cover seals in accordance with ANSI N14.5, with a combined standard leak rate that shall be less than 1.0 x 10^{-4} ref cm^3/s. The staff finds the leak test demonstrates that the seal is maintained when replacing lid bolts. The staff reviewed Drawing 10421-71-1 and found that it properly identified the lid bolt material in accordance with 10 CFR 71.33(a)(5).

Removal of the Cavity Spacer

The applicant proposed removal of the cavity spacer requirement so as to allow the TN-40 to be used for transportation without reopening it to install the cavity spacer. The applicant stated that the TN-40 storage cask in use under SNM-2506 does not currently have a cavity spacer installed. The applicant also stated that the new version of the operating procedure recognizes that the TN-40 has already been previously loaded and will be used for transportation. The preparation for transportation is done after the TN-40 has been previously loaded and the TN-40 has been in use for storage of spent nuclear fuel. The applicant's basis for acceptability of the change is that removal of the cavity spacer requirement will result in an initial gap of 1.45 in. resulting in plastic deformation in the fuel assembly due to the accident condition free drop, and that the fuel rod has a stable lateral deflection below 0.3 in., consistent with the previously evaluated displacement following a plastic deformation.

This resulted in changes to SAR sections 1.4.1, 2.10.7.2, 2.10.7.2.2, 2.10.7.2.3, 2.10.7.3, 2.10.7.5, 2.10.11.3, 6.4.2, and chapter 7 (all sections), SAR figure 2.10.7-12, figure 2.10.7-19, and figure 2.10.7-20, SAR figure 2.10.7-21 to be added, and SAR Drawings 10421-71-7 and 10421-71-43.

The staff reviewed the changes in SAR sections 2.10.7.2, 2.10.7.2.2, 2.10.7.3 regarding determination of the structural adequacy of the fuel cladding using the guidance in NUREG-2224, "Dry Storage and Transportation of High Burnup Spent Nuclear Fuel." The applicant stated that, "Per NUREG-2224, a lower bound 2% offset strain limit is used to determine the structural adequacy of the fuel cladding." When accounting for the stiffness of the fuel pellets, the applicant states that this provides 1.96 times the flexural rigidity of the cladding alone. In accounting for uncertainties and standard deviations, a factor of 1.25 is suggested for cladding only analyses, per NUREG-2224 section 2.3.4. The applicant uses this factor of 1.25 in the evaluation of the 1.45 in gap analysis. The staff reviewed the approach proposed by the applicant and, based on the guidance in NUREG-2224, the staff determined that the use of a factor of 1.25 for the Zircaloy-4 cladding is acceptable for determination of the structural adequacy of the fuel cladding in meeting the requirements of 10 CFR 71.35(a) and 10 CFR 71.55(d)(2) and therefore the removal of the cavity spacer requirement is acceptable from a materials point of view.

As explained above, the staff concluded that, from a structural perspective, the applicant's calculated maximum principal strain of 2.09% for the 1.45 in. initial gap case is acceptable based on the staff's independent evaluation of the minimum factor of safety of 1.05 indicating that cladding rupture can be excluded based on the experimental data for cladding failure and, as a result, cladding integrity is reasonably assured.

The staff reviewed the SAR Drawings 10421-71-7, "Impact Limiter Spacer Details," and 10421-71-43, "Impact Limiters Details" and found they were in accordance with 10 CFR 71.33 and are therefore acceptable.

SAR Drawings

The applicant proposed changes to three drawings, SAR Drawings 10421-71-41, 10421-71-42, and 10421-71-44 for updates to the impact limiter, the basis being that the changes are to make fabrication easier and to incorporate lessons learned from the fabrication of other transportation casks. This resulted in changes to SAR Drawings 10421-71-41, 10421-71-42, and 10421-71-44.

The staff reviewed the changes to SAR Drawings 10421-71-41, 10421-71-42, and 10421-71-44 and found that welds and NDE symbols were identified per AWS 2.4 and changes affecting the impact limiter were in accordance with 10 CFR 71.33 and are therefore acceptable.

Conclusion

The NRC staff concludes, based on the review of the statements, and representations in the application, that the materials used in the package design have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The applicant described in section 1.1 of the application that the TN-40 is to be licensed for one-time transportation. That one-time use could include any sequence of shipments, specifically for shipment of the spent fuel the TN-40 contained during storage. The TN-40 is shipped as exclusive use with a personnel barrier.

The maximum temperatures are unchanged from the staff's previous review. The maximum normal operating pressure of the TN-40 is 15.7 psig and a cask cavity pressure of 100 psig is conservatively used in the structural analysis, which is also unchanged from the staff's previous review.

The TN-40 has not had a thermal acceptance test / thermal survey over the outer surface that was described in the TN-40 SAR rev. 3, section 3.4.7, item #2 on page 3-14A, ADAMSML100210335 (proprietary); however, the TN-32B, Docket No. 72-1021, has been validated by TN and the U.S. Department of Energy/Electronic Power Research Institute (DOE/EPRI) as described in the references 18 and 19, respectively, that are listed on page 3-23 of the TN-40 SAR.

The applicant showed on page 3-18 of the SAR that the TN-32B and the TN-40 are similar in: design and key dimensions, the use of heat transfer in the radial direction, and the use of conservative radial gaps in the thermal model that bound the fabrication gaps. The TN-40 thermal evaluation also demonstrates that there is significant margin for the fuel cladding and the seals.

The applicant concluded that the thermal performance of the as fabricated TN-40 package will continue to be bounded by the existing thermal evaluation in chapter 3 of the TN-40 SAR; the staff finds this to be acceptable based on the staff's review of the applicant's evaluation on pages 3-17 and 3-18 of the SAR. Therefore, the staff finds it to be acceptable that a thermal acceptance test, as listed in section 8.1.7 of the SAR, is not required for the TN-40 package.

Evaluation Findings

The staff reviewed the package description and evaluation and concludes that they satisfy the thermal requirements of 10 CFR Part 71. Based on the review of the statements and representations in the application, the NRC staff concludes that the thermal design has been adequately described and evaluated, and that the thermal performance of the package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

The objective of this review is to verify that the package design satisfies the containment requirements of 10 CFR Part 71 under NCT (10 CFR 71.71) and HAC (10 CFR 71.73).

The applicant has requested, among other changes, the removal of the specification of the specific test method to be used for helium leakage testing, prior to shipment of the TN-40 package from the SAR, specifically from chapters 7, *Operating Procedures* and 8, *Acceptance Tests and Maintenance Program*. The revised SAR specifies that, prior to the first transportation of a TN-40 system loaded with spent fuel, a leakage test of the packaging will be performed to ensure that leakage from the package does not exceed the allowable combined reference air leakage rate of 1.0 $\times 10^{-4}$ ref cm^3/sec for the containment boundary.

Currently, the TN-40 package, must undergo a leakage test of the entire confinement boundary utilizing a “test envelope” method, as described in section 7.3.1.7A of the previous SAR, specifically page 7-5 of E-23861/Rev. 14. (ML13221A069), prior to shipment.

The applicant has provided a report entitled: “Evaluation and Recommendations for Leak Test of TN-40 Packages Prior to Shipment” (DI-10421-06), prepared by Leak Test Specialists (LTS) of Orlando, FL, which summarizes the possible alternatives to the “evacuated envelope” type of leak test which, as mentioned above, is specifically required under the current version of the SAR; however, the applicant has argued that conducting this type of test on a TN-40 package in the field, prior to shipment, may not be practicable.

Removal of the requirement as it currently appears in the SAR would provide the users of the TN-40 package with additional flexibility to employ an alternate leakage test that still meets the requirements in ANSI N14.5, prior to transport of a loaded TN-40 package in the future. The specific changes that were made to chapters 7 and 8 of the TN-40 SAR are discussed below. The applicant did not request any changes to the containment boundary components as currently defined in chapters 1 and 4 of the SAR and displayed in SAR figures 1-1 and 4-1.

Proposed SAR Changes

The applicant assumed that all operations of the TN-40 cask would start from the storage condition, where the system was already loaded with spent fuel with the lid emplaced and had been drained, dried and inerted (filled with helium) and then the seals in the lid were leak tested in accordance with ANSI N14.5.

In a supplement to their amendment application, the applicant requested that specific language be added to the operations chapter (chapter 7) of the SAR (Revision 17A) in section 7.1.3, in the note on page 7-3, in order to specify that if the lid bolts of the TN-40 must be replaced in preparation for transportation, and this replacement takes place in accordance with section 7.1.3, step 4 of the SAR, then a pre-shipment leakage test will not be required following the lid bolt replacement, assuming a perioding leakage test has been completed within a 12 month period prior to shipment.

This provision is based on the applicant's assertion that the process of replacing the lid bolts on the TN-40 one bolt at a time does not disturb the lid seal nor cause a loss of sealing as demonstrated by the applicant in a calculation, which was reviewed by the staff.

Descriptions of loading of the TN-40 package have been removed from the SAR (specifically by removing portions of section 7.1 from Revision 17A of the TN-40 SAR). The applicant has proposed to remove the description of a leakage test by a "test envelope" method, prior to shipment, that appeared in section 7.3.1.7A of the previous SAR, specifically page 7-5 of E-23861/Rev. 14. (ML13221A069).

In chapter 8 of the SAR, "Acceptance Tests and Maintenance Program", the applicant proposed to make minor changes to the text of the SAR, to coincide with the proposed changes to chapter 7 of the SAR, related to the leakage rate test to be administered to loaded TN-40 transportation packages prior to shipment.

Staff Review of LTS Report

The staff reviewed the TN-40 Leak Test Report entitled "Evaluation and Recommendations for Leak Test of TN-40 Packages Prior To Shipment", (December 7, 2021, Leak Testing Specialists, DI-10421-06) which was provided by the applicant as Enclosure 6 to the application. The TN-40 Leak Test Report provided a review of several potential testing approaches in addition to the evacuated envelope testing approach that was originally specified in the previous version of the TN-40 SAR. A description of each leakage test method, including specific concerns with the methods, and potential mitigations to the concerns highlighted for each method, were provided in the report.

The last section of the report, "Recommendations and Suggestions" provide some guidance to the applicant on leak testing of the previously loaded and stored TN-40s that are the subject of this amendment request. Since the applicant considers the report proprietary, only a general summary of the main points, as well as some of the recommendations and suggestions provided in the report, is provided below.

The report concluded that:

- Any gas accumulated in the annular space of the package should not be disturbed prior to any leakage testing.
- Consider leakage rate testing of the package boundary vs. the containment boundary.
- Measured leakage rates for the package must be summed.
- Development of any leak testing process should be started well in advance of any proposed shipment.

It is incumbent upon the entity responsible for demonstrating that a TN-40 package prepared for shipment meets the requirements for allowable leakage, provided for in 10 CFR 71.51, to implement an appropriate leakage test method that demonstrates compliance with the regulations, and any applicable standards.

It should be noted that the execution of a leakage test on a package prepared for shipment, including the development of a test procedure, the testing set up, the conduct of the test, the taking of data during the test, and the recording of results, as well as any post-test activities, including the final test report, are all subject to the NRC oversight, including inspection and review.

Findings

The changes proposed have no impact on the ability of the TN-40 design to maintain integrity of the containment boundary and prevent leakage or release of the radioactive contents of the package in accordance with the regulations specified in 10 CFR Part 71.
The staff reviewed the calculation, submitted by the applicant, which addresses replacement of lid bolts and the effect of lid bolt replacement on the maximum seal compression of the lid seal on the TN-40 package, and found that the applicant's conclusion that the lid seal of the TN-40 would not be impacted and the sealing of the lid would be maintained if the bolts were replaced in accordance with SAR section 7.1.3, step 4, is acceptable, and thus, it may be reflected as such in chapter 7 of the SAR.

The staff's approval of the applicant's request to remove the specification of leak testing method from their SAR does not provide relief from any commitments currently made by the applicant to conduct leakage testing of the TN-40 package in accordance with ANSI N14.5. The NRC staff remains committed to applying risk insights to applications that propose alternative methods from leak testing to demonstrate that the containment performance of the package meets applicable regulatory requirements.

Considering the above, the need to conduct a leakage test of the entire containment boundary of the package remains a requirement in SAR section 8.2.2 and, therefore, an appropriate test method will need to be determined and implemented by the user of the TN-40 package prior to shipment of any TN-40 loaded with spent nuclear fuel. The NRC should be made aware of the test method selection and provided with ample notification of any TN-40 package leakage testing that occurs.

The following certificate condition was introduced in this regard: Within 60 days of the first shipment of a shipping campaign involving any TN-40 package, the Certificate holder will notify the NRC of the leakage test method chosen to demonstrate compliance with the regulations in 10 CFR Part 71 related to leakage from the TN-40 package.

Based on review of the statements and representations in the application, the NRC staff finds that the operating procedures and acceptance and maintenance tests and have been adequately described and meet the requirements of 10 CFR Part 71.

7.0 OPERATING PROCEDURES

The staff reviewed the applicant's description of package operations to ensure that it is operated and used consistent with how the package was analyzed to meet regulatory dose rate limits. The package operations descriptions contain the essential elements of operations for using the package. Where alternates to sequences or operations are acceptable, the operations descriptions include these alternate sequences and operations. The staff finds that, based on its review, the operations descriptions in the application are consistent with these considerations.

One of the significant changes in this amendment request is the removal of the specification of the method to be used to perform this leak test of the containment boundary, in order to provide some flexibility regarding the choice of method used to perform this leak test.

The staff's approval of the applicant's request to remove the specification of leak testing method from the SAR does not provide relief from any commitment to conduct leakage testing of the TN-40 package in accordance with ANSI N14.5. The NRC staff is committed to applying risk insights to a future submittal proposing alternative methods from leak testing while demonstrating that the containment performance of the package always meets applicable regulatory requirements.

The applicant has also shown that replacing the lid bolts one at a time does not perturb the lid seal and does not cause a loss of sealing. The staff reviewed the calculation which addresses replacement of lid bolts and the effect of lid bolt replacement on the maximum seal compression of the lid seal on the TN-40 package, and found that the applicant's conclusion that the lid seal of the TN-40 would not be impacted and the sealing of the lid would be maintained if the bolts were replaced in accordance with SAR section 7.1.3, step 4, is acceptable.

The NRC staff has reviewed the description of the operating procedures and finds that the package will be prepared, loaded, transported, received, and unloaded in a manner consistent with its design.

8.0 ACCEPTANCE TESTS AND MAINTENANCE

Considering the above, the need to conduct a leakage test of the entire containment boundary of the package remains a requirement in SAR section 8.2.2 and, therefore, an appropriate test method will need to be determined and implemented by the user of the TN-40 package prior to shipment of any TN-40 loaded with spent nuclear fuel. The NRC should be made aware of the test method selection and provided with ample notification of any TN-40 package leakage testing that occurs.

The staff has reviewed the identification of the codes, standards, and provisions of the quality assurance (QA) program applicable to the package design and finds that they meet the requirements specified in 10 CFR 71.31(c) and 10 CFR 71.37(b). The staff has reviewed the identification of the codes, standards, and provisions of the QA program applicable to maintenance of the packaging and finds that it meets the requirements specified in 10 CFR 71.31(c) and 10 CFR 71.37(b). The staff has reviewed the description of the routine determinations for package use transport and finds that they meet the requirements of 10 CFR 71.87(b) and 10 CFR 71.87(g).

CONDITIONS

The following changes were made to the Conditions of the CoC:

Item no. 3.a was modified to reflect the new address of the CoC holder.

Item no 3.b was modified to refer to the latest application "TN-40 Transportation Packaging Safety Analysis Report, Revision 17, dated November 2023."

Condition No. 5(a)(3) was revised to update the licensing drawings to their latest revisions.

The following conditions were removed from the CoC:

> 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.

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-cm3/sec (at a sensitivity of 5.0x10-5 ref-cm3/sec or less) in compliance with ANSI N14.5.

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.

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.

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.

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.

The conditions were renumbered:

Condition No. 6(e) was added and reads as follows: Within 60 days of the first shipment of a shipping campaign involving any TN-40 package, the Certificate holder will notify the NRC of the leakage test method chosen to demonstrate compliance with the regulations in 10 CFR Part 71 related to leakage from the TN-40 package.

Condition No. 11 was revised to show the new expiration date of the certificate of December 31, 2028.

CONCLUSION

Based on the statements and representations contained in the application, and the conditions listed above, the staff concludes that the Model No. TN-40 package has been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

Issued with certificate of compliance No. 9313, Revision No. 4.