SAFETY EVALUATION REPORT (NON-PROPRIETARY)
Docket No. 71-9313
Model No. TN-40 Package
Certificate of Compliance No. 9313
Revision No. 0

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SUMMARY

By 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, and 27, 2011. Transnuclear, Inc. (TN) requested that the U.S. Nuclear Regulatory Commission approve the Model No. TN-40 as a Type B(U) package for transporting the spent nuclear fuel assemblies discharged from Prairie Island Nuclear Generating Plant (PINGP).

The packaging consists of a basket made from 40 stainless steel boxes joined by fusion welded steel plugs and separated by aluminum and Boral plates. The 160-in. long basket is placed inside the 163-in. long containment vessel which is made from a 1.5-in. thick steel cylinder, welded to a 1.5-in. thick bottom steel plate, and attached to a 4.5-in. thick closure lid outer plate with double metallic seals by 48 bolts. A 6-in. thick carbon steel gamma shield plate is also welded to the inside of the lid outer plate. The containment vessel is surrounded radially and on the bottom by 8-in. thick and 8.75-in. thick carbon steel as gamma shielding. Radial neutron shielding is provided by 4.5-in. thick borated polyester resin compound cast into long, slender aluminum alloy containers. A pair of impact limiters, consisting of balsa wood and redwood blocks, is encased in sealed stainless steel shells and is attached to the bottom and top of the cask by bolts to brackets welded to the outer shell of the casks. The impact limiters are also attached to each other by 13 tie rods. The nominal external dimensions, with impact limiters, are about 261 in. long by 144 in. wide. The weight of the loaded package is 271,500 pounds.

The package was evaluated against the regulatory standards in 10 CFR Part 71, including the general standards for all packages, standards for fissile material packages, and performance standards under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). The applicant demonstrated the structural integrity of the package by analyses and subjecting a one-third scale model to drops and punctures simulating the HAC described in the regulations. The tests and analyses showed that for NCT and HAC the package satisfies the regulatory requirements.

The TN-40 packages are already loaded for storage as authorized under SNM License No. 2506 (10 CFR Part 72). Some acceptance and maintenance tests which are normally performed on transportation packaging during fabrication or prior to loading, as a way to satisfy 10 CFR 71.85 and 71.87, have not been performed on the loaded TN-40 packaging units. In lieu of acceptance and maintenance tests, the user of the package will perform other tests and

analyses which are discussed in the following chapters and summarized in the “Conditions” section of this SER prior to shipment. These alternative tests and analyses satisfy regulatory requirements, in combination with the single use condition for each package. The single use condition prohibits any reuse of the packaging after unloading the original content if the packaging is used for transportation under 10 CFR Part 71. The single use of the limited number of 29 packages mitigates potential uncertainties associated with the sensitivity of the specific alternative tests and analyses methods. Thus, the staff has reasonable assurance that the TN-40 package design and acceptance tests satisfy 10 CFR Part 71. The approval of these alternative tests and analyses for this package design does not necessarily represent generic applicability to other future package design approvals for general license use.

The NRC staff reviewed the application using the guidance in NUREG 1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." Based on the statements and representations in the application, as supplemented, and the conditions listed below, the staff concluded that the package meets the requirements of 10 CFR Part 71 for single use of TN-40 transportation packages for transporting spent nuclear fuels discharged from the PINGP.

1.0 GENERAL INFORMATION

1.1 Packaging

The TN-40 is designed to transport up to 40 Pressurized Water Reactor (PWR) spent nuclear fuel assemblies discharged from the 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. The TN-40 packaging consists of a basket assembly, a containment vessel, a package body which also functions as the gamma shield, a neutron shield, and impact limiters. Four trunnions, which are part of the packaging, are used for lifting purposes. A transport frame, which is not part of the packaging, is used for tie-down purposes.

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 which sandwich 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 open dimension of each cell 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 containment boundary 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. Helium assists in removal of decay heat and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation which might otherwise lead to gross cladding rupture. The overall containment vessel length is approximately 170.5 in. with a wall thickness of 1.5 inches. The cylindrical cask cavity has a nominal diameter of 72.0 in. and a length of 163 inches. The lid outer plate is 4.5 in. thick and is fastened to the body by 48 lid closure bolts. Double metallic seals are provided for the lid closure. To preclude air in-leakage, the cask cavity is pressurized with helium above atmospheric pressure.

The cask cavity can be accessed using two penetrations through the lid. These penetrations are for draining and venting. 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.

The carbon steel packaging body which also functions as the gamma shielding is around the shell and the bottom inner plate of the containment vessel. The gamma shield completely surrounds the containment vessel shell and bottom plate. The 8.0 in. thick gamma shield shell and the 8.75 in. thick bottom shell are SA-105, SA-516, Grade 70, or SA-266 Class 4 material. A 6.0 in. thick shield plate is also welded to the inside of the lid outer plate.

Radial neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield shell. The resin compound is cast into long, slender aluminum alloy containers. The total radial thickness of the resin and aluminum is 4.50 inches. The array of resin-filled containers is enclosed within a 0.50 in. thick outer steel shell (SA-516, Grade 55 or equivalent) constructed of two half cylinders. In addition to serving as resin containers, the aluminum containers provide a conduction 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 internal pressure increase that may be caused by heating of the resin enclosure for HAC.

The impact limiters are attached to each other using 13 tie rods and to the cask by bolt attachment brackets welded to the outer shell in eight locations (four bolting locations per impact limiter). The impact limiters consist of balsa wood and redwood blocks, encased in sealed stainless steel shells (A-240, Grade 304) that maintain a dry atmosphere for the wood and confine the wood when crushed during a free drop. The impact limiters have internal radial gussets for added strength and confinement. The impact limiters have an outside diameter of 144 in., and an inside diameter of 92 in. to accommodate the cask ends. The bottom impact limiter is notched to fit over the lower trunnions. The impact limiters extend axially 37.75 in. from either end of the cask, and overlap the sides of the cask by 12.25 inches. Thirteen 1.5 in. diameter tie-rods are used to hold the impact limiters in place. The tie-rods span the length of the cask and connect to both impact limiters via mounting brackets. The impact limiters are also attached to the outer shell of the cask with eight 1.5 in. diameter bolts. The bolts are inserted through brackets (welded to the cask outer shell) and threaded into each impact limiter. There are a total of eight bracket sets, four per impact limiter. Each impact limiter is provided with nine fusible plugs that are designed to melt during a fire accident, thereby relieving excessive internal pressure. Each impact limiter has two lifting lugs for handling, and two support angles for holding the impact limiter in a vertical position during storage. The lifting lugs and the support angles are welded to the stainless steel shells. An aluminum spacer is placed on the cask lid prior to mounting the top impact limiter. The purpose of the aluminum spacer is to provide a smooth contact surface between the lid and the top impact limiter. The top plate of the spacer has 48 holes to allow clearance for the lid bolt heads. The lip of the spacer is designed to make up the difference between the lid and cask outer diameters so that the top impact limiter cavity mates with a surface of constant diameter.

Threaded holes are provided in the lid for attachment of component lifting devices. These are used as attachment points for sling systems or other lifting tools. These threaded holes are equally spaced 90° apart as shown on drawing 10421-71-4. Prior to transport, any attachments will be removed. Access to these threaded holes is prevented by the presence of the top impact limiter. Four trunnions, which form part of the cask body, are attached for lifting and rotating of the cask. Two of the trunnions are located near the top of the body, and two near the bottom. The upper trunnions are welded to the gamma shield shell. The lower trunnions are welded to the gamma shield shell and bottom shield, and are used for rotating the cask between the vertical and the horizontal positions.

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.

The applicant has described the packaging in sufficient detail in the Safety Analysis Report (SAR) to provide an adequate basis for its evaluation.

1.2 Contents

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 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

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 cask including the BPRAs and TPAs.

XIV. The boron-10 (B-10) in the neutron poison plates in the cask must be uniformly distributed in the plates with a minimum areal density of 10 mgram/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 shall not exceed 200 years.

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

Assembly Type	Exxon14x14 Standard	Exxon/ANF 14x14 High Burnup	Exxon/ANF 14x14 Top Rod	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 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 ID (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)	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 SER.

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

	Minimum Assembly Average 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 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.

The content has been described by the applicant in sufficient detail in the SAR to provide an adequate basis for its evaluation.

1.3 Criticality Safety Index

Criticality Safety Index (CSI): 0.0

1.4 Drawings

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

10421-71-1, Rev. 5.
10421-71-2, Rev. 2, sheets 1 and 2.
10421-71-3, Rev. 2.
10421-71-4, Rev. 0.
10421-71-5, Rev. 0.
10421-71-6, Rev. 0.
10421-71-7, Rev. 2.
10421-71-8, Rev. 0.
10421-71-9, Rev. 0.
10421-71-10, Rev. 0.
10421-71-40, Rev. 1.
10421-71-41, Rev. 1.
10421-71-42, Rev. 0.
10421-71-43, Rev. 0.
10421-71-44, Rev. 0.

Drawings provided in the SAR contain information which provides an adequate basis for the package's evaluation against 10 CFR Part 71 requirements. Each drawing is identified, consistent with the text of the SAR, and contains keys or annotation to explain and clarify information on the drawing.

2.0 STRUCTURAL EVALUATION

The objective of this review is to verify that the structural performance of the package has been adequately evaluated for the tests and conditions specified under NCT and HAC and the package design has adequate structural integrity to meet the requirements of 10 CFR Part 71.

2.1 Description of Structural Design

2.1.1 Descriptive Information, Including Weights and Centers of Gravity

The TN-40 transportation packaging consists of three major structural components: (1) the cask body, (2) the fuel basket, and (3) the impact limiters. Principal structural features of these components are explained below.

2.1.1.1 Cask Body

The cylindrical cask body is comprised primarily of a 1.5-in. thick inner shell and an intermediate 8-in. thick gamma shield shell to which a ½-in. thick outer shell is welded to provide a 4.50 in. annulus for positioning the resin-filled aluminum containers to provide primary neutron shielding. The 170.5-in. long by 75-in. diameter inner shell, together with the shell flange, bottom inner plate, lid outer plate, lid bolts, penetration cover plates and bolts, and the inner metallic seals of the lid seal and vent and drain seals, constitutes the containment boundary. The SA-203, grade D or E, inner shell is shrunk-fit inside the SA-266, CL4, SA-516, Grade 70, or SA-105 gamma shield shell, which, in turn, serves to protect the inner shell from being challenged by the loadings associated with the NCT and HAC events. Other structural components include two 11.25-in. diameter upper trunnions and two 8.88-in. diameter lower trunnions, which are fabricated with SA-105 or SA-266 Class 4 forgings and welded to the gamma shield shell to facilitate lifting and rotating of the cask.

2.1.1.2 Fuel Basket

The fuel basket is an assembly of 40 stainless steel cells joined by fusion welding them to steel plug spacers to allow placement of one poison and two aluminum plates between adjacent cell walls for forming a sandwich panel. The aluminum plates, which can bend independently of the cell walls in resisting out-of-plane loading, provide heat conduction paths from the fuel assemblies to the cask inner shell. The poison material is necessary for criticality control. The open dimension of each cell is 8.05-in. square for keeping a minimum clearance of 1/8 in. around the fuel assemblies for the 160-in. long fuel basket.

2.1.1.3 Impact Limiters

Two impact limiters made of balsa and redwood are used to protect the cask for impact loading in a cask drop event. Drawings 10421-71-40 through -44 present design details, including the shell enclosure, gusset partitions, and specifications for wood densities and moisture contents, for the upper and lower impact limiters of the cask. The impact limiters have an outside diameter of 144 in. and an inside diameter of 92 in. to accommodate the cask ends. They are attached to each other using 13 tie rods and each to the cask body by bolt attachment brackets welded to the outer shell in four locations.

2.1.1.4 Weights and Centers of Gravity

Table 2-6 of the application lists the calculated weights of the major package components. The system total weight, including fuel, impact limiters, and tie rods, is 271,460 lbs, which is bounded by the 271,700 lbs used for analyzing the package structural performance. The center-of-gravity location of the package is approximately 91.4 in. measured along the axial centerline from the bottom of the cask.

2.1.2 Design Criteria

Section 2.1.2 of the application summarizes the structural design criteria for the TN-40 system, including the applicable codes and standards as well as load combinations. These design criteria are reviewed as follows.

2.1.2.1 Codes and Standards

Sections 2.1.2.1 through 2.1.2.4 of the application present the applicable codes and standards, including stress allowable and performance criteria for various structural failure modes, for the structural design of the cask body, fuel basket, impact limiters, and trunnions, respectively.

The containment boundary as part of the cask body is designed to the maximum practical extent per the American Society of Mechanical Engineers (ASME) Code, Section III, Subsection NB and Appendix F stress requirements. The inner shell buckling evaluation follows the ASME Code Case N-284, "Metal Container Shell Buckling Design Methods, Section III, Division 1, Class MC." The lid bolt fatigue analysis considers appropriate ASME Code Appendix I fatigue curves. They are consistent with the Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessel, Rev. 1," design criteria. The cask body non-containment, gamma shield intermediate shell, and neutron shield outer shell are designed per the ASME Code, Section III, Subsection NF, criteria. As summarized in Section 2.3.4 of the application, the cask body and closure lid brittle fracture potential is evaluated per the NUREG/CR-3826, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater than Four Inches Thick," and NUREG/CR-1815, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick," fracture toughness criteria.

As stated in Section 2.1.2.2 of the application, the fuel basket is designed in accordance with ASME Code, Subsection NB to the maximum practical extent. This is deviated from NUREG/CR-3854, "Fabrication Criteria for Shipping Container," which provides that Subsection NG should be considered for the basket criticality control. As an ASME Code alternative, Section 2.11 notes that the basket fabrication and welding procedures are qualified by special inspections and tests. Since the NCT, Level A stress limits specified in Subsection NB are the same as Subsection NG and, for HAC, both Subsections NB and NG require use of Appendix F, Level D stress limits, the staff concludes that the applicant's stress evaluation approach meets the intent of the Subsection NG criteria as provided in NUREG/CR-3854 and NUREG/CR-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

There exists no industry code or standard for evaluating impact limiters. The impact limiters, which absorb or dissipate energy by crushing the energy-absorbing balsa and redwood, are designed to limit the maximum cask body inertia loads while remaining attached to the cask body for all NCT and HAC free drops. The impact limiter stainless steel outer shell and inner gussets, which support and protect the wood blocks under the NCT environmental conditions, are allowed to buckle and crush during the package drop events.

The front lifting trunnions of the cask are designed with a minimum factor of safety of six against yield and ten against ultimate strengths which bound the 10 CFR 71.45(a) requirement of a minimum safety factor of of three against yield for all lifting attachments that are structural parts of the package. Section 8.1.2 of the application notes that the front trunnions were load tested to 1.5 times of the design load for the single-load-path lift. This deviation from the ANSI (American National Standards Institute) N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More for Nuclear Materials," load test provision at three times the design load was previously evaluated and determined by the NRC to be acceptable for the cask storage use at the Prairie Island site. Since the package is proposed for a single use for the Prairie Island fuel, the staff agrees with the applicant that the load test configuration meets the intent of the NUREG-1617 provision on trunnion acceptance load testing per the NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," ANSI N14.6, or other appropriate specifications.

Tables 2-2 through 2-5 of the application summarize stress allowables for the containment boundary, lid cover bolt, non-containment structure, and basket, respectively. Additionally, Section 2.11 lists alternatives to the ASME Code alternatives and their justifications and compensatory measures.

The use of applicable codes and standards meets the intent of the regulatory requirements of 10 CFR 71.31(c), and is acceptable.

2.1.2.2 Load Combinations

The package is evaluated for a number of load combination cases for meeting the 10 CFR Part 71 requirements. Each load case as characterized by a combination of individual loads and applicable initial conditions is evaluated for the most limiting or bounding effects on structural performance. Table 2-11 of the application lists initial conditions, including ambient temperatures, internal pressures, and fabrication stresses, for all individual load cases evaluated and load combinations thereof. The load combination approach follows the Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Casks," guideline and is acceptable.

2.2 Material Properties

2.2.1 Contents

Forty unconsolidated undamaged 14x14 PINGP Pressurized Water Reactor (PWR) assemblies containing undamaged rods with Zircaloy-4 cladding are to be transported in the TN-40 package. The spent fuel rods must have a burnup <45 GWd/MTU and a maximum enrichment of <3.85% U-235. These assemblies must be of the five Exxon or Westinghouse designs specified in Section 1.2.3 of the SAR. The fuel may contain burnable poison rods and TPAs.

2.2.1.1 Damaged Fuel

A definition of damaged fuel consistent with ISG -1, Rev. 2 (Damaged Fuel) was provided in the SAR and added to the CoC. Known or suspected damaged fuel assemblies and fuel rods with cladding defects greater than pin holes or hairline cracks, or excessive bow will not be transported. No debris or assemblies with missing rods will be transported. Some TN-40 casks may have been loaded at a substantial time before transportation since this cask has been previously licensed for storage only. The physical condition of the fuel was known at the time of loading for storage. Once loaded, the fuel is dried either for storage under the storage SER or according to the procedure in SAR Section 7.1.3 acceptable to the staff. Although loaded prior to ISG-11 Rev. 3 (Cladding Considerations for the Transportation and Storage of Spent Fuel), the staff finds that the fuel was stored under atmospheric and temperature conditions that provides reasonable assurance of no additional cladding breaches. The applicant has indicated that in all cases where the cask has been loaded for storage, the maintenance records will be reviewed to determine if air may have entered the cask. This statement has been added to the CoC. If the records indicate air leakage into the container, the condition of fuel would be confirmed using the methodology given in ISG-22 (Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel) to determine the allowable content conditions are satisfied. The staff concludes this approach is sufficient to ensure fuel oxidation does not increase the number of damaged rods during storage prior to transport.

The maximum fuel cladding temperature will not exceed 400°C (752°F) for normal operations and 570°C (1058°F) for HAC. An inert cover gas will be used at all times. Under these conditions, as recommended in ISG-11, Rev. 3, fuel with zirconium base cladding and burnup <45 GWd/MTU is not expected to degrade.

The staff concludes that the definition of undamaged fuel and the reviews of fuel in storage assure that damaged fuel will not be transported.

2.2.1.2 Characteristics and Properties

The compositions of the alloys used in the assemblies (Table 5-5) were checked by the staff and found to be within specifications. Assembly and rod specifications in other tables were spot checked. While in most cases there was agreement, in some cases there were small discrepancies with the staff's reference values. However, these uncertainties were small and had insignificant implications for the criticality calculations.

The modulus of elasticity of the Zircaloy-4 cladding shown in Section 2.10.7.3 of the SAR were confirmed by the staff, using the Geehold-Bayer formula, and found to be acceptable. The staff finds this is appropriate since the data base was built on measurements of lower fluence material and the cask only contains fuel with a burnup <45 GWd/MTU. The thermal conductivity for the Zircaloy cladding given on page 3.7.1-2 of the SAR agrees with the MATPRO values. The emissivity of the oxidized Zircaloy rods is given on SAR page 3.7.1-2 as 0.8. The staff finds that this is typical of the emissivity of a thin layer of zirconium oxide [MATPRO] and shouldn't be affected by any further oxidation of the fuel rod while in the cask.

The values for the thermal conductivity of the UO_2 given in the SAR (SAR pages 3.7.1-1, 2), calculated from the SCALE code, do not agree with experimental data in MATPRO. Even though the values differ by 20-30%, the effect on the effective fuel traverse conductivity is small. The artificial value of the traverse conductivity used in the SAR is lower than predicted from either the SCALE or MATPRO conductivity. Using this lower conductivity, the ANSYS model predicts a maximum fuel temperature below the allowable maximum. Since no attempt was made to reconcile the thermal conductivities from the MATPRO experimental data, and SCALE calculations, the staff endorses neither of these values. Since a lower conductivity used in the SAR predicts acceptable temperatures for the fuel, reconciliation of the conductivity values will not be pursued by the staff.

2.2.1.3 Drying

After the cask is drained, the casks that are directly loaded for transport will be reflooded to replace seals and will be backfilled with helium in accordance with the recommendation in ISG-22. The cask drying is done according to the recommended procedure. The cask is evacuated to 4 x 10^{-4} MPa or less and then isolated from the pump. If the vacuum holds without raising another 4 x 10^{-4} MPa after 30 minutes the drying is satisfactory, otherwise the procedure is repeated. This is a method acceptable to the staff.

2.2.2 The Cask

2.2.2.1 Cask Materials

The inner shell and the bottom plate are made of SA-203, Grade D or E. The shell flange is SA-350 Grade LF3 and the lid outer plate is constructed with SA-350 Grade LF3 or SA-203 Grade E. The gamma shield shell and the bottom shield are SA-266, CL4, SA-516, Grade 70, or SA-105. The lid shield plate is constructed from SA-105 or SA-516 Grade 70. The cask interior is metal-sprayed with an aluminum/zinc alloy coating. All materials of construction are listed on Drawings Nos. 10421-71-1 and 10421-71-41. The staff checked that all materials used in this system can be subjected to a minimum environmental temperature under normal transport conditions of -40°C (-40°F) without adverse affects as required by 10 CFR 71.71(c)(2).

On page 2.10.1-3 of the SAR, the applicant states the ultimate strength, yield strength, Young's modulus and thermal expansion coefficient for the cask body, lid and bolts, as a function of temperature. Staff checked these values against American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section II, Part D, and all values were found to be accurate. The thermal conductivity for SA-203 stainless steel in the containment shell and SA-516 grade stainless steel for the outer shell and lid were both found to be correct. The values of the material thermal and mechanical design parameters for the lid bolt analysis were also checked and found to be accurate. The stress allowables on the closure bolts as a function of temperature given in SAR Table 2.10.2-3 were found by the staff to be accurate as were the mechanical and thermal properties for the steel and aluminum components of the basket given in SAR Table 2.10.5-1. These properties were all checked by the staff against ASME B&PV Code, Section II Part D. A hemispheric emissivity of 0.3 for 304 stainless steel, which the staff found to be consistent with the value in MATPRO, was used in the thermal analysis.

2.2.2.2 Welds and Codes

The ASME Code, Subsection NB rules for materials, design, fabrication, and examination are applied to all the components to the maximum practical extent. The containment vessel is designed to the ASME Code, Section III, Subsection NB, Article 3200, to the maximum extent practical. The containment vessel has materials selected in accordance with NB-2500, and is fabricated and examined in accordance with NB-4000 and NB-5000. The containment boundary welds consist of the circumferential welds attaching the bottom inner plate and the shell flange to the inner shell, and longitudinal weld(s) on the rolled plate, closing the cylindrical inner shell. Weld material conform to NB-2400 and the materials specification requirements of Section III, Part C, of the ASME B&PV Code. The containment vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200.

The gamma shield shell and neutron shield outer shell are designed, fabricated, and inspected, in accordance with the ASME Code Subsection NF to the maximum extent possible. Non-containment welds are inspected in accordance with the NDE acceptance criteria of ASME B&PV Code Subsection NF. Structural and structural attachment welds are examined by the liquid penetrant or the magnetic particle method, in accordance with Section V, Article 6, of the ASME Code, and acceptance standards in accordance with Section III, Subsection NF, paragraphs NF-5340 and NF-5350. The welders and welding procedures are qualified in accordance with Section IX of the ASME Code.

Any alternatives to the ASME Codes and alternative codes are listed by component along with the reference ASME Code and section, code requirement and alternatives cited. These alternatives are acceptable to the staff.

The staff concludes that the cask is constructed, welded, and inspected to the proper codes.

2.2.2.3 Fracture Toughness of Ferritic Steel

The cask body and closure lid are ferritic steel and are subject to fracture toughness requirements in order to assure ductility at the lowest service temperature of -29°C (-20°F). The analysis considers a weld defect of 1.26 cm in depth and 12 cm in length at 10 critical locations as depicted in SAR Figure 2.10.4-1. The calculations show that under both NCT and HAC, the applied stress intensity factors for those weld cracks are below the fracture toughness of the base material, SA-266, Class 4. The staff concludes that the defects would be stable, and would not pose safety issues.

Fracture toughness data are needed for the heat affected zone (HAZ) and the filler metal since these are the locations of the cracks in the welds. Accordingly, fracture toughness data of these materials should be used. The application stated that the toughness of these materials is higher than the toughness of the base material, which was used in the evaluation. The evaluation in the SAR shows that the TN-40 cask materials meet the fracture toughness criteria of NUREG/CR-3826 (Recommendations for Protecting against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater Than 4 Inches Thick) and NUREG/CR-1815 (Recommendations for Protecting, against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to 4 Inches Thick).

2.2.2.4 Gamma and Neutron Shield

Gamma shielding is provided around the inner shell and bottom plate of the containment vessel by an independent shell and bottom plate of carbon steel (SA-105, SA-516, Grade 70 or SA-266 Class 4).

The neutron shielding is provided by a proprietary borated polyester resin compound that surrounds the gamma shield shell and it is subject to thermal and radiation fields during service. These fields have the potential for degrading properties of the material including its thermal conductivity. The neutron shield resin is expected to withstand, without degradation, the maximum temperature of 149°C (300°F) it expects to see under normal operation. The neutron shield material has been tested by Transnuclearie Paris for thermal stability. Short term tests were conducted at temperatures above these maximum temperatures, which are expected during normal storage operation. The test results tend to be conservative (higher release than expected in an actual shield) due to the thinness of the samples. On the other hand, even with an apparent saturation of the weight loss, extrapolations from 100 hours to 20 years contain uncertainty. As a result, the staff cannot take these tests alone as proof of thermal stability.

Stability of the neutron shield is indicated by satisfactory dose rate measurements made periodically on many other types of casks using the same resin formulation at other sites. These dose measurements show no sign of shield deterioration. Dose rate and temperature measurements on the exterior of the cask will be made prior to shipment to assure that 10 CFR 71.47 requirements are met. These measurements will be an on site confirmatory test that the thermal and shielding capabilities of the neutron shielding material have not degraded while the cask is in the storage mode prior to transport. Based on the CERN testing, and dose surveillance to be made prior to shipment, the staff concludes that the resin used in the neutron shield is stable.

The staff concludes that radiation surveys of the cask exterior prior to shipment (SAR Section 8.2.4) will assure the limitations specified in 10 CFR 71.47 are sufficient and that the shielding material has not significantly degraded during the storage period.

2.2.2.5 Lubricants

Loctite N-5000 Nuclear Grade or Neolube is used on the bolt threads (SAR Section 2.4.4.3). Loctite N-5000 is a nickel based lubricant made for use with 304 stainless steel. According to the technical data sheet it has very low halide content and an operating range of 129 to 1315°C (264 to 2399°F). Neolube is a low halide graphite based lubricant made for use on stainless steel and other materials. According to the technical data sheet it has been used in fuel rods and is compatible with UO_2 pellets. It has an operating range of -57 to 204°C (-70 to 400°F) and can withstand fields up to 1×10^9 Rads. The staff concludes these materials have an applicable temperature range and compatibility for the designated purpose.

2.2.2.6 Seals

Double metallic O-ring seals of the Helicoflex HND type are used on the lid and the two lid penetrations (Drawing No. 10421-71-4, Rev. 0). The metallic seals have a stainless steel liner with an aluminum jacket and contain a Nimonic 90 or equivalent material spring. The seals have a long-term maximum operating temperature of 350°C (662°F) and can operate up to

550°C (1022°F) for short terms before annealing occurs. All seating surfaces are stainless steel clad. The staff concludes that no adverse chemical or galvanic interaction of the seal materials will occur.

2.2.3 Fuel Basket

2.2.3.1 Materials and Properties

The basket is constructed of type 304 stainless steel plates and 6061-T6 aluminum and designed to the ASME B&PV Code, Subsection NB to the maximum practical extent (SAR Section 2.1.2.2). The plates are formed into boxes by fusion welding. These boxes are separated by panels consisting of two aluminum plates sandwiching a poison plate. The basket is assembled by passing steel plugs through the bounding poison plates and fusion welding to the adjacent box section. The aluminum plate, outer plates, and basket periphery plates are made of SB-209 6061-T651 aluminum alloy. Staff confirmed that Young's modulus of both materials at 232°C (450°F) agreed with values reported in ASME B&PV Code, Section II, Part D. The thermal conductivity of the AL 6061 and type 304 stainless steel used in the basket (SAR pages 3-4) were checked by staff against ASME B&PV Code, Part D, and found to be accurate. Calculations[1] provided by the applicant demonstrate that insignificant creep of the aluminum plates would occur at the maximum service temperature over the lifetime of the basket.

2.2.3.2 Neutron Poison

Boral plates are used for the neutron poison; 75% credit is taken for the B-10 in the Boral (SAR 6.4.2). The minimum areal density of the B-10 is 10 mg/cm^2. Boral has been previously qualified and accepted by the staff as a neutron poison for this storage cask. The poison plates consisting of Boral sandwiched between two sheets of aluminum do not have any structural requirements and they serve no structural purpose.

Radiation resistance test results[2] are reported up to 7 x 10^{11} Rads gamma. The test specimens were in spent fuel pool water for nine years and were found to be severely oxidized. The oxidation could be removed from the specimens by brushing them with a wire brush. Aside from the corrosion, the specimens showed no other signs of physical deterioration. Neutron attenuation testing and neutron radiography, confirmed by chemical analysis, showed no loss of boron carbide. Tensile tests indicated no change in the ultimate strength. Mechanical testing, conducted on Boral heated to 316°C (600°F) for one week, showed no deterioration of the mechanical properties. These tests would be a conservative heat treatment based on the most severe thermal conditions that the Boral is expected to experience during vacuum drying. The staff concludes that the neutron absorber will not degrade due to the radiation dose or thermal environment in the storage cask.

The staff concludes that the neutron absorbers will perform their intended function.

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[1] TN Technical Report No. E-25768, Rev 0, "Evaluation of Creep of NUHOMS Basket Aluminum Components under long-term Storage Conditions", November 2007.

[2] Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications, EPRI, 2006 edition.

2.2.4 Galvanic Interactions/Gas Generations

Prior to transport, the interior of the cask was dried and backfilled with a helium atmosphere for storage or will be dried and backfilled with a helium atmosphere according to Section 7.1.3 of the SAR. Due to the lack of moisture and oxygen, the helium atmosphere does not support chemical or galvanic reactions with the steel or aluminum components of the basket or cask, or the Zircaloy components of the fuel assembly.

The aluminum/zinc coating may react with borated pool water; however, this reaction does not present a safety issue. Any hydrogen that may be generated would be removed prior to transport when the cask is bolted shut and the interior is vacuum-dried and backfilled with helium. Since the cask is a bolted closure, analysis[3] has shown that galvanic interactions and hydrogen generation are insignificant in the TN-40 cask. The staff concludes that galvanic interactions and gas generation are insignificant.

2.2.5 Impact Limiters

The impact limiters consist of balsa and redwood encased in stainless steel sheets. The specified adhesive used to glue the wood parts will meet the strength properties specified in Federal Specification MMM-A-188C. Typical properties of the balsa and redwood were specified by the applicant in Table 2.10.8-1, 2 of the SAR. Staff spot checked these values against those reported in the "Wood Handbook" and other references. Staff found the applicant's values to be accurate. The thermal conductivity for wood stated on SAR pages 3-7 agreed with the "Wood Handbook." A temperature limit of 110°C (230°F) was set for the wood to prevent excessive reduction of the structural properties.

Wood is an anisotropic material with materials properties that depend on the relative direction of the impact with relation to the grain direction. Properties are listed with the appropriate grain direction specified.

2.2.6 Aging Management Evaluation Prior to Shipment

No specific aging management plan is required during storage. This is due to the materials of construction of the cask and the content. In addition, any aging management plan that may be implemented at PINGP for long term storage use of the package under SNM License No. 2506 should be sufficient to mitigate significant aging effects. This should not be taken as a general precedent. The parts of the system that may undergo aging management are listed below and the reason for not requiring aging management is given, based on available information.

Aging Management Evaluation

Component	Reason
Fuel	The cask is properly vacuum dried and the low burnup fuel behavior in storage should be consistent with results of long term testing data that indicates no significant degradation for those conditions.

[3] Hydrogen Generation Analysis Report for TN-40 Cask Materials, Test Report No. 61123-99N, Rev 0, Oct 23, 1998, National Technical Systems.

Neutron Absorbers	The neutron flux is low and should not significantly deplete the B-10 content. The poison plates consisting of Boral sandwiched between two sheets of aluminum do not have any structural requirements and they serve no structural purpose. See SER Section 2.2.3.2 under basket neutron absorbers for discussion of aging effects during storage.
Basket	The basket is constructed of 304 stainless steel plates and 6061-T6 aluminum. The structural capabilities of the basket are provided by the stainless steel which is not expected to significantly degrade at the temperature and radiation levels in this cask during storage.
Cask Body	The ISFSI is not situated in a coastal region or known to be subjected to industrial pollutants that could cause significant degredation. The radiation dose and temperature during storage are too low to affect materials properties.
Neutron Shield	The package must undergo a neutron survey prior to shipment.
Seals	The seals will be replaced and leak tested prior to shipment.
Gamma Shields	The package must undergo a gamma survey prior to shipment.

2.2.7 Materials Evaluation

The SAR adequately describes the materials used for cask packaging components important to safety and the suitability of those materials to perform their intended function in sufficient detail for staff to evaluate their effectiveness.

The applicant has met the requirements of 10 CFR 71.33(a)(5)(ii), 43(f), and 47(a). Materials used for criticality control and shielding are adequately designed and specified to perform their intended function.

The selection of materials adequately protects the spent fuel cladding against degradation that might otherwise lead to gross rupture of the cladding.

The material properties of cask packaging components important to safety will be maintained during normal and accident conditions of operation so the spent fuel can be retrieved without posing operational safety problems.

The materials properties of cask packaging components important to safety will be maintained during all conditions of operation so the spent fuel can be safely transported within one year of loading.

The applicant has met the requirements of 10 CFR 71.43(d). The TN-40 cask employs materials that are compatible with wet and dry spent fuel unloading operations and facilities. These materials should not significantly degrade over time or react with one another during any of the conditions of transport analyzed by the staff. The staff expects the overpack to be subject to the maintenance and aging management programs for storage, that have or will be established under SNM-2506 at PINGP.

2.3 Fabrication and Examination

As noted in Section 2.11 of the application, with exception for the fuel basket, fabrication and examination specifications are prescribed for the packaging by referencing mainly Section III of the ASME Boiler and Pressure Vessel Code. Specific sections, divisions, subsections, and articles of the ASME Code for fabrications and examinations, including weld inspection, of different packaging components are also delineated in the drawings of Appendix 1.4, as appropriate, and are discussed in Section 2.1.2 of the application. The staff reviewed the applicant's approaches and concludes that they meet the intent of regulatory requirements of 10 CFR 71.31(c).

2.4 General Standards for All Packages

2.4.1 Minimum Package Size

The application notes the overall package dimensions of 260.87 in. in length and 144 in. in diameter, which exceed the minimum dimension requirement of 4 in. This meets the 10 CFR 71.43(a) requirements on minimum package size.

2.4.2 Tamper-Proof Feature

The application notes that the only access path into the package is through the closure lid with access ports and associated lid closure bolts. During transport the top impact limiter entirely covers and prevents access to the cask closure lid and the vent and access port penetrations in the lid. A wire security seal is installed in the front impact limiter attachment tie rod prior to shipment. The presence of this seal demonstrates that unauthorized opening of the package has not occurred, which meets the tamper-proof requirements of 10 CFR 71.43(b).

2.4.3 Positive Closure

Positive fastening of all access openings through the containment vessel is accomplished by the bolted closures. This precludes unintentional opening and meets the positive closure requirements of 10 CFR 71.43(c).

2.5 Lifting and Tie-Down Standards for All Packages

2.5.1 Lifting Devices

The TN-40 transportation package is equipped with two front and two rear trunnions for cask lifting and rotation, respectively. The front trunnions, which are groove welded to the cask body, in a non-redundant lift, are capable of supporting six and ten times the cask design weight of 250,000 lbs, without producing stresses, in either the shoulder or the welds, greater than the material yield and ultimate strengths. The load factor of six against yield strength is greater than the minimum safety factor of three required by 10 CFR 71.45(a). Table 2-8 of the application presents the trunnion stress results at the shoulder and weld locations. Section 2.5.1.1 calculates stresses in the trunnion-to-gamma shield welds, which govern the design, with acceptable minimum stress margins of safety of 0.06 and 0.40 against the yield and ultimate strengths, respectively. By noting that the capacity of the welds is less than that of the gamma shield shell, the application demonstrates that, for an excessive load, failure of the trunnion welds, as part of the lifting attachment points, would not impair the ability of the package to meet other Subpart C general license requirements of 10 CFR Part 71. This meets the lifting standards, including the excessive load provision, in accordance with 10 CFR 71.45(a).

2.5.2 Tie-Down Devices

Section 2.5.2 of the application notes that the longitudinal inertia forces experienced by the transport package, per 10 CFR 71.45(b), are resisted by the steel end restraints, which flush up against the impact limiters while the vertical and lateral forces are resisted by a dual saddle/strap tie-down system. Because of this tie-down configuration, the staff agrees with the applicant's assessment that there are no tie-down devices that are a structural part of the package. This meets the requirements of 10 CFR 71.45(b).

2.6 General Considerations for Structural Evaluation of Package

The application evaluates structural performance of the package by analysis and by testing.

For evaluation by analysis, both the finite element analysis and closed-form solution techniques are used. In performing the safety evaluation, the staff reviewed the analysis assumptions to ensure that the analysis methods had been implemented properly and results interpreted appropriately. The staff also focused on relevant boundary conditions used in structural analyses, including applicable temperatures, pressures, and drop orientations, for evaluating the most limiting results for the package.

For evaluation by testing, the applicant performed the fusion weld qualification tests, the fuel compartment wall panel load limit tests, and the scale-model cask drop impact limiter tests. A fusion weld qualification testing program, as presented in Section 2.10.5.4 of Appendix 2.10 to the application, is used to demonstrate that the welds joining the fuel compartment walls to the cylindrical spacer plugs are stronger than the base metal. The load limit tests of Section 2.10.5.5.3 provide additional information on temperature effects on wall panel performance for demonstrating that the fuel basket and compartment walls will not buckle when subject to the HAC side drop accident. As to the scale-model drop tests of the packaging, they are used primarily to confirm structural adequacy of the impact limiter design and to aid in determining the most damaging cask drop orientations and corresponding bounding baseline decelerations for cask structural components evaluation.

In the following the staff reviewed general considerations for the package with respect to structural evaluation by the finite element analysis and by testing.

2.6.1 Evaluation by Analysis

2.6.1.1 Finite Element Analysis Codes

As described in Appendix 2.10 to the application, the general-purpose finite element analysis code, ANSYS, is used to perform quasi-static stress analyses of the cask body and fuel basket for various loading conditions. For the 30-ft cask side-drop accident, the code is also used for analyzing fuel clad stress and the fuel basket buckling capacity. The explicit formulation dynamic analysis code, LS-DYNA, is used to perform transient dynamic analysis for evaluating structural integrity of the fuel rods and closure lid bolts subject to the 30-ft cask end-drop accident.

2.6.1.2 Finite Element Analysis Models

Appendix 2.10 to the application describes finite element analyses and associated modeling details for key packaging components, including the cask body consisting of the inner, gamma shield, and outer shells and the fuel basket with steel-to-aluminum-to-steel sandwiched construction for fuel compartment walls. For the element type selections, in addition to those for contact interfacing, typically used are brick and shell elements for the cask body and fuel basket components, respectively. As loadings and component configurations dictate, appropriate half-symmetry or periodic slice models are used.

In evaluating finite element models, the staff reviewed model attributes and corresponding assumptions to ensure that they had properly been implemented. This included considerations of nodal coupling, temperature-dependent material properties, force and displacement boundary conditions, and load combinations for effects of temperature, pressure, and drop orientation to result in the most limiting or bounding conditions. As considered for individual components, the application also presents key stress factors of safety, which must be shown greater unity, as ratios of the at-temperature stress allowables and the corresponding calculated stresses. For the fuel basket stress and buckling analyses, inelastic material properties with a 5% strain-hardening rate are considered for both the stainless steel and aluminum plates.

2.6.2 Evaluation by Testing

2.6.2.1 Qualification Tests of Basket Fusion Plug Welds

Section 2.10.5.4 of Appendix 2.10 to the application presents the testing program for the fuel basket fusion welds, which demonstrates that the welds are stronger than the base metal. As noted in Section 2.1.2.1 of this safety evaluation report (SER), this allows implementation of the ASME Code, Subsection NB or NG stress limits criteria for the fuel basket structural evaluation.

2.6.2.2 Load Limit Tests of Fuel Compartment Walls

Section 2.10.5.5.3 of Appendix 2.10 to the application presents the basket compartment wall load limit tests performed originally for the TN-40 storage cask system of the Prairie Island ISFSI (Docket 72-10). A series of six load capability tests, three at room temperature and three at elevated temperatures, were performed on the 8.05-in. long by 24-in. wide panels replicating essentially the prototypical, sandwiched wall construction subject to a uniformly applied in-plane compression along the hinged, opposite wide sides of a panel. Although the basket design weld spacing is 8 in. along the basket axial direction, the panels were tested with three spacing variations of 6 in., 8 in., and 12 inch. Considering the test performed at 276°C (529°F) for the case with a 12-in. spacing, which is more susceptible to buckling than the one with an 8-in. spacing, the applicant demonstrated a minimum tested buckling load of 9,219 lbs/in. for the TN-40 test wall panel. As reviewed in Section 2.8.1.4 of the SER below, together with the buckling load calculation, the staff determines that there is reasonable assurance that the fuel basket will not suffer buckling failure when the package is subject to a design basis side-drop deceleration of 72 g.

2.6.2.3 Scale Model Cask Drop Tests

Section 2.10.9 of Appendix 2.10 to the application presents the 30-ft drop tests of a 1/3-scale model of the package to demonstrate primarily that: (1) the crush depths of the impact limiters are acceptable and the impact limiters do not bottom out or lock up in that neither the cask neutron shield nor trunnions would touch the target; (2) the impact limiters remain attached to the cask body and in position after free drops; and (3) the measured cask rigid-body decelerations are bounded by the baseline decelerations used in the package design analysis. After the 30-ft free drop testing series, a 40-in. puncture end-drop was performed on a previously crushed impact limiter to evaluate puncture depth and damage to the impact limiter shell casing.

Using four impact limiters, in six 30-ft drops and one puncture drop, packaging orientations for the sequentially performed tests and their designations were: (1) side-drop-1, (2) C.G.-over-corner-drop, (3) side-drop-2, (4) 20-degree slap down, (5) end-drop-1, (6) puncture drop on end, and (7) end drop-2. The tests were conducted at prevailing ambient temperatures except for the impact limiter used in the first end-drop, which had been chilled at -29°C (20^{o}F) for 48 hours before re-installed on the test body. As discussed in the application, measured data from the first side- and end-drop tests were determined either unusable or inexplicable, which necessitated retests of two packaging configurations. The side-drop-2 used the undamaged portion of the two previously tested impact limiters, while the end-drop-2 used the No. 2 impact limiter, which had not been chilled again but deemed relatively undamaged in its center portion in a previous test and, therefore, reusable for a retest of different orientation to extract cask end-drop deceleration response.

Section 2.10.9.6 of Appendix 2.10 to the application summarizes the accelerometer data, crush depth measurements, and damage assessments as applied to the 0^o side-drop, 64^o C.G.-over-corner-drop, 20^o slap down-drop, and 90^o end-drop tests. Also presented is the puncture drop test, which resulted in the puncture bar penetration through the outer shell with insignificant internal damage to the impact limiter. In all of the drop tests, both impact limiters remain attached to the test article during and after the drops although some tie rods and brackets were damaged. Single tear of the impact limiter shell casing up to a few inches in length was also observed for most cask drop events except for the puncture drop. As listed in Table 2.10.9-1 and displayed below, the rigid body deceleration measurements are consistent with those of previously approved TN series of spent fuel transportation casks of comparable weights and wood impact limiter constructions.

30-FT Drop Orientation	Measured Rigid-Body Deceleration Reduced for Prototypical Application
90^o End Drop	54 g, Axial
0^o Side Drop	51 g, Transverse
CG-Over-Corner Drop, 64^o	34 g, Axial
20^o Slapdown Drop (Second Impact)	58 g, Transverse 62 g, Transverse (extrapolated to the outer surface of the cask lid)

2.6.2.4 Determination of Baseline G-Loads for Structural Evaluation

Section 2.7.1 of the application evaluates impact limiter test results presented in Section 2.10.9 of Appendix 10 to the application and determines the baseline rigid body decelerations for the package structural analysis.

By applying an impact limiter crush strength temperature correction factor of 1.15, which was established for the similarly constructed TN-68 transportation cask (CoC No. 9293, Docket No. 71-9293), the applicant derives below the corresponding rigid body decelerations applicable to the temperature condition cold at -20^{o}F for evaluating the cask body.

30-FT Drop Orientation	Bounding Test G-Load	Temperature Cold Corrected	Selected Baseline G-Load, Cask Body
End Drop	54 g, Axial	1.15 x 54 = 62 g	68 g
Side Drop	51 g, Transverse	1.15 x 51 = 59 g	68 g
CG-Over-Corner Drop	34 g, Axial	1.15 x 34 = 39 g	41 g, axial
20^{o} Slapdown Drop (Second Impact)	62 g, Transverse (at cask lid)	1.15 x 62 = 71 g	75 g

Considering a combined correction factor of 1.24 (1.15 x 1.08 = 1.24) to account also for a dynamic load factor (DLF) of 1.08, the baseline g-loads for the temperature cold fuel basket structural analysis are calculated as follows.

30-FT Drop Orientation	Bounding Test G-Load	Temperature Cold/ DLF corrected	Selected Baseline G-Load, Fuel Basket
End Drop	54 g, Axial	1.24 x 54 = 67 g	67 g
Side Drop	51 g, Transverse (mid-section)	1.24 x 51 = 63 g	63 g (uniform)
20^{o} Slapdown Drop (Second Impact)	58 g, Transverse (at basket top)	1.24 x 58 = 72 g	72g (top/bottom)

A DLF of 1.11 is considered in calculating the correction factors for the fuel rod side-drop evaluation. Similarly, for a combined correction factor of 1.27 (1.15 x 1.11), the applicant selects the following baseline g-loads for the fuel rod side-drop quasi-static structural analysis.

30-FT Drop Orientation	Bounding Test G-Load (g)	Temperature Cold/ DLF corrected	Selected Baseline G-Loads, Fuel Rod
Side Drop - Transverse	51 g, Transverse	1.27 x 54 = 65 g	75 g
Slapdown – 2nd Impact Transverse	58 g, Transverse	1.27 x 58 = 73 g	75 g

The selected rigid body baseline decelerations above envelope the applicable deceleration g-loads derived from the tests. Therefore, they are acceptable as bounding conditions for the quasi-static structural analysis of the packaging components.

2.7 Normal Conditions of Transport

Section 2.6 of the application presents analyses to demonstrate that the conditions and tests specified in 10 CFR 71.71(c) for NCT will impose no adverse effects on the structural performance of the package.

Table 2-10 of the application lists 15 individual load cases for NCT. Table 2-11 summarizes these loads and 15 load combinations for determining stresses in the cask body for the specified NCT conditions and tests. The structural analyses for the individual loads, including pressure, temperature, or mechanical, and the combinations thereof, are presented in various sections of Appendix 2.10. As reviewed in the following, the load combination results, which also include effects of bolt pre-load and fabrication stresses, form the basis for demonstrating structural adequacy of the package for meeting the 10 CFR 71.71(c) requirements.

2.7.1 Heat

Chapter 3 of the application considers the insolation, decay heat, and ambient temperature ranging from -20^{o}F to 100^{o}F to calculate steady state temperature distributions within the cask body and basket components. These results are used to establish the component at-temperature stress allowables and for determining the maximum normal operating pressure (MNOP) of 15.7 psig for the package evaluation.

Chapter 3 of the application calculates the thermal expansion between the inside diameter of the cask inner shell and outside diameter of the basket and concludes that, for a NCT temperature at 100^{o}F, differential thermal expansion (DTE) effects do not cause adverse thermal stresses in the fuel basket. This is acceptable. As also depicted in Table 2-11 for load combination case N1, in addition to the common individual loads associated with lid bolt pre-load and fabrication, thermal stresses and stresses due to a bounding cask internal pressure of 100 psig are considered for evaluating structural performance of the cask body. Table 2-13 summarizes peak stresses in various cask body components. As reported in Table 2-14, the governing inner shell peak stress intensities are 13.01 ksi and 13.47 ksi for the primary membrane and membrane-plus-bending stress categories, respectively. The corresponding factors of safety are 1.51 and 2.18, which are greater than unity and are acceptable. The stress evaluation demonstrates that the package meets the structural performance requirements of 10 CFR 71.71(c)(1) for the heat condition.

2.7.2 Cold

The cold environment condition is evaluated in Section 2.6.2 of the application, and considers an ambient temperature of -40°C (-40°F). Table 2-14 reports the governing stress intensity factors of safety of 1.40 and 2.02, which occur in the containment boundary inner shell, for the primary membrane and membrane-plus-bending stress categories, respectively. These factors of safety are acceptable to demonstrate that the package meets the structural performance requirements of 10 CFR 71.71(c)(2) for the cold condition.

2.7.3 Reduced External Pressure

Section 2.6.4 of the application considers an external pressure drop of 3.5 psig and conservatively uses a cask cavity internal pressure of 100 psig to analyze cask body stresses. This results in the same load combination as that for the hot environment condition. Thus, for the containment boundary inner shell, the same acceptable factors of safety apply. This demonstrates that the package meets the structural performance requirements of 10 CFR 71.71(c)(3) for the reduced external pressure.

2.7.4 Increased External Pressure

Section 2.6.3 of the application considers an increased external pressure of 20 psia and conservatively uses a net external pressure of 25 psig to analyze cask body stresses. The conservatively applied pressure is combined with other loads, including the -20^{o}F temperature for thermal stress consideration, to result in the same governing inner shell stress intensity factors of safety for the cold environment= condition reviewed above. This demonstrates that the package meets the structural performance requirements of 10 CFR 71.71(c)(4) for the increased external pressure.

2.7.5 Vibration

Sections 2.6.5 and 2.6.6 of the application consider transportation rail loadings of NUREG 766510, "Shock and Vibration Environments for Large Shipping Containers on Rail Cars and Trucks," to establish the resultant transverse loads of 6.65 g and 0.42 g for the shock and vibration conditions, respectively. In addition to load combination evaluation for the cask body components, they are considered in the fatigue analysis of the containment boundary. On the basis of 450 shipments, Section 2.6.13 determines a cumulative damage factor of 0.319 for the containment vessel for the combined effects of bolt preload, cask lifting, test pressure, road shock/vibration, pressure and temperature fluctuations, and 1-ft normal condition drop. The shock/vibration effects on the lid bolts are separately presented in Section 2.10.2 of Appendix 2.10 to the application, which determines that rail shock load in 50 shipments will result in a cumulative damage factor of 0.51 out of the total of 0.68. The load combinations evaluation of the cask body components and fatigue damage factor calculations for the inner shell and lid bolts demonstrate, in aggregate, that the package meets the structural performance requirements of 10 CFR 71.71(c)(5) for the vibration condition.

2.7.6 Water Spray

Section 2.6.7 of the application notes that all exterior surfaces of the cask body are metal and are not subject to soaking or structural degradation from water absorption. The staff agrees with the assessment and concludes that the package meets the requirements of 10 CFR 71.71(c)(6) for the water spray condition.

2.7.7 One-Foot Free Drop

Section 2.6.8 of the application presents structural analyses performed for the cask body components for six load combination cases in which the top, bottom, and side drops are combined with the closure bolt preload effect, fabrication stress, as well as applicable temperature and pressure loadings. The selection of the 1-ft side and end drops as governing conditions are consistent with that of other previously approved large TN spent fuel casks and is, therefore, acceptable. Table 2-13 summarizes stress results for 15 load combination cases for key cask components, including the lid, flange, inner shell, and gamma shield shell, bottom shield, and trunnion. Table 2-14 lists linearized stress evaluation for the most critically stressed cask body locations. For the cases involving the NCT free drops, the minimum factors of safety are 1.40 and 2.02 for the primary membrane and primary membrane-plus-bending stress intensities of the inner shell, respectively. They are greater than unity and are, therefore, acceptable.

Section 2.6.14 of the application describes the fuel basket structural analysis for end and side drops. Tables 2.10.5-4 through 2.10.5-6 of Appendix 2.10 to the application summarize stress results for the basket components. The calculated primary membrane and primary membrane-plus-bending stress intensities are all shown below the at-temperature allowable and are acceptable.

The staff reviewed the applicant's evaluation of the cask components and agrees that the structural performance of the TN-40 system satisfies the requirements of 10 CFR 71.71(c)(7) for the NCT free drop tests.

2.7.8 Corner Drop

The corner drop test, per 10 CFR 71.71(c)(8), does not apply because the cylindrical package weighs more than 100 kg (220 lbs) and is not of fiberboard or wood construction.

2.7.9 Compression

The compression test, per 10 CFR 71.71(c)(9), does not apply because the weight of the package exceeds 5,000 kg (11,000 lbs).

2.7.10 Penetration

Section 2.6.11 of the application notes that, due to a lack of external protuberances, the one meter (40 in.) drop of a 13 pound steel cylinder of 1-1/4 in. diameter, with a hemisphere head, is of negligible consequence to the package. The staff agrees with the assessment and concludes that the package meets the requirements of 10 CFR 71.71(c)(10) for the penetration test.

2.8 Hypothetical Accident Conditions

Section 2.7 of the application performs safety analysis of the TN-40 package subject to the HAC, per the 10 CFR 71.73 requirements. In the following the staff reviewed structural evaluations of the cask body, fuel basket, and impact limiter attachment for applicable HAC conditions and tests, including the 30-ft free drops associated with the baseline decelerations reported in Section 2.7.1 and subsequently found to be acceptable in Section 2.6.2.4 above in this SER.

2.8.1 Thirty-Foot Free Drop

2.8.1.1 Cask Body

Section 2.10.1 of Appendix 2.10 to the application provides details of the cask body structural analysis using an ANSYS finite element model comprised primarily of SOLID45 eight-node brick elements. A total of 20 individual loading cases are considered for load combinations for maximum stresses in the cask components, such as the lid, shell flange, inner shell, bottom plate, and gamma shield shell. The cases include bolt preload and lid seating pressure, fabrication stress due to inner shell shrink fit, internal and external pressures, hot and cold temperature distributions, transport tie-down, vibration, and shock forces, front trunnion lifting loads, cask-end and side-drop unit decelerations, and CG-over-corner and slapdown drops with concurrently applied axial and transverse deceleration components. Figures 2.10.1-9 through -24 depict loading and displacement boundary conditions for individual load cases. As provided in detailed assumptions and calculations, for various cask free-drops, inertia loads associated with the cask internals and impact limiters are simulated as applied pressures with appropriate distribution patterns. The implementation of loading and boundary conditions follows common structural analysis practices and is acceptable.

Section 2.7 of the application presents the evaluation and reporting method for cask body stresses by combining factored results of applicable individual load cases with appropriate stress multipliers, considering the design baseline deceleration g-loads. The stress multipliers used to linearly extrapolate individual case results for calculating load combination stress intensities are presented in Table 2.16. Table 2.10.1-2 of Section 2.10.1 to Appendix 2.10 lists maximum nodal stress intensities for individual load cases after applying appropriate stress multipliers. Table 2-19 summarizes the linearized stress intensities for evaluating structural adequacy of the most critically stressed locations of cask component sections of interest. These results are all within the allowable stress intensity limits, and are, therefore, acceptable.

2.8.1.2 Lid Bolt

Section 2.10.2 of Appendix 2.10 to the application performs a quasi-static bolt stress analysis, per NUREG/CR-6007, “Stress Analysis of Closure Bolts for Shipping Casks,” to evaluate the ability of the closure lid bolt to maintain a leak tight seal under the NCT and HAC tests and conditions. Individual loadings considered include bolt preload, gasket seating load, and loads associated with internal pressure, temperature, free-drop impact, and puncture.

By applying a bolt torque of 1,150 ft-lb to the closure bolt, a preload of 74,250 lbs, which corresponds to a pre-tension stress of 50,000 psi, is introduced to each of the 48 1.375-in. diameter bolts made of the SA-320 Grade LA43 alloy steel with minimum yield strength of 105 ksi at room temperature. Section 2.10.2.5 of Appendix 2.10 to the application determines that all bolt stress criteria, including average tensile, shear, combined stress, interaction equation, and bearing allowable are satisfied. However, for the load combination with the most damaging internal pressure of 100 psi and the CG-over-corner HAC free-drop, the bolt preload is shown lost to result in a seal decompression of 0.003 in. for some closure lid bolts. This is less than the allowable decompression of 0.04 in. and, thereby, demonstrating that the seal remains closed and the internal contents will not leak out during and after the 30-ft drop accident.

Section 2.10.2.7 of Appendix 2.10 to the application evaluates closure bolt fatigue performance for 50 spent fuel shipments by considering effects of the operating preload, test pressure, rail vibration/shock, pressure and temperature fluctuations, and 1-ft NCT free drop. The cumulative damage usage factor is determined to be 0.68, which is less than unity and is acceptable.

10 CFR 71.73(c)(1) requires that free-drop tests must be conducted with the specimen striking an essentially unyielding, horizontal surface in a position for which maximum damage is expected. By assuming that physical gaps may potentially exist between the cask and its contents, Section 2.10.11 of Appendix 2.10 to the application also performs a LS-DYNA transient dynamic analysis of an integral packaging model. The analysis is aimed at evaluating the lid bolts performance due to a delayed, secondary impact as associated with a scenario for which the contents moving independently from the cask but starting at the same terminal

velocity begin to catch up and land on the decelerating cask lid in a 30-ft cask top end-drop accident.

A finite element model of the cask system, which includes the impact limiter, overpack, as well as fuel assembly and basket equivalents, is used in a LS-DYNA transient dynamic analysis to capture relevant structural performance of the closure lid and lid bolts. To validate the model, Section 2.10.11.2 of Appendix 2.10 to the application focuses on benchmarking the impact limiter model by correlating the calculated values with the measured response of a dummy cask, a prototypical equivalent of the rigid steel cylinder used in the 1/3-scale impact limiter drop tests of Section 2.10.9. As displayed in a rigid body acceleration time-history plot, the calculated response peak is seen to envelope the temperature cold corrected cask end-drop baseline deceleration of 62 g. The calculated response pulse shape and duration also agree well with those recorded for the 1/3-scale impact limiter drop tests. This demonstrates that the impact limiter finite element model is adequately implemented.

Section 2.10.11.3 of Appendix 2.10 to the application adapts the benchmarked impact limiter model to evaluate delayed impact effects on the closure lid and the bolt. This is done by replacing the cask dummy with detailed model attributes of the overpack and lid closure system. It also includes appropriate mass and axial stiffness properties of the basket and fuel assemblies and associated gaps for simulating effects of delayed impact of the contents onto the cask lid during the 30-ft cask top end-drop accident.

Section 2.10.11.4 of Appendix 2.10 to the application evaluates transient dynamic responses of the lid closure system. An initial preload of 74.25 kips is introduced to the 1-3/8-in. SA-540 Grade B23 Class 1 lid bolts with the at-temperature tensile and yield strengths of 165 ksi and 147 ksi at 150 ^{o}F, respectively, for the 30-ft cask top-end drop analysis. The two gap cases considered are: (1) the base case with zero gaps between the cask and contents and (2) the bounding case with a 1.45-in. gap between the cask and fuel assembly, which account for differential thermal growth and radiation growth, and 3.0-in. gap between the cask and basket. For the basic case, in addition to satisfactory closure lid stress performance, the maximum average bolt axial stress of 62.3 ksi is seen much below the criteria of smaller of the yield strength, S_y, of 147 ksi and 0.7 times of the tensile strength, $0.7S_u$, of 115.5 ksi (0.7 x 165 = 115.5). The maximum membrane-plus-bending stress intensity, $P_m + P_b$, in the bolt is acceptable by virtue of insignificant bolt prying action associated with the closure system impact response to the cask end-drop accident. For the bounding case, the calculated maximum lid stress intensity of 41.8 ksi is less than the allowable of 49 ksi. The maximum average bolt axial stress of 139.1 ksi is below S_y, of 147 ksi but over $0.7S_u$, of 115.5 ksi. Correspondingly, the calculated maximum von Mises stress of 150.5 ksi and conservatively estimated membrane-plus-bending stress intensity of 151.92 ksi indicate that the maximum stress at the bolt cross section periphery is below the tensile strength, S_u, of 165 ksi, which satisfies the ASME Section III, Appendix F-1335.1, allowable tensile stress criterion.

The closure bolt maximum average stress reviewed above does not meet the ASME Code, Section III, Appendix F, criterion for tensile strength consideration. However, recognizing that the Appendix F criteria are for the quasi-static analysis results and in the absence of consensus industry standards for bolt stresses by the transient dynamic analysis, the staff, nevertheless, relies on the results to gain insights into delayed impact effects on bolt performance. As described in Chapter 7 of the application, prior to transport, a 0.75-in. thick by 71.75-in. diameter aluminum spacer will be installed between the cask and the payload. The resulting gap of 0.7-in. between the cask and fuel, which is much smaller than the assumed 1.45-in. in the analysis, tends to significantly mitigate the delayed impact effects on the maximum bolt average stress. Thus, given the rigor of the LS-DYNA finite element models used and the associated sensitivity analysis considering gap sizes, the staff has reasonable assurance to agree with the applicant's conclusion that the closure bolt design is adequate for the fuel loading configuration subject to the 30-ft cask end-drop accident.

Section 2.10.11.4 of Appendix 2.10 to the application determines the maximum separation between the lid and the cask body during the impact to be 0.051 in., which is greater than the static maximum allowable decompression of 0.04 in. of the seals. As depicted in a basket gaps-lid separation time-history plot, because of elastic bolt behavior, a gap will develop for a short duration of about 5 milliseconds and return to closed configuration subsequently. Therefore, the staff agrees with the applicant's assessment that the helium or radioactive material might leak but the amount during this short time period is insignificant.

On the basis of the above, the staff has reasonable assurance to conclude that the closure lid bolt system will perform adequately under the NCT and HAC tests and conditions, including the 30-ft cask top-end drop accident with assumed gaps between the cask and its contents.

2.8.1.3 Impact Limiter Attachments

Section 2.10.9 of Appendix 2.10 to the application presents the free-drop test results. The impact limiters were demonstrated to remain attached to the cask body during the 1/3-scale packaging drop tests. On this basis, the staff agrees with the applicant's conclusion that the attachment design is structurally sufficient.

2.8.1.4 Fuel Basket

Section 2.10.5 of Appendix 2.10 to the application evaluates the fuel basket by finite element analyses and by fuel compartment wall load limit tests, as appropriate, to demonstrate structural capabilities. Figure 2.10.5-2 delineates a three-dimensional finite element model, which is an 8-in. long representative slice of the basket, for the side-drop structural analysis with pressure simulated inertia loads applied on compartment wall panels. The model includes typical physical attributes of the two intermediate aluminum plates and a middle Boral plate sandwiched between the adjacent stainless steel tubes, which are, in turn, connected by fusion plug welds represented by short pipe elements. Section 2.10.5.2.2 describes element connectivity of the basic model involving displacement coupling of the steel tube walls and aluminum intermediate plates in the out-of-plane direction to simulate through-thickness support provided also by the intervening Boral plate.

Section 2.10.5.2 of Appendix 2.10 to the application presents stress analysis of the fuel basket using the basic, node-coupling, model. Considering a fuel assembly weight of 1,300 lb, inertia effects corresponding to the side-drop baseline 75-g deceleration are simulated with uniform pressures applied on compartment walls for the azimuth orientations of 0^o, 45^o, and 90^o, which are assumed to bound conditions associated with the most damaging drop angles. Tables 2.10.5-7 through -9 list governing results for the membrane and membrane-plus-bending stress intensity categories. They are much lower than the at-temperature allowables, for the basket components including the stainless steel tubes and rails as well as aluminum plates. Since the fusion welds connecting adjacent fuel compartment walls are demonstrated in a testing program, as described in Section 2.10.5.4, to be stronger than the base metal, the weld structural capability is deemed to be adequate by virtue of acceptable stress results for the fuel compartment walls. These evaluations, in aggregate, demonstrate adequate stress performance of the fuel basket. Additionally, for criticality safety consideration, the application uses compartment deformations to calculate a governing maximum relative deflection of 0.023 in. between two opposite compartment walls.

Section 2.10.5.3 of Appendix 2.10 to the application considers material nonlinearity, gaps, and initial imperfections of compartment walls in using the ANSYS large displacement and stress stiffening options, to evaluate plastic buckling loads of the fuel basket. Five drop orientations, at 0°, 30°, 45°, 60°, and 90° azimuth angles, are evaluated. In a sensitivity analysis considered for four azimuth angles of 0°, 30°, 45°, and 90°, Section 2.10.5.5.1 replaces the displacement node-coupling boundary conditions of the basic model with the ANSYS CONTAC52 elements to allow the two face plates of a sandwiched wall panel to separate away from each other for the compartment wall undergoing load-induced deformations. Table 2.10.5-11 presents the calculated buckling capabilities for those drop orientations, each with and without node coupling boundary conditions. The minimum buckling capacity of 88.54 g exceeds the design baseline decelerations of 72 g of slapdown drop and 63 g of the side drop accident, which correspond to the buckling capacity factors of safety of 1.23 (88.25/72 = 1.23) and 1.40 (88.52/63 = 1.40), respectively. By noting that, for the slapdown drop, a higher than actual fuel basket temperature was assumed in selecting material yield strengths and Young's moduli of elasticity for analyzing the basket top and bottom segments, Section 2.10.5.6 performs a reevaluation of basket performance for a decreased, but realistic, basket temperature. This is done by recognizing that the average temperature of 210°F in the basket periphery, which is applicable to the top and bottom ends of a fuel assembly in the cask slapdown drop, is much lower than the 336 °F used in the analysis. Making use of the test results from the compartment wall load limit test program reviewed in the paragraph below, the applicant determines that, at a temperature of 210°F, the fuel basket buckling capability can be estimated to be 96.9 g with a calculated safety factor of 1.35 (96.9/72 = 1.35) against the slapdown drop accident.

Section 2.10.5.5.3 of Appendix 2.10 to the application presents the basket compartment wall load limits tests performed originally for the TN-40 storage cask system of the Prairie Island ISFSI (Docket 72-10). A series of six buckling capability tests, three at room temperature and three at elevated temperatures of 365°F, 405°F and 529°F were performed on the 8.05 in. long by 24 in. wide panels, which replicate essentially the prototypical, sandwiched wall construction, subject to a uniformly applied in-plane compression along the hinged, wide panel sides. Although the basket design plug weld spacing is 8 in. along the basket axial direction, panels were tested with three individual axial weld spacing of 6 in., 8 in., and 12 inches. Considering a slight plate thickness difference between those of the test specimens and the prototypical construction, and for the test performed at 529°F for the panel with a 12-in. weld spacing, which is more susceptible to buckling than the one with 8-in. spacing, the application demonstrates a minimum testing buckling load of 9,219 lbs/in. Together with the test data reduced for five other tests, the application notes the average collapse loads of 13,777 lb/in. and 10,858 lb/in. at a room temperature 70°F and an elevated temperature of 433°F, respectively. From the two temperature dependent collapse loads, the applicant determines, by interpolation, that, at 210°F, the basket buckling load is 9.4% higher than that calculated for the temperature of 336°F. As reviewed in the previous paragraph, a 9.7% increase in the calculated buckling capacity corresponds to a factor of safety of 1.35 (88.54 x 1.097/72 = 96.9/72 = 1.35) against the slapdown drop accidents.

Section 2.10.5.6 of Appendix 2.10 to the application performs a summary evaluation of the fuel basket buckling capability considering the calculated minimum factor of safety of 1.35

acceptable although it is less than the NUREG/CR-6322, "Buckling Analysis of Spent Fuel Basket," provision (1.41 to 2.21) for the stainless steel basket construction. For the technical rationales provided by the applicant, the staff finds that adequate justifications have been provided for making the case because (1) the model configuration and analysis methodology used for a 360 degree sector of the TN-40 basket tend to mitigate the concerns commensurate with NUREG/CR-6322 guidelines for evaluating individual members of no integral boundary conditions associated with a fuel assembly, (2) the emerging consensus fuel basket design standards of the November 2010, draft ASME Code, Division 3, Subparagraphs 3229.2 and 3229.3 with a minimum factor of safety of 1.33 for the accident condition, and (3) the pressure simulated tributary mass distribution, which is much more than that of the actual at the fuel assembly ends, for resulting in a significant over-estimate of the design baseline deceleration of 72 g for the slapdown drop accident. Thus, the staff has reasonable assurance that the fuel basket will have adequate factors of safety against buckling failure for the 30-ft cask side and slapdown drop accidents.

On the basis of the review above, the staff agrees with the applicant's conclusion that the structural performance of the package satisfies the requirements of 10 CFR 71.73(c)(1) for free drop HAC.

2.8.2 Crush

The TN-40 package weighs more than 500 kg (1,100 lbs) and has an overall density greater than that of water. Therefore, the requirement of 10 CFR 71.73(c)(2) on the crush test does not apply.

2.8.3 40-inch Puncture Test

The application evaluates effects of a 40-in. free drop of the package onto an upright 6-in. diameter mild steel bar by both analysis and puncture drop test of a 1/3-scale packaging model. The drop test, as presented in Section 2.10.9 of Appendix 2.10 to the application for the end-drop orientation, demonstrates that the impact limiters will protect the ends of the cask body in that the puncture bar will be stopped by a thin wedge of the impact limiter wood that was compacted between the top of the puncture bar and the inner shell of the impact limiter.

Section 2.7.2 of the application presents an analysis of local damages to the gamma shield shell as well as the overall effects on packaging components associated with the cask body side landing on the puncture bar. Considering the maximum inertia force corresponding to the puncture bar yield strength of 50 ksi, the puncture bar force is determined to be 5.14 g, which is bounded by that of the 30-ft side-drop. Using an empirical equation of the Oak Ridge National Laboratory Report NSIC-22, "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants," the required shell thickness to prevent puncture bar penetration is calculated to be 2.69 in., which is much less than the gamma shield shell thickness of 8 inches. On this basis the staff concludes that the package meets the requirements of 10 CFR 71.73(c)(3) for the puncture test.

2.8.4 Thermal

Section 3.1.4 of the thermal portion of the SER discusses internal pressures which is discussed in further detail in Sections 4.2.2 and 4.3.3 of the SAR. Section 2.7.3 of the application notes the ANSYS transient thermal analysis of the cask for the fire accident from which the maximum thermal gradient is used as input for thermal stress analysis. As listed in Table 2-18, the resulting stresses in the cask components, including the lid, flange, inner shell, and gamma shield shell, are all acceptable. This meets the requirements of 10 CFR 71.73(c)(4) for the thermal test.

2.8.5 Immersion - Fissile Material

The package is subject to a head of water of 0.9 m, which is equivalent to an external pressure of 1.3 psig. This pressure is negligibly small compared to the external pressure of 290 psig for which the package is determined to be acceptable in Section 2.9 of this safety evaluation. On this basis, the staff agrees with the applicant's conclusion that the package meets the requirements of 10 CFR 71.73(c)(5).

2.8.6 Immersion - All Packages

The immersion test requirements are met because the effect of an external pressure of 21.7 psig caused by immersion under 50 feet of water is of negligible consequence, compared to that associated with an external pressure of 290 psig as evaluated in Section 2.9 of this safety evaluation. On this basis, the staff concludes that the package meets the requirements of 10 CFR 71.73(c)(6) for the immersion test for all packages.

2.9 Special Requirements for Irradiated Nuclear Fuel Shipments

Section 2.7.4.3 of the application evaluates the package to withstand an external pressure of 290 psig by performing a finite element analysis of the containment boundary components. Additionally, a buckling evaluation following the methods of ASME code Case N-284 was performed in Appendix 2.10.10 to include the effects of fabrication induced compressive stresses with those due to the 290 psi immersion pressure. The results show that the design has significant margins of safety. This demonstrates that the inner shell containment boundary is capable of withstanding the pressure without collapse, buckling, or water in-leakage for meeting the deep immersion test requirements of 10 CFR 71.61.

2.10 Internal Pressure Test

Section 8.1.2 of the application notes the pressure test at 25 psig on the cask assembly in accordance with ASME, Section III, paragraph NB-6200 or NB-6300. The test pressure, which is greater than 150% of the MNOP of 15.7 psig, meets the internal pressure test requirements 10 CFR 71.85(b).

2.11 Fuel Rods

Appendix 2.10.7 of Appendix 2.10 to the application evaluates the structural performance, including fuel rod buckling, of a typical PWR assembly to the 30-ft HAC side and end-drop events.

Section 2.10.7.1 follows a quasi-static analysis approach used previously by TN for fuel rod side-drop evaluation for a baseline deceleration of 75 g. As summarized in Table 2.10.7-3, the maximum calculated clad stress is less than the yield strength of the irradiated Zircaloy clad. Hence, the staff agrees with the applicant's conclusion that the fuel cladding will not fail under the HAC side-drop load.

Using the LS-DYNA code for transient dynamic analysis, Section 2.10.7.2 of the application evaluates fuel assembly structural performance under the 30-ft cask end-drop accident. The Westinghouse14x14 STD fuel assembly is analyzed as it is deemed more susceptible to structural bending deformation compared to other axially loaded fuel, including the Exxon fuel assemblies.

Section 2.10.7.2.1 validates a single-pin finite element model for analyzing cask end-drop effects on the spent fuel assembly, considering a coupled dynamic system comprised of the lumped cask mass, lateral springs representing the spacer grids, contact surfaces representing the basket compartment wall, and an axial spring simulating the impact limiter stiffness. As depicted schematically in Figure 2.10.7-4, the model replicates essentially all attributes of the B&W 15x15 fuel assembly, single-pin model reported in the paper, "Spent Nuclear Fuel Structural Response when Subject to an End Impact Accident (H. E. Adkins, Jr. et al., 2004)." Figures 2.10.7-8 through 2.10.7-10 display a close correlation of time-history results from the applicant's model and those of the paper, which include fuel rod axial displacement, velocity, global acceleration, and maximum and minimum axial strains. On this basis, the staff has reasonable assurance to conclude that the applicant's modeling approach is acceptable for calculating dynamic behavior of a fuel rod subject to the 30-ft cask end-drop accident.

Section 2.10.7.2.2 of Appendix 2.10 to the application adapts the validated single-pin modeling approach for analyzing the TN-40 spent fuel. This is performed by replacing relevant physical attributes of the validated model with those of the TN-40 transportation package loaded with the Westinghouse 14x14 STD fuel. It considers fully integrated shell elements for the fuel clad, lateral gap from the outside diameter of the rod to the basket wall, fuel cladding material properties, cladding thickness reduction to account for oxidation, and cask weight. Other modeling parameters evaluated in sensitivity analyses for determining bounding fuel rod performance include contact springs and gaps between the fuel bottom and cask, cask to ground impact limiter spring force-deflection representation, rod bowing between grid spacers, fuel pin internal pressure, and temperature dependent impact limiter crush strength effects. Since a maximum principal strain theory is being considered acceptable by the staff for evaluating either brittle or ductile failure of high burn-up spent fuel, the staff considers it equally applicable for the TN-40 spent fuel with regular burn-up. To demonstrate elastic fuel clad behavior, which suggests that a fuel assembly will deform but retain its original geometry during and after the 30-ft cask end-drop accident, the calculated maximum principal strain must be shown below the yield strain of 0.84% ($92.4 \times 10^3 / 10.98 \times 10^6 = 0.0084$) for the fuel clad with a Young's modulus of 10.98e6 ksi and yield strength of 92.4×10^3 ksi.

Section 2.10.7.2.2 of Appendix 2.10 to the application reports maximum calculated principal strains for three representative sensitivity analysis cases plus a bounding response case for which an initial gap of 1 in. is assumed between the pin and cask. The impact limiter force-deflection representation of the bounding case corresponds to a peak package deceleration of 62 g at the -29°C (-20°F) temperature cold condition. Since the calculated maximum principal strain of 0.70% is below the permissible, yield strain of 0.84%, the staff agrees with the applicant's conclusion that the fuel assembly cladding will not fail in the 30-ft cask end-drop accident. Thus, staff has reasonable assurance that fuel cladding will behave elastically and no permanent plastic deformation needs to be considered for the fuel criticality evaluation.

2.12 Evaluation Findings

The staff reviewed the statements and representations in the application by considering the regulations, appropriate Regulatory Guides, applicable codes and standards, as well as acceptable engineering practices. The staff concludes that the structural design is adequately described and evaluated and the structural performance of the package meets the requirements of 10 CFR Part 71, on the basis of the review findings, including:

1. The package structural design description meets the requirements of 10 CFR 71.31.

2. The codes and standards used in package design are acceptable.

3. To the maximum credible extent, there are no significant chemical, galvanic, or other reactions among the packaging components, among package contents, or between the packaging components and the contents in dry environment conditions. The effects of radiation on materials are considered and package containment is constructed from materials that meet the requirements of RGs 7.11 and 7.12.

4. The lifting and tie-down systems for the package meet the requirements of 10 CFR 71.45.

5. The package structural evaluation standards meet the requirements of 10 CFR 71.35, provided that the following conditions are satisfied: (1) The 48 as-installed 1.375-in. diameter SA-320 Grade LA43 closure lid bolts are replaced by the SA-540 Grade B23 Class 1 bolts of the same configuration, and (2) Prior to transport, a 0.75-in. thick by 71.75-in. diameter aluminum spacer is installed between the cask lid and the payload.

6. The packaging structural performance under the NCT will result in no substantial reduction in the effectiveness of the packaging.

7. The packaging will have adequate structural integrity under the HAC to satisfy the requirements of 10 CFR Part 71.

8. The containment structural performance will meet the 10 CFR 71.61 requirements for irradiated nuclear fuel shipments.

9. The containment structure is capable of meeting the 10 CFR 71.85(b) requirements for pressure test without yielding.

3.0 THERMAL

The purpose of this review is to verify that the package design meets the thermal requirements of 10 CFR Part 71 under NCT and HAC. The staff reviewed the thermal aspects of the TN-40 transportation package to verify that package performance has been adequately evaluated for the tests specified under NCT and HAC and that the package design satisfies the thermal requirements of 10 CFR Part 71.

3.1 Description of the Thermal Design

3.1.1 Packaging Design Features

In its transport configuration, the TN-40 packaging consists of a basket that is a welded assembly of stainless steel fuel compartment boxes separated by aluminum and poison plates which form a sandwich panel. The aluminum provides heat conduction paths from the fuel assemblies to the basket periphery plates and the cask inner shell. The inner shell is surrounded by a thick-walled, forged steel gamma shield shell, which is surrounded by radial neutron shielding material that is contained in long, slender aluminum boxes. The aluminum boxes are designed to fit tightly together against the enclosed steel outer shell and therefore improve the heat transfer across the neutron shield. A set of impact limiters consisting of balsa and redwood, encased in stainless steel shells, are attached to either end of the cask body during the shipment. The impact limiters serve as an insulator and provide protection to the lid and bottom regions during the HAC fire. A personnel barrier is also mounted to the transport frame to prevent unauthorized access to the package.

3.1.2 Content Heat Load Specification

The TN-40 is designed to transport 40 intact PWR fuel assemblies with or without fuel inserts. The total decay heat of the contents will not exceed 19 kW and the decay heat per assembly will not exceed 0.475 kW.

3.1.3 Summary Tables of Temperatures

Table 3-1 of this SER contains the maximum temperatures calculated by the applicant are given in. The summary tables of the temperatures of package components, Tables 3-1, 3-2, and 3-3 of the SAR, were verified to include the impact limiters, containment vessel, lid O-ring seal, fuel cladding, basket, and radial neutron shield. These temperatures were consistent with the temperatures presented throughout the SAR for both the NCT and HAC. The staff confirmed that the summary tables contained the design temperature limits for each of the critical components for both the NCT and HAC. For the HAC, the applicant reported the maximum transient temperatures for essential components, as well as the approximate time at which the maximum temperatures were reached. Based on analysis performed by the applicant, for the hypothetical fire accident, all components remained below their material property limits. The staff reviewed the temperatures and design temperature limit criteria for the package components and found to be consistent throughout the SAR.

Table 3-1 – Component Temperatures

Component	Temperature (°C (°F))		
	NCT	HAC	Maximum Allowable NCT/HAC
Outer Shell	101 (214)	584 (1,084)	*
Radial Neutron Shield	109 (229)	--	149 (300)
Inner Shell	122 (251)	206 (403)	*
Basket Rail	125 (257)	166 (330)	*
Basket (Fuel Compartments)	229 (444)	249 (480)	*
Gamma Shield Shell	120 (248)	368 (694)	*

Fuel Cladding	257 (495)	276 (529)	400 (752) / 570 (1,058)
Impact Limiter Wood	107 (224)	[3.]	110 (230)
Impact Limiter Surface	46 (114) [2.]	777 (1,431)	*
Bottom Inner Plate	112 (234)	173 (343)	*
Lid	89 (192)	143 (289)	*
Vent and Drain Port Seal	89 (192)	140 (284)	280 (536) / 280 (536)
Lid O-ring Seal	91 (195)	163 (325)	280 (536) / 280 (536)
Average Cavity Gas [1.]	174 (345)	197 (387)	*
Accessible Surface Temperature in Shade	57 (134)	N/A	85 (185) / N/A

1. An average cavity gas temperature of 176°C (348°F) and 227°C (441°F) is considered for calculating cavity gas pressure for NCT and HAC, respectively.
2. From Enclosure 3 to TN E-25513, impact limiter accessible surface temperature with personnel barrier and insolance. Without the personnel barrier and no insolence, the maximum accessible surface temperature is 98°C (208°F) which exceeds the allowable limit of 85°C (185°F). Therefore the personnel barrier cannot be removed for transport operation at the maximum analyzed heat load of 22 kW.
3. Temperature of impact limiter wood not provided for HAC due to wood char.
* The components perform their intended safety function within the operating range.

3.1.4 Summary of Pressures in the Containment Vessel

A discussion of the pressure in the containment vessel under the NCT and HAC is presented in the “Containment” section of the SAR. Staff reviewed this section and found it to be consistent with the pressures presented in the “General Information” and “Structural Evaluation” sections of the SAR. The maximum normal operating pressure was 15.7 psig for the containment vessel. The maximum pressure reported for the accident condition was 55.9 psig for the containment vessel. The design pressure for the cask cavity is 100 psig.

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

The applicant provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the package. The applicant used surface absorptivity values to model solar insolation into the package and emissivity values to model radiative heat transfer interaction between the environment and the package. The staff reviewed the thermal properties used for the analysis of the package. Although there were minor discrepancies between reference values and the values used in the analysis, TN demonstrated through sensitivity studies that the effect on temperature was insignificant. The staff determined that the values used were appropriate for the materials specified. The approach used by the applicant for applying solar insolation loads was consistent with the Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (NUREG-1617).

The applicant listed the properties of air in the SAR. These properties were utilized to analyze the conditions of the package required by 10 CFR Part 71 during normal conditions, low ambient temperature, and accident conditions.

3.2.2 Technical Specifications of Components

The applicant provided a reference (in Section 3.6 of the SAR) for the technical specification of the pre-fabricated package component which included Helicoflex seals (double metallic O-rings). The seals have a minimum and maximum temperature rating of -40°C (-40°F) and 280°C (536°F), respectively.

3.2.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of package containment, radiation shielding, and criticality were specified. The maximum allowable fuel cladding temperature of 570°C (1,058°F) is used by the applicant as a limit for the fuel cladding during HAC. This limit is justified and supported by the ISG-11, Rev. 3 "Cladding Consideration for the Transportation and Storage of Spent Fuel."

3.3 General Considerations for Thermal Evaluations

3.3.1 Evaluation by Analyses

The staff confirmed that the methods used for the thermal analyses were identified and sufficiently described to permit a complete and independent verification. The applicant used the ANSYS® finite element analysis code to perform the thermal evaluation of the package.

The applicant assembled several analysis models of the TN-40 to determine the temperatures that the components would experience during normal and accident conditions. The models are described below.

3.3.2 Thermal Models

The thermal model is a three dimensional model which has 90° symmetry and includes the complete package length. The model includes the geometry and material properties of the impact limiters, trunnions, neutron shield, cask shells, cask bottom plate, cask lid, basket, and fuel assemblies. The model simulates the effective thermal properties of the fuel with a homogenized material occupying the volume within the basket where the 144 in. active length of the fuel is stored. All gaps used in the model were specified in the SAR.

The neutron shielding consists of 60 resin-filled aluminum containers placed between the gamma shield shell and outer shell. The applicant included an air gap of 0.01 in. at the NCT thermal equilibrium condition between the aluminum resin boxes and the adjacent shells. No radiation heat transfer is accounted for across the gaps.

The basket structure is composed of 40 stainless steel boxes with two 0.25 in. thick aluminum plates and one 0.075 in. thick poison plate placed between adjacent boxes. The basket portion of the thermal model simulates the conduction paths provided by the aluminum plates, the stainless steel boxes and the fuel (modeled as a homogenous material). Virtually no thermal conductance is credited to the Boral® poison material. Aluminum plates (0.38 in. thick) are welded to the basket periphery to increase the surface area for heat transfer while providing some structural support for the basket. The aluminum rails, bolted to the inner shell, are sized so that heat is conducted from the basket periphery across a small gap. All decay heat is transferred from the basket to the inner shell across a helium gap via conduction.

Uniform gaps used in the basket model are described in Section 3.4.1.1 of the SAR. The use of uniform gaps is a conservative assumption, Although there will be imperfect contact between the adjacent aluminum plates, they will be in contact with each other at most of the locations. Benchmarking with test data from the TN-24P showed that the assumption of a 0.01 in. gap between adjacent plates in a basket similar to the TN-40 basket is conservative such that the maximum fuel cladding temperatures predicted in the finite element analysis are 42°C (108°F) to 43°C (109°F) higher than those obtained by measurements in the test. The 0.16 in. gap between the aluminum rail and basket plate is equal to the largest cold gap size and is therefore conservative. The 0.09 in. gap was modeled as 0.16 in. between the small conduction plate of the large rail and basket plate. TN performed a sensitivity analysis to see the effect of doubling the size of this gap to 0.32 inch. The analysis showed that there was a +1°F increase in the basket plate and fuel cladding temperatures which is a negligible increase in temperature. Appendix 3.7.2 of the SAR shows that the calculated hot gap between the basket and cask inner shell is smaller than the assumed gap of 0.1 in. and is therefore conservative.

The finite element model of the basket includes a representation of the spent nuclear fuel that is based on a fuel effective conductivity model. The decay heat of the fuel, 0.55 kW per assembly (22 kW total per cask), with a peaking factor of 1.2 was applied directly to the fuel elements. The total decay heat load used in this analysis exceeds the decay heat limit of the contents, 19 kW, and is therefore conservative. The effective properties for the homogenized fuel assemblies are described in Appendix 3.7.1 of the SAR.

The redwood and balsa within the impact limiters are modeled as a homogenized region with material properties as described in Section 3.2 of the SAR. TN showed by performing a heat balance on the TN-40 cask model for NCT at 38°C (100°F) ambient in the shade that the impact limiter gaps have no adverse effect on the maximum temperatures of the cask components.

3.3.3 Heat Dissipation

Heat is dissipated from the surface of the packaging by a combination of radiation and natural convection as described in Section 3.4.1.4 of the SAR. The Nusselt number correlations are valid for a wide range of Rayleigh numbers covering laminar to turbulent natural convection regimes. An explicit assumption of the natural convection regime is not required for the calculation of the convection coefficients.

3.3.4 Thermal Analysis Results

For the normal operating conditions, the applicant performed a steady-state evaluation of the entire model, which produced a maximum fuel cladding temperature of 257°C (495°F), which is below the limit of 400°C (752°F). The maximum lid O-ring seal temperature under normal conditions is 96°C (195°F), which is below the limit of 280°C (536°F). The maximum radial neutron shield temperature is 109°C (229°F), which is below the limit of 149°C (300°F). The maximum wood impact limiter temperature is 107°C (224°F), which is below the limit of 110°C (230°F).

The applicant utilized the model for the HAC analysis, but modified it to include crushed impact limiters discussed in Section 3.6.1 of this SER to properly assess the effects of the accident conditions on the package. These analyses produced a maximum fuel cladding temperature of 276°C (529°F) at 26 hours after the fire, which is below the limit of 570°C (1058°F). Under these conditions, the maximum seal temperature was shown to be 163°C (325°F) at one hour after the fire. This seal temperature for the 30 minute fire accident is below the limit of 280°C (536°F).

The TN-40 casks subject to transport are already loaded and each cask is limited to single use. To ensure the adequacy of the cask thermal performance in lieu of fabrication tests the following factors were taken into consideration. The analyzed decay heat is 22 kW which is greater than the content decay heat limit of 19 kW. The thermal margins on the fuel cladding and seals are discussed above during NCT. Thermal performance tests conducted on the TN-32, a similar design to the TN-40, showed that the thermal model adequately considers the insulating effect of the neutron shield, gaps, and uncertainties expected in the cask fabrication. A radiological survey over the cask outer surface will be performed prior to transport to serve as an indicator of the existence of excessive defects, cracks, or void spaces through the cask shells. A thermal survey over the outer shell, lid, and bottom plate to determine the maximum outer surface temperatures, as well as detect hot or cold spots in the surface profile will also be performed prior to transport. The measured maximum temperature will be compared to the calculated maximum outer shell temperature based on the thermal model described in the SAR with appropriate adjustments for decay heat and ambient temperature. An acceptance criterion of ±25°F (temperature difference between calculated and measured values) will be used for the temperature survey based on analytical temperature differences across gaps and the fuel cladding and seal temperature margins for NCT. This test will serve as an indicator of the thermal performance of the cask. Finally, all containment seals will be leak tested before shipment which serves as an indicator that the containment seals were not affected by the thermal performance of the cask. These factors ensure the adequacy of the thermal performance of the cask in lieu of fabrication tests.

3.3.5 Evaluation by Tests

Evaluation by thermal test of the package is not necessary since a thermal analysis has been performed.

3.3.6 Confirmatory Analyses

The staff reviewed the ANSYS models used in the thermal analysis. The staff checked the ANSYS models to confirm that the proper material properties and boundary conditions were used. In cases where the improper material properties or boundary conditions were used in the analysis package, the applicant provided sensitivity studies to show that there was minimal effect on peak temperatures as discussed in Sections 3.2.1 and 3.7 of this SER.

3.3.7 Effects of Uncertainties

The staff considered the applicant's thermal evaluations and ensured that they addressed the effects of uncertainties in thermal and structural properties of materials and in analytical methods. Because of significant design margins, the staff found reasonable assurance that the applicant used appropriate considerations throughout the application.

3.4 Evaluation of Accessible Surface Temperatures

The accessible surfaces of the TN-40 package include the personnel barrier and the outermost vertical and radial surfaces of the impact limiters, no surfaces of the cask body are accessible. The applicant analyzed the impact limiter surfaces under normal conditions in the shade and determined that the accessible impact limiter surfaces would not exceed 41°C (106°F).

The applicant described the personnel barrier that surrounds the cask body as having an open area of 80%. The applicant states that the presence of the barrier has negligible effect on heat transfer between the cask surface and the environment. The applicant performed a radiative heat transfer balance on the personnel barrier and determined that the personnel barrier would reach a maximum temperature of 57°C (134°F). Therefore the accessible surface temperatures in still air at 38°C (100°F) and in the shade remain below 85°C (185°F) in an exclusive use shipment.

Without the personnel barrier and without insolance, the maximum accessible surface temperature is 98°C (208°F) which exceeds the allowable limit of 85°C (185°F) in an exclusive use shipment. Therefore the personnel barrier cannot be removed for transport operation at the maximum analyzed heat load of 22 kW.

3.5 Thermal Evaluation under Normal Conditions of Transport (NCT)

3.5.1 Heat

The applicant performed steady-state calculations for an ambient temperature of 38°C (100°F) with solar insolation and a maximum decay heat of 0.55 kW per assembly utilizing the model described in Section 3.3 of this SER. Each fuel assembly in the model is represented as a three dimensional rectangular solid and is given an effective thermal conductivity according to the values calculated by the applicant in Appendix 3.7.1 of the SAR. The staff reviewed the applicant's model, procedures used to analyze NCT, and procedures used to apply the decay heat to the fuel regions and found them to be acceptable.

Although the maximum wood impact limiter temperature reaches 107°C (224°F), while the limit is 110°C (230°F), 11% of the wood in the rear impact limiter and approximately 5% of the wood in the front impact limiter of the TN-40 cask has temperatures between 71°C (160°F) and 107°C (224°F). Even if the high temperature wood with the grain oriented parallel to the cask axis loses 10% of its strength, it will have higher crush strength than the lower temperature wood with the grain oriented perpendicular to the cask axis. Therefore the effect of the high temperature wood on the crush analysis is insignificant.

3.5.2 Cold

At an ambient temperature of -40°C (-40°F) and no applied decay heat, the entire package will approach a temperature of -40°C (-40°F). The applicant reported temperatures based on an analysis of the models described in Section 3.3 of this SER for ambient temperatures of -29°C (-20°F) and -40°C (-40°F) in Table 3.2 of the SAR. The applicant concluded that package components, including the containment structures and the seals, would continue to function at this low temperature. The staff reviewed the information provided by the applicant and agrees with the applicant's assessment.

3.5.3 Maximum Normal Operating Pressure (MNOP)

The applicant calculates the MNOP within the containment vessel for NCT in the "Containment" section of the SAR. The maximum pressure reported for normal conditions is 15.7 psig for the containment vessel. The design pressure for the cask cavity is 100 psig. The MNOP for containment vessel is within the limits set by the applicant.

3.5.4 Maximum Thermal Stresses

The applicant reports maximum thermal stresses for NCT in Table 2.10.1-2 of the SAR. All thermal stresses are below the allowable stresses for critical package components.

3.6 Thermal Evaluation under Hypothetical Accident Conditions (HAC)

3.6.1 Initial Conditions

A full-length 90° symmetric package model as described in Section 3.4.1 of the SAR is used for the evaluation. The model is modified to represent two crushed impact limiters as described in Section 3.5.3 of the SAR. During the pre-fire, convection and radiation from the external surface of the model replicate the NCT analysis (38°C (100°F) ambient). During the fire, a constant convective heat transfer coefficient of 4.5 Btu/(hr-ft^2-°F) is used. All gaps are removed during the fire and restored immediately after the fire. A 30 minute 802°C (1475°F) temperature fire with an emittance of 0.9 and a surface absorptivity of 0.8 is applied to the model. An emissivity of 0.9 and an absorptivity of 1.0 are used for the package external surfaces after the fire accident condition. Appendix 3.7.3 of the SAR discusses a sensitivity study that documents the effects of the fire emissivity of 1.0 on the thermal performance of the TN-40. The decay heat load used in this analysis is 22 kW from 40 assemblies (0.55 kW/assembly) with a peaking factor of 1.2.

To maximize the effect of the fire on package components during and after the fire, the impact limiter finite element model developed in Section 3.4.1.2 of the SAR was modified to reflect deformation due to the 30-foot drop tests. The maximum amount of crush experienced by the impact limiter in a given direction is assumed to occur everywhere on the limiter. Based on the side-drop, the impact limiters are modeled with a uniform diameter of 117.2 in. for the top impact limiter and 116.8 in. for the bottom impact limiter. Based on the corner-and end-drops, the impact limiters are modeled with a uniform axial length of 20.4 inches. The applicant showed that deformations predicted using the ADOC computer code were in agreement with the 1/3 scale dynamic testing.

The applicant states that although the impact limiters are locally deformed during the 30-foot drop, they remain in place on the cask. The applicant also states that although unlikely, the worst-case damage due to hypothetical puncture conditions may result in the tearing of the outer steel skin of the front impact limiter, crushing the wood out of the damaged area, and exposing the partially contained wood to the hypothetical fire conditions. Based on a study of fire performance of wood at elevated temperatures and heat fluxes, the finite element model of the inner surface of the impact limiter inner cover is exposed to 600°C (1,112°F) maximum char wood temperature for 30 minutes immediately after the end of fire. No heat dissipation is considered for the open surface of the torn wood segment after the period, assuming that the surface is entirely covered with a thin layer of low conductivity wood char. The worst case scenario occurs when a middle segment of wood (I.D. 44 in. to O.D. 88 in., 90°) is torn.

3.6.2 Maximum Temperatures and Pressures

The maximum component temperatures are bounded by the case of deformed impact limiter with torn middle segment. Table 3-1 of this SER presents the bounding maximum temperatures of the package components during and after the fire event for deformed and torn impact limiter. While the aluminum boxes containing the neutron shielding provide a conduction path between the outer shell and the gamma shield during the HAC fire, the boxes are thermally modeled as described in Section 3.3.2 of this SER, and the components in Table 3-1 of the SER are below the maximum allowable temperature limits. The bounding values calculated for the maximum seal and the fuel cladding temperatures are 163°C (325°F) and 276°C (529°F), respectively. While the transient average cavity gas temperature peaks at 198°C (387°F), an average cavity gas temperature of 227°C (441°F) is used for calculating the cavity pressure. The corresponding peak cavity pressure assuming 100% fuel failure is 55.9 psig, which is less than the design pressure of 100 psig.

The applicant states that the TN-40 maintains containment during the postulated sequential drop, puncture, and fire accident. The neutron shield will off-gas during the fire event. A pressure relief valve is provided on the outer shell to prevent the pressurization of the outer shell. The shielding integrity of the neutron shielding is assumed to be lost after the fire event. The maximum seal temperature is 163°C (325°F), below the 280°C (536°F) limit, and the fuel cladding temperature is 276°C (529°F), below the limit of 570°C (1,058°F).

3.6.3 Maximum Thermal Stresses

The applicant reports maximum stresses for HAC fire accident load combination in Table 2.18 of the SAR. All thermal stresses are below the allowable stresses for critical package components.

3.7 Appendices

The applicant provided three appendices to Chapter 3 of the SAR, Appendix 3.7.1, “Effective Thermal Properties for the Fuel Assembly,” Appendix 3.7.2, “Justification of Hot Gap between Basket and Cask Inner Shell,” and Appendix 3.7.3, “Sensitivity Study for Effects of the Fire Emissivity.”

In Appendix 3.7.1, in order to determine the effective fuel assembly thermal conductivity, effective fuel density, and effective specific heat, the applicant reviewed the 14x14 PWR fuel assemblies to be transported in the TN-40 cask. This review allowed the applicant to select the fuel assembly or parameters that would provide the most conservative effective thermal conductivity.

The applicant calculated the effective conductivity values in the axial and transverse directions separately. The transverse fuel effective conductivity (k_{trans}) is determined by creating a two-dimensional quarter symmetry finite element model of the fuel assembly centered within a basket compartment using the ANSYS computer code. The outer surfaces, representing the fuel compartment walls, are held at a constant temperature, and a decay heat is applied to the fuel pellets within the model. The 2-D model simulates heat transfer by radiation and conduction, includes the fuel rods and guide tubes, and helium is used as the fill gas in the fuel assembly. A fuel assembly heat load of 0.671 kW, a negligible difference from the 0.675 kW decay heat load per assembly for the TN-40 storage cask that has no effect on k_{trans}, is used for heat generation. The applicant neglected radiation between the fuel pellet and cladding. No convection is considered within the fuel assembly model, heat transfer from the fuel rods to the fuel compartment walls is through conduction and radiation.

The applicant performed a sensitivity study comparing the effect of UO_2 conductivity values obtained from SCALE and MATPRO on k_{trans}. Using the fuel pellet thermal conductivity from MATPRO, k_{trans} is 1.5% lower than using fuel pellet thermal conductivity from SCALE. However, both the SCALE and MATPRO sets of calculated k_{trans} are 20% higher than those used in the original analysis which can be seen for comparison in Figure 3.7.1-4 of the SAR. The original analysis k_{trans} values are lower due to the use of an incorrect Stefan-Boltzmann constant, which appeared to be due to a unit conversion error.

Because the k_{trans} values used in the original analysis are lower than the k_{trans} values based on the correct Stefan-Boltzmann constant, the NCT results in Section 3.4.2 of the SAR are considered conservative and were not changed. The applicant performed a sensitivity study to show the effects on the HAC component temperatures. The results of the study showed that the HAC results in Section 3.5.4 of the SAR are considered conservative and were not changed. The results of this study are presented on page 3.7.1-7 of the SAR. The staff notes the HAC study results take into account the lower initial NCT temperatures based on the k_{trans} values that were arrived at using the corrected Stefan-Boltzmann constant.

Values for fuel assembly effective axial conductivity, effective transverse conductivity, effective specific heat, and effective density are given in Section 3.7.1.6 of the SAR.

Appendix 3.7.2 of the SAR shows that the calculated hot gap between the basket and cask inner shell is smaller than the assumed gap of 0.1 in. and is therefore conservative.

Appendix 3.7.3 of the SAR discusses the effects of changing the fire emissivity from 0.9 to 1.0. Table 3.7.3.1 of the SAR shows the results of the sensitivity study. The cask outer surface transient temperature increases the most by 30°C (54°F) while the lid seal increases by 4°C (8°F) and fuel cladding increases by 1°C (2°F). Steady state component temperatures increase by at most 1°C (2°F). The applicant concludes that the effect of increasing the fire emissivity from 0.9 to 1.0 lasts only for a short period of time on the outermost components of the package exposed to the fire.

The staff evaluated the information provided in these appendices to make its safety findings regarding the adequacy of the design when compared to the requirements in 10 CFR Part 71.

3.8 Evaluation Findings

1. The staff has reviewed the package description and evaluation and has reasonable assurance that the information provided satisfies the applicable thermal requirements of 10 CFR Part 71.

2. The staff has reviewed the material properties and the component specifications used in the thermal evaluation and has reasonable assurance that the information

provides sufficient basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

3. The staff has reviewed the methods used in the thermal evaluation and has reasonable assurance that the models are described in sufficient detail to permit an independent review of the package thermal design. The application of the analysis methods, presented in the SAR, to this package design has been found to be adequate.

4. The staff has reviewed the accessible surface temperatures of the package, as it will be prepared for shipment, and has reasonable assurance that the requirements of 10 CFR 71.43(g), for packages transported by exclusive-use vehicle, have been satisfied.

5. The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that the package material and component temperatures will not extend beyond the specified allowable limits during NCT consistent with the tests specified in 10 CFR 71.71.

6. The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-term limits during HAC consistent with the tests specified in 10 CFR 71.73.

7. In lieu of a thermal acceptance test that has not been performed on already loaded casks, a temperature survey will be performed on each loaded cask and the results compared to calculated outer shell temperatures from SAR thermal model analysis with appropriate adjustments for decay heat and ambient temperature. In addition, all containment seals will be helium leak tested at the package final destination. These two factors, in addition to conservatisms in analysis decay heat compared to content decay heat, thermal testing on similar designs, and the performance of radiological surveys prior to transport, ensure the adequacy of the thermal performance of the casks in lieu of a traditional thermal acceptance test for the TN-40 package.

4.0 CONTAINMENT

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

4.1 Description of Containment System

4.1.1 Containment Boundary

The containment vessel of the TN-40 cask is designed to prevent the leakage of radioactive material from the cask cavity and maintain an inert atmosphere in the cask cavity. The containment boundary components of the TN-40 package consist of inner shell, bottom inner plate, shell flange, lid outer plate, vent port cover, drain port cover, and associated seals and bolts. The containment vessel has two penetrations of drain port and vent port in the lid. A double seal mechanical closure is provided for each penetration and a bolted cover is incorporated to each penetration.

The applicant specified that helium is maintained in the cask cavity to assist heat removal and protect fuel assemblies against fuel cladding degradation, in compliance with 10 CFR 71.43(d). Additionally, there is no significant degradation of any safety components caused directly by the effects of chemical, galvanic, or other reactions or by reactions combined with the effects of long term exposure of the materials to neutron or gamma radiation, high temperatures, or other possible ambient or operating conditions.

The staff verified that components of the containment system are well defined in Chapters 1 and 4 of the SAR, displayed in SAR Figures 1-1 and 4-1, SAR drawings of 10421-71 (sheets 1-10), and confirmed that the TN-40 package design is described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR 71.31(a)(1), 71.31(a)(2), 71.33, and 71.43. The staff also verified that: (1) the maximum temperatures under NCT and HAC are below the limits and below the melting/ignition points and (2) the package is made of non-reactive materials and is filled with an inert, non-explosive gas mixture of helium, xenon, krypton, and iodine to assure that there will be no significant chemical, galvanic, or other reactions, in compliance with 10 CFR 71.43(d).

4.1.1.2 Welds

The containment boundary welds of the TN-40 package are circumferential welds attaching the bottom inner plate and the shell flange to the inner shell. Also the longitudinal welds on the rolled plate, closing the cylindrical inner shell, and the circumferential welds, attach the rolled shells together, are the containment welds.

The applicant delineated that the welding is performed using qualified processes and qualified personnel, according to the ASME Boiler and the Pressure Vessel (B&PV) Code. Both base materials and welds are examined in accordance with the requirements of ASME B&PV Code. The applicant specified NDE requirements for welds on the drawings of SAR 10421-71 (sheet 3) and performed all NDE in accordance with approved procedures.

The staff reviewed the drawings of SAR 10421-71 (SAR, Drawings 10421-71-1 to 10421-71-10), Chapter 4, “Containment,” Chapter 7, “Operating Procedures,” and Chapter 8, “Acceptance Tests and Maintenance Program” for the containment related welds of the TN-40 cask and ensured that all containment boundary welds are to be examined and inspected appropriately in accordance with ASME B&PV Code, Section III, Division 1, Subsection NB.

4.1.1.3 Closure Bolt

The TN-40 package includes a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by pressure that may arise within the package, in compliance with 10 CFR 71.43(c). The applicant provided the closure bolt analysis in SAR Appendix 2.10.2. The bolt torque required to seal the metallic seals located in the lid and to maintain containment under normal and accident conditions is ~1125 ft-lb and is provided in the Drawing No. 10421-71-1 of SAR Appendix 1.4. The bolt torque required to seal the metallic O-ring seals located in the vent and drain port covers is around 40~44 ft-lb with the lubricant of Neolube, Loctite N-5000, or equivalent used to achieve this torque.

The applicant specified that the TN-40 package contains no valve, only quick connect couplings in the vent and drain ports and these couplings are not part of the containment. The applicant enhanced the closure by providing the vent and drain ports with a double seal mechanical enclosure to retain any leakage from the failure of the couplings.

4.1.1.4 O-Ring Seals

The TN-40 package has double metallic O-ring seals on the lid and the two lid penetrations to provide the long term stability and possess the high corrosion resistance (see SAR drawing 10421-71-4). After the loading for transport, the helium leakage rate test should be performed on the entire containment boundary (including all lid and cover seals), by placing the cask in a test envelope, to an acceptable total cask leakage below 1.0 x 10^{-4} ref-cm^3/sec, in accordance with ANSI N14.5.

The applicant performed both NCT and HAC analyses in Chapter 3 of the SAR to evaluate the temperatures of O-ring seals at the closure lid. The applicant predicted the maximum O-ring seal temperatures of 90°C (195°F) under NCT and 163°C (325°F) under HAC, which are below the allowable limit of 280°C (536°F) for Helicoflex metallic seals under NCT and HAC. The staff confirmed that the temperature of containment boundary seals at the closure lid will remain within their specified allowable limits under both NCT and HAC and the thermal performance of containment seals under the design heat load satisfies 10 CFR 71.51(a)(1) and 71.71 under NCT and 10 CFR 71.51(a)(2) and 71.73 under HAC.

The applicant showed the allowable temperature range of O-ring seals in SAR Table 3-1. The staff verified that the thermal performance evaluation under extreme cold conditions (ambient air temperature of -40°C (-40°F)) requires no thermal calculation because the -40°C (-40°F) temperature is within the allowable operating temperature range of from -40°C (-40°F) to 280°C (536°F) for the TN-40 package and therefore ensured that the O-ring seals remain intact under cold conditions and meets the requirement of 10 CFR 71.71. The staff also confirmed that the O-ring seals are appropriate for its intended use and no galvanic or chemical reaction will occur between the seal and the packaging or its contents, in compliance with 10 CFR 71.43(d).

4.1.1.5 Description of Containment System Summary

The staff reviewed the containment design features presented in SAR Chapters 1 and 4 and verified that the application defines the boundary of the containment system, including containment inner shell, lid outer plate, closure bolts, inner O-rings, shell flange, vent port cover plate, bolts at vent port, seals at vent port, drain port cover plate, bolts at drain port, and seals at drain port (SAR Figure 4-1). The applicant also adequately describes the vent and drain boundary penetrations and their method of closure. The description of the containment system is in compliance with 10 CFR 71.33. The staff ensured that all components of the TN-40 containment system are shown in SAR drawings Nos. 10421-71-3, 10421-71-4, and 10421-71-5.

4.1.2 Codes and Standards

The applicant specified in SAR 4.1.1 that the containment vessel is designed to ASME B&PV Code, Section III, Subsection NB, Article 3200 to the maximum practical extent. The containment vessel is fabricated and examined in accordance with NB-2500, NB-4000, and NB-5000. The weld materials meet the requirements of ASME B&PV Code, Section III, Division 1, Subsection NB and material specifications of Section II, Part C of ASME B&PV Code. The containment vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200.

The staff reviewed SAR Chapter 2 and NUREG/CR-3019, “Recommended Welding Criteria for Shipping Containers,” and determined that the information regarding the components of the containment boundary is consistent with the descriptions of welds, bolt, material construction, and applicable codes and standards presented in SAR Chapter 2, “Structural Evaluation.”

4.1.3 Special Requirements for Damaged Spent Nuclear Fuel

The TN-40 cask would be used to transport intact PWR fuel assemblies with or without fuel inserts. Damaged fuel assemblies are not allowed in the package. The staff determined in its review that helium would be maintained in the cask cavity to assist heat removal, maintain an inert atmosphere in the cask cavity, and protect fuel assemblies against fuel cladding degradation.

4.1.4 Special Requirements for Shipment of Plutonium

The applicant specified in Section 4.5 of the SAR that the TN-40 transport packaging support having plutonium in the solid form in the fuel rods of spent fuel assemblies, and meets 10 CFR 71.63, which specifies that, “shipment containing plutonium must be made with the contents in solid form, if the contents contain greater than 0.74 TBq or 20 Ci of plutonium.”

4.2 Containment Under Normal Conditions of Transport (NCT)

The TN-40 cask is designed, constructed and prepared as a Type B package for shipment so that there is no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of 10^{-6} A_2 per hour, in accordance with 10 CFR 71.51(a)(1) under the tests specified in 10 CFR 71.71 for NCT.

4.2.1 Pressurization of Containment Vessel

The applicant specifies in Section 4.2.2 of the SAR that the TN-40 cask cavity is drained, dried, and evacuated prior to backfilling with helium at the end of fuel loading operations. If the TN-40 cask contains design basis fuel and has been in storage, the maximum cask cavity temperature is 205°C (401°F) under the ambient air of 38°C (100°F), the maximum solar load, and the MNOP of 2.2 atm (17.6 psig).

Staff agrees that the temperature of 205°C (401°F) and the pressure of 2.2 atm (17.6 psig) are appropriate as the base values for the MNOP calculation because of the continuity when the TN-40 is converted from its storage configuration to the transportation configuration.

4.2.2 Cavity Gas Temperature

The applicant presented in SAR Table 3-1 that the maximum cavity gas temperature is 176°C (348°F) under the hot environmental conditions and the maximum initial cavity pressure is 2.0 atm just prior to the shipment during NCT. The staff checked the TN-40 Prairie Island ISFSI Safety Analysis Report, Revision 0, for storage and confirmed that both temperature (348°F) and pressure (15.7 psig) during normal transport are bounded by the temperature (401°F) and the pressure (17.6 psig) during normal storage when there is no fuel rod rupture. The bounding conditions of the TN-40 transportation package, under NCT, are in compliance with the requirements of 10 CFR 71.35 and 71.71.

4.2.3 Maximum Normal Operating Pressure (MNOP)

The applicant delineated in Section 4.2.2 of the SAR that the mechanisms contributing to the containment pressurization are the ideal gas heating and the release of the fission gas from the fuel rods. The applicant also applied a 3% fuel rod failure with the cavity gas mixture of 97.7% helium from cask backfill operations and rod pre-pressurization, with the balance consisting of 2.0% xenon, 0.2% krypton, and 0.1% iodine. This gas mixture is not explosive. The determination of fission gases is based on the grams of fission gases from SAS2H /ORIGEN-S computer runs, which utilize fuel with 45,000 MWd/MTU bundle average exposure, 3.8 wt% U-235 initial bundle average enrichment and a 15 year cooling time. The applicant derived the MNOP of 30.4 psia or 15.7 psig, using the method described in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," and the conditions of

- 30% release rate of fission gas from fuel pellets into the gap between the fuel pellets and the cladding,
- maximum cavity gas temperature of 348^{o}F (initial condition) under hot environment conditions, and
- neglect of the gas volume inside the fuel rods when calculating the cask free volume.

The applicant noted in Section 4.2.2 of the SAR that the TN-40 containment system with MNOP greater than 19.7 psia (5.0 psig) must be subjected to a structural pressure test with test pressure at least 1.5 times MNOP in accordance with 10 CFR 71.85(b). Therefore, the TN-40 cask would normally be tested at a pressure of 25.0 psig in a hydrostatic pressure test to verify the capability of the containment system to maintain its structural integrity without visible leakage. However, this hydrostatic pressure test can be exempted because it is bounded by the internal pressure test (with 25.0 psig) as described in Section 2.10 of this SER.

The staff reviewed the assumptions and methods used in the MNOP calculation, described in Section 4.2.2 of the SAR, and performed a confirmatory analysis of the calculation of the MNOP using MS Excel. The staff confirmed that the evaluation of the MNOP (30.4 psia or 15.7 psig) in the TN-40 package design is acceptable and the proposed hydrostatic test at a pressure of 25.0 psig provides assurance of the integrity of the package and meets the requirements of 10 CFR 71.85(b) and ASME B&PV code, Section III, Subsection NB.

4.2.4 Containment Criteria

The applicant specified in Section 4.2.1.1 of the SAR that three sources are considered to determine the releasable airborne material from the TN-40 cask:

- the residual activity on the cask interior surfaces due to loading operations;
- the fission- and activation-product activity on the fuel assembly surfaces due to corrosion-deposited material (crud deposition); and
- the radionuclides within the individual fuel rods due to cladding breaches (gases, volatiles, and fuel fines).

The applicant neglected the sources of residual activity on the cask interior surface due to loading operations because this is negligible when compared to the crud deposition on the fuel rods, but conservatively assumed that both crud deposition and cladding breaches occur instantaneously after fuel loading and closure operations for a maximum of source releases. The staff agrees with neglecting the residual activity on the cask interior surface because of its minor contributions to the release of airborne material and the conservative assumption that both, crud deposition and cladding breaches, occurs simultaneously.

The applicant used the radionuclide inventory and A_2 values listed in Table 4-1 of the SAR and calculated the source activity from the release of crud, volatiles, gases, and fuel fines using the methodology in 10 CFR 71.71 and NUREG/CR-6487, "Containment Analysis for Type B Packages Used to Transport Various Contents," a 14x14 WE STD fuel assembly (39,000 MWd/MTU burnup, 3.3 wt% U-235 initial bundle average enrichment and 15 years cooled). The resulting activity concentrations of volatiles (3.64×10^{-6} Ci/cm^3), gases (9.65×10^{-5} Ci/cm^3), fines (3.54×10^{-7} Ci/cm^3), and crud (4.21×10^{-6} Ci/cm^3) are displayed in SAR Table 4-2. Then, the applicant calculated the mixture effective A_2 value of 56.6 Ci (SAR Table 4-3) by using the relative release fraction for each inventory. Finally, the applicant calculated the standard reference leakage rate of 1.0×10^{-4} ref-cm^3/sec, under a cavity gas pressure of 2.5 atm and a conservative cavity gas temperature of 431^{o}F (or 495^{o}K) under NCT.

The staff verified the radionuclide inventory and A_2 values for all radionuclides in the TN-40 cask by checking the Table A-1 in Appendix A of 10 CFR Part 71, the equations of source activity in NUREG/CR-6487, and the staff performed a confirmatory analysis using MS Excel and following the guidance of ANSI N14.5 and validated the calculations of the effective A_2 value of 56.6 Ci and the standard reference leakage rate of 1.0×10^{-4} ref-cm^3/sec. The staff also verified that the data in SAR Table 4-1, "Radionuclide Inventory and A_2 values)," Table 4-2, "Activity Concentration by Source," Table 4-3, "Effective A_2 Values," and Table 4-4, "Permissible Leakage Rates," are consistent and acceptable for determining the NCT leakage rates of the TN-40 package, in accordance with ANSI N14.5 for fabrication, maintenance, periodic, and pre-shipment verification.

4.2.5 Compliance with Containment Criteria

The applicant determined in SAR, section 4.2.1, that the permissible standard leakage rate under NCT is 1.0×10^{-4} ref-cm^3/sec. The staff accepted that the spent fuel contents of the TN-40 were fully described to determine the containment criteria (including fuel type, fuel amount, percent enrichment, burnup, cooling time, and decay heat) to demonstrate that there was no release or dispersal of radioactive contents, as demonstrated to a sensitivity of 10^{-6} A_2 per hour. The staff found that the TN-40 package meets the containment requirements of 10 CFR 71.51(a)(1) for NCT with no dependence on filters or a mechanical cooling system, which is also in compliance with 10 CFR 71.51(c).

4.3 Containment Under Hypothetical Accident Conditions (HAC)

The containment requirements under HAC are specified by 10 CFR 71.51(a)(2) as "no escape of krypton-85 exceeding 10 A_2 in one week and no escape of other radioactive material exceeding a total amount of A_2 in one week." The review procedures for containment under HAC are analogous to those under NCT.

4.3.1 Pressurization of Containment Vessel

The applicant conservatively assumed 100% fuel rod rupture with the cavity gas mixture consisting of 60.5% helium (from cask backfill operations and rod pre-pressurization), 34.3%

xenon, 3.4% krypton, and 1.5% iodine, which are calculated with depletion code SAS2H under HAC and displayed in SAR Table 4-6. The applicant used this non-explosive gas mixture for evaluating the internal pressure of TN-40 containment vessel. The applicant added the calculated pressure of 2.57 atm due to 100% fuel rod failure to the initial pressure of 2.23 atm at NCT and obtained the cavity gas pressure of 4.80 atm (70.6 psia or 55.9 psig), which is well below the package design pressure of 7.80 atm (114.7 psia or 100 psig).

The staff agreed with the conditions used for the pressure calculation of the package under HAC, including temperature, pressure, and release of gases from fuel rod cladding breaches, and confirmed the calculated cavity gas pressure under HAC through:

- evaluating the conservative assumption of 100% fuel failure;
- examining that the released gases (xenon, krypton and iodine) from the rod failure are not explosive, based on their chemical properties and characteristics, and that there are no additional pressures generated by explosion; and
- validating the pressure calculations through a confirmatory analysis using MS Excel and the methodology in ANSI N14.5.

4.3.2 Containment Criteria

The releasable source term, maximum permissible release rate, maximum permissible leakage rate, and conversion to the reference to the reference air leakage rate should be based on the TN-40 package conditions and 10 CFR Part 71 containment requirements under HAC.

The applicant used the radionuclide inventory and A_2 values listed in SAR Table 4-1 and calculated the source activity from release of crud, volatiles, gases, and fuel fines following the same procedures as described under NCT (see Section 4.2.4 of this SER). The applicant displayed the resulting activity concentrations of volatiles (1.21 x 10^{-4} Ci/cm^3), gases (1.91 x 10^{-4} Ci/cm^3), Krypton-85 gas (3.03 x 10^{-3} Ci/cm^3), fines (1.18 x 10^{-5} Ci/cm^3), and crud (2.81 x 10^{-5} Ci/cm^3) in SAR Table 4-2 under HAC 100% fuel rod failure. The applicant derived the mixture effective A_2 value of 8.8 Ci using the relative release fraction for each inventory and calculated both the maximum permissible leakage rate of 4.11 x 10^{-2} cm^3/s and the standard reference leakage rate of 1.27 x 10^{-2} ref-cm^3/sec (SAR Table 4-4), under a cavity gas pressure of 5.5 atm and a cavity gas temperature of 531°F under HAC.

The staff verified the calculation of the maximum permissible leakage rate of 4.11 x 10^{-2} cm^3/s and the standard reference leakage rate of 1.27 x 10^{-2} ref-cm^3/sec through the confirmatory analysis using the Excel spread sheet and following the guidelines of ANSI N14.5. The staff verified that the data in SAR Tables 4-1, 4-2, 4-3, and 4-4 are consistent and acceptable for determining the leakage rate of the TN-40 package under HAC.

4.3.3 Compliance with Containment Criteria

The staff confirmed that both HAC with Krypton-85 and without Krypton-85 are limited to the permissible standard leakage rate of 1.27 x 10^{-2} ref-cm^3/s (air) to meet the allowable release rates of 10 A_2 per week for HAC with Krypton-85 and A_2 per week for HAC without Krypton-85, in compliance with 10 CFR 71.51(a)(2).

The staff agreed that the containment criteria and its allowable leakage rate for NCT satisfies the containment requirements for both normal and accident conditions and demonstrates that the performance of the containment boundary is met under NCT and HAC.

Table 4-4 Permissible Leakage Rates of TN-40 under NCT and HAC

Case	Effective A_2 (Ci)	Allowable Release Rate	Allowable Release Rate (C_i/sec)	Concentration C_i (C_i/cm^3)	Permissible Leakage Rate (cm^3/sec)	Permissible Standard Leakage Rate (ref-cm^3/sec)
NCT	56.6	10^{-6} A_2/hr	1.57×10^{-8}	1.05×10^{-4}	1.50×10^{-4}	1.00×10^{-4}
HAC	8.8	A_2/week	1.45×10^{-5}	3.52×10^{-4}	4.11×10^{-2}	1.27×10^{-2}
HAC (Krypton-85)	270	10 A_2/week	4.46×10^{-3}	3.03×10^{-3}	1.48	See Note

Note: The HAC with Krypton-85 (10 A_2 per week) is bounded by HAC without Kr-85 (A_2 per week). There is no need to calculate this value.

The applicant will use helium for fabrication, maintenance, periodic, and pre-shipment leak rate tests. The following is a summary of the allowable leakage rates in section 4.4 of the SAR:

- The user will perform the helium leakage rate test with a resulting acceptance criterion of 1.0×10^{-4} ref-cm^3/sec and a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less to meet the intent of the fabrication, maintenance, periodic, and pre-shipment leak rate tests in accordance with ANSI N14.5;

- Within 12 months of shipment, the entire containment boundary (using a test envelope) of the package must be leak-tested in order to ensure the containment requirements in Part 71 are met in accordance with ANSI N14.5.

The staff confirmed that the allowable leak rate of 1.0×10^{-4} ref-cm^3/sec for fabrication, maintenance, periodic, and pre-shipment leak tests of TN-40 meets the requirements of 10 CFR 71.51(a)(1) and 71.51(a)(2).

4.4 Evaluation Findings

Based on the containment evaluation of the TN-40 transportation package, the staff concluded that the containment design of the TN-40 package has been adequately described, and evaluated and that the package design satisfies the containment requirements of 10 CFR Part 71 under NCT and HAC.

1. The staff has reviewed the description of the containment system and has reasonable assurance that the information provided satisfies the containment requirements of 10 CFR Part 71.

2. The staff has reviewed the calculations used to derive the leakage rates under NCT and has reasonable assurance that there shall be no release or dispersal of

radioactive contents, as demonstrated to a sensitivity of 10^{-6} A_2 per hour, in compliance with 10 CFR 71.51(a)(1).

3. The staff has reviewed the calculations used to derive the leakage rates under HAC and has reasonable assurance that there shall be no release or dispersal of radioactive contents, as demonstrated to a sensitivity of A_2 in one week, in compliance with 10 CFR 71.51(a)(2).

4. The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that under the tests specified in 10 CFR 71.71, the package satisfies the containment requirements for NCT.

5. The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that the containment of the package will not exceed the specified allowable short-term limits during HAC consistent with the tests specified in 10 CFR 71.73.

Limitations and Conditions in CoC

Specific limitations and conditions are specified for approval of this application and use of the TN-40 package. These limitations and features are included in the CoC as conditions of approval:

1. 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 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5.

2. Within 12 months prior to shipment, the user shall perform a leak rate test of the entire containment boundary, with an acceptance criterion of 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-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.

5.0 SHIELDING REVIEW

5.1 Description of the Shielding Design

5.1.1 Packaging Design Features

Shielding for the TN-40 package is provided mainly by the cask body. The cask body is made up of the containment vessel, the gamma shielding, and the lid. For neutron shielding, a borated polyester resin compound surrounds the gamma shield shell radially. The gamma shield is provided around the inner shell and the bottom inner plate of the containment vessel, by an independent shell and bottom plate of carbon steel.

The 8-in. thick gamma shield shell and the 8.75 in. thick bottom shell are SA-105, SA-516, Grade 70, or SA-266 Class 4 material. A 6.0-in. thick shield plate (SA-105 or SA-516, Grade 70) is also welded to the inside of the outer plate lid.

Radial neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield shell. The resin compound is cast into long, slender, aluminum alloy containers. The total radial thickness of the resin and aluminum is 4.50 inches. The array of resin-filled containers is enclosed within a 0.50-in. thick outer steel shell (SA-516, Grade 55) constructed of two half cylinders.

The resin material is unsaturated polyester cross-linked with styrene, with approximately 50 wt% mineral and fiberglass reinforcement.

For transport, wood filled impact limiters are installed on either end of the cask and provide additional shielding for the package's axial ends and some radial shielding for the area at either end of the radial neutron shield.

5.1.2 Summary Table of Maximum Radiation Levels

Dose rates around the TN-40 package are determined by choosing a design-basis source and using it with a three dimensional Monte Carlo N Particle (MCNP) model. The selected source term is that which the applicant determined results in the highest dose rate at a 2 meter distance from the vehicle's side (assumed to be a 10-foot wide open railcar) for the proposed contents under NCT. This source term corresponds to a bounding assembly type with a burnup of 42,000 MWd/MTU, a minimum enrichment of 2.35 wt% Uranium-235 (U-235) and a 30-year decay time. The corresponding NCT dose rates are presented in Table 5-2 of the application. The Table 5-2 HAC dose rates are for fuel of the same burnup and enrichment but with a 24.4-year decay time. Staff finds the use of this source term acceptable since the resulting HAC dose rates show significant margins to the corresponding regulatory limit, which margins will be even greater for the design-basis source with the 30-year decay time. Staff reviewed the dose rates presented in Table 5-2 and finds there is reasonable assurance that the 2-meter side dose rate from the vehicle edge bounds the dose rates from the proposed TN-40 contents. The basis for this finding is described later in this chapter of the SER.

In general, a single source term may not yield the bounding dose rates for every location around the package. Differences in the shielding configuration at different locations on the package, such as around the trunnions and at the package axial ends, influence what type of source term is bounding for that area of the package. However, the location of the most limiting dose rates and the source term that results in bounding dose rates for that location can be determined and used to evaluate the proposed contents for acceptability. The applicant sought to do this for the TN-40 shielding evaluation, selecting the location 2 meters from the vehicle edge along the axial side of the package. Staff finds that it is not clear that this location is the location for the most limiting dose rates for the source terms represented by the burnup, enrichment, and cooling time parameter (BECT) combinations in Tables 5-8 and 5-9. For other packages, bounding dose rates at the axial ends have been higher than the side dose rates and establishment of a minimum separation distance between the package ends and the vehicle edge has been necessary in order to use the side dose rates to characterize the allowable contents with regard to radiation source term. This minimum separation distance was made a condition in the CoC for those packages. The current application does not propose such a minimum separation distance; therefore, the dose rate at 2 meters from the vehicle is taken to be at 2 meters from the package ends. However, for the current application, the applicant has proposed a condition that the minimum cooling time for all proposed BECTs be 30 years. Given this minimum cooling time and based upon the applicant's evaluations and staff's independent evaluations (described later in this chapter of the SER), which include comparisons with other BECTs at 30 years cooling time and considerations of the hardware Co-60 Curie content estimates and the various uncertainties, staff finds there is reasonable assurance that the selected BECT (42,000 MWd/MTU, 2.35 wt% U-235, 30 years cooling) is the bounding source term for the proposed contents and that the location at 2 meters from the vehicle edge along the package's axial side is appropriate for evaluating the acceptability of the proposed contents (i.e., that the bounding side dose rate is the most limiting dose rate for the proposed BECTs in Table 1-2 of the application).

A final comment is needed to clarify the correct application of surface dose rate limits for exclusive use shipments such as for the proposed package. The limits for package contact dose rates at the package's axial ends were initially listed in Table 5-2 as 1000 mrem/hr but corrected to 200 mrem/hr. This correction was made because in order for the initially cited value to be the correct limit, there would need to be some enclosure around the impact limiters, which is not consistent with the licensing drawings. 10 CFR 71.47(b)(1) states that for exclusive use shipments the dose rate limit on the external package surface is 200 mrem/hr unless three conditions exist. One of those conditions is that the shipment is made in a closed transport vehicle; the current case assumes an open railcar. Also, for compliance with 49 CFR 173.441(b), the 200 mrem/hr limit applies to the sides of any enclosure used with a package on a flatbed vehicle (open railcar) as expanded upon in DOT guidance in RAMREG-001-98, "Radioactive Material Regulations Review" (see Figure 9 of that document). The basic principle is that the dose rate limit is 200 mrem/hr for any part of the package or enclosure surface that is accessible. So whether the impact limiters are considered part of the package or part of the personnel barrier (i.e., the enclosure), the surface dose rate limit is 200 mrem/hr due to their accessibility. Staff also notes that the applicant does not assume a minimum separation distance between the ends of the impact limiters and the vehicle edge. Thus, the impact limiters could be at the vehicle edge and therefore the 200 mrem/hr limit would apply because of this configuration. Staff notes that this difference does not affect the ability of the package to meet the 10 CFR Part 71 shielding requirements.

5.2 Radiation Source

The proposed contents for the package are limited to 40 unconsolidated 14x14 PWR fuel assemblies with Zircaloy cladding that have been irradiated at PINGP and are not damaged. According to the applicant, the fuel may be transported with or without fuel inserts (namely, BPRAs and TPAs). The specifications for the proposed fuel assembly types and inserts are presented in Section 1.2.3 of the application. Appropriate specifications are also captured in the CoC. The assemblies' hardware includes inconel grid spacers. Table 5-4 indicates that the guide and instrument tubes are modeled as stainless steel; however, these items are actually Zircaloy. They are modeled as stainless steel in order to capture the contribution to the source term from that portion of the BPRA that is located within the active fuel and the plenum. The assemblies may also have natural uranium axial blankets 6 and 6.2 in. long.

The staff evaluated the applicant's method of estimating the BPRA in-core gamma source and finds that it will significantly underestimate the amount of Cobalt-60 (Co-60) from the BPRAs. The proposed contents include BPRAs that have a maximum equivalent burnup of 30,000 MWd/MTU and a minimum of 25 years cooling time (assuming a Cobalt impurity level of 1500 ppm in steel). There is no variation in either of these parameters. However, the estimated activation from the various BECTs will result in variable predictions of Co-60 amounts. With a minimum cooling time of 30 years and the uncertainty in the use of a Cobalt impurity in steel of

800 ppm, the BPRA source may be under-predicted by several tens of Curies. The amount is dependent upon the burnup. For the dose rates at 2 meters from the vehicle edge, using the response function indicates that the dose rate may be under-predicted by about 0.5 mrem/hr. There is also some uncertainty added to the calculated dose rates at other areas of the package. Thus, in general, the staff finds that this method of estimating the BPRA source can be non-conservative and under-predict the Cobalt source; however, given the margins (evaluated later in this SER chapter) that exist for the current application, the staff finds this method to be sufficient for the current application.

The applicant used the TPA source in the assembly top end and gas plenum region and not the BPRA source to account for the non-fuel hardware contribution in these areas of the fuel assemblies. The hardware characteristics of BPRAs and TPAs are similar in these areas. Additionally, the TPA source is based upon a much higher burnup than is the BPRA source. Based upon these considerations, and the fact that both inserts cannot be simultaneously located in the same assembly, staff finds representing the non-fuel hardware source in these areas with the TPA source to be acceptable.

The SAS2H/ORIGEN-S modules of the SCALE code were used to generate gamma and neutron source terms for the bounding assembly, the Westinghouse 14x14 Standard assembly. This assembly was selected as bounding because it contains the greatest fuel mass, which staff finds to be consistent with the shielding analyses for other packages. Source terms were generated for minimum initial enrichments ranging from 2.00 wt% to 3.85 wt% U-235. The fuel was irradiated for a constant time of 400 effective full power days per cycle. Burnup values range from 17,000 MWd/MTU to 45,000 MWd/MTU using a specific radiation power between about 15 and 25 MW/assembly. Calculations used an operating cycle history with a 30-day down time between cycles. Details of the analysis are given in Section 5.2 of the application.

In evaluating the applicant's source term determination, staff noted that the calculation was performed with the default minimum number of libraries per cycle. Using the default value for this parameter is not sufficient, as this number of libraries may not be sufficient to allow for convergence of the source term calculation. Staff analyses indicate that the minimum number of libraries needed for source convergence is about 6 to 8 per cycle, though some portions of the source spectrum may require additional libraries to achieve convergence. Based upon its evaluations, staff has reasonable assurance that the fuel's gamma source term for the currently proposed contents for this package will be overall conservatively estimated with the default number of libraries. In the cases of contents with lower burnup values, the neutron source may be under-predicted, but the effect is expected to be insignificant considering that for those burnup values the neutron source contributes only to a minor extent to the dose rate. Staff evaluations indicated that the Curies of Co-60 per gram of Cobalt may be under-predicted, particularly for the lower burnup values. Yet, due to the large margins associated with the minimum 30-year cooling time and the relatively small degree of under-prediction, staff finds use of the default parameter (i.e., number of libraries per cycle) to be acceptable for the current application.

Models of the assemblies assumed uniform fuel enrichment. In other words, no source calculations were performed with the natural uranium blankets explicitly modeled. The source terms were determined for minimum assembly average enrichments and maximum assembly average burnup values. The calculation of these enrichment and burnup values included the axial blankets. Also, the applicant used a bounding burnup profile in the dose rate calculations that is based upon burnup profiles of Prairie Island spent fuel assemblies, including assemblies with and without blankets. Based upon these descriptions and staff's independent evaluation of source term calculations with axial blankets explicitly modeled versus homogenized with the rest of the fuel, the staff finds analyses with uniform enrichments acceptable for the length of axial blankets described in the application, as supplemented.

A final note on the design-basis source term is also needed. For purposes of the applicant's evaluation method, the applicant used the BECT of 42,000 MWd/MTU, 2.35 wt% U-235 and 24.4 years cooling. The applicant determined this BECT bounds the other BECTs in Tables 5-8 and 5-9. The bounding BECT for the proposed contents (Table 1-2 of the application) has the same burnup and enrichment but with a minimum cooling time of 30 years. Thus, discussion of the design-basis source (term) or BECT for the evaluation described in the following sections refers to the BECT with 24.4 years cooling time except when the text explicitly refers to the bounding or design-basis BECT for the proposed contents, which is the 30-year cooled BECT.

5.2.1 Gamma Source

Table 5-6 describes the design-basis gamma source term for each assembly axial zone (e.g., active fuel, plenum, etc.). This gamma source includes the contribution from a TPA at a maximum equivalent burnup of 125,000 MWd/MTU cooled for a minimum of 13 years (assuming a Cobalt impurity level of 800 ppm in steel) that is also shown separately in Table 5-7. The proposed limits for TPAs set the minimum cooling time to 25 years, which staff finds acceptable to account for higher Cobalt impurity levels. The types of proposed TPA contents are those without water displacement rods extending into the active fuel; thus, they only contribute to the source in the assembly top end fitting and gas plenum zones.

The applicant presented the gamma source spectrum in the 18-group structure consistent with the SCALE 27n-18γ cross section library. The conversion of the source spectrum from the default ORIGEN-S energy grouping to the SCALE 27n-18γ energy grouping was performed directly through the ORIGEN-S code.

The gamma source from the fuel assembly hardware was primarily from the activation of Cobalt. This activation contributes primarily to SCALE Energy groups 36 and 37, with an energy range of 1.00~1.66 MeV. The applicant assumed a Cobalt impurity level of 4.69 g/kg (4,690 ppm) in inconel and 0.80 g/kg (800 ppm) in steel (see Table 5-5). The selected impurity level was the highest identified level for inconel that the applicant found. Staff finds this impurity level in inconel to be acceptable and notes that it is consistent with levels assumed in other approved applications. Staff notes that some literature indicates that steel components of assemblies of a similar vintage indicate that Cobalt levels have been found to be as high as 2200 ppm.

The applicant performed an evaluation to compare the total predicted Curies of Cobalt using their assumed impurity levels versus those determined from measurements of hardware from fuel assembly types that are similar to the proposed package contents. These measurements showed a reduced level of Cobalt in Inconel as well as an increased level of Cobalt in steel (1500 ppm). The comparison indicated that the applicant's assumed impurity levels were conservative versus the measured levels. Staff notes that this argument applies only to the dose rates along the package side.

The assembly end hardware and fuel insert end materials are predominantly steel; thus, the package end dose rates would be significantly affected by the assumed Cobalt impurity. The applicant therefore provided an evaluation of the differences in Cobalt level estimates for the assembly hardware zones and found that the initial impurity level may under-estimate the amount of Cobalt by up to nearly half the amount determined with the 1500 ppm impurity level. However, the applicant relies upon the proposed contents' minimum decay time of 30 years (a minimum of about one half-life greater than for the longest cooled BECT in Table 5-9, which is the design-basis BECT) as an offset. Based upon these considerations, the package end dose rates are expected to be similar to those predicted for the design-basis BECT with the originally assumed Cobalt impurity. The resulting NCT package end dose rates from this design-basis BECT are given in Table 5-2; the NCT package side dose rates in Table 5-2 are from the bounding BECT of the proposed contents. Based upon these arguments and the minimum cooling times for the proposed contents (see Section 1.2.3 of the application), staff finds the assumed Cobalt levels in steel to be acceptable. Staff also finds the use of the package end dose rates from the evaluation's design-basis BECT to estimate the NCT package end dose rates for the proposed contents' design-basis BECT to be acceptable.

Staff reviewed the scaling factors used to account for spatial and spectral variations of the neutron flux outside the active fuel zone and finds them acceptable. The applicant also accounts for the (n,γ) interactions.

5.2.2 Neutron Source

The total neutron source for the design-basis BECT is shown in Table 5-6 of the application. The neutron source was comprised mainly of Curium-244 (Cm-244). Therefore, in order to perform the MCNP dose rate analyses, the Cm-244 energy spectrum was used to represent the neutron spectrum for the spent fuel contents. Thus, SAS2H/ORIGEN-S was used only to calculate the total neutron source strength for each evaluated BECT. Subcritical multiplication is also conservatively accounted for by assuming a fresh fuel composition at 3.00 wt% enrichment. The staff finds this to be acceptable.

The neutron source is not linearly dependent with burnup, and therefore calculations were performed to determine the axial neutron source distribution. The resulting axial neutron peaking factors are shown in Table 5-12 of the application.

5.3 Shielding Model

The TN-40 cask is designed to provide both gamma and neutron shielding. Two base models were constructed. The first model corresponds to the neutron transport problem and the second corresponds to the gamma transport problem. The Monte Carlo computer code MCNP, Version 4C2, was used for calculating the gamma and neutron doses in this analysis. A more recent version of MCNP (MCNP5 1.40) was used to perform the calculations with models addressing shielding tolerances. MCNP is a well established code for performing multi-dimensional transport calculations suitable for complex geometries such as spent fuel transportation casks. The staff finds this code, and these versions of the code, to be acceptable for use in this application.

The model specifications for shielding for both NCT and HAC are presented in Section 5.3 of the application. A description of the shielding configuration is presented in Section 5.3.1 of the application. The major difference between the NCT and HAC models is the assumed loss of the impact limiters and neutron shielding in the HAC model. Based upon a review of the licensing drawings and the evaluations of the NCT and HAC tests (see Chapters 2 and 3 of this SER), the staff finds the shielding models to be acceptable. The HAC models assume the assemblies retain their geometric configuration (i.e., they don't reconfigure). Given the significant margin to the dose rate limits for HAC and independent shielding evaluations of damaged fuel as well as the structural evaluations (see Chapter 2 of this SER) for the proposed contents, the staff finds this assumption to be acceptable.

As explained in the application, allowable contents specifications were determined using a response function method. MCNP was used to develop the response functions and to perform dose rate calculations for the design-basis sources (i.e., the design-basis source in the evaluation and the proposed contents' design-basis source). The models for these two calculations use the previously stated base models; however, the model geometry details differ between the models for the response function and the design-basis dose rate calculations. While the dose rate calculation models the package, including tolerances on the package side, as described in the licensing drawings, the response function calculation models the neutron shield uniformly at its greatest nominal thickness, including at the cutout regions around the trunnions. Also, the response function models use nominal dimensions and neglect the axial lids. Other than the use of nominal dimensions, the model differences should not significantly affect the response functions, developed for the location at 2 meters from the vehicle edge at the package's axial mid-plane. Based upon its review of the descriptions of the models and the development of the response functions and the evaluation uncertainties and margins, the staff finds the modeling method to be acceptable.

5.3.1 Configuration of Source and Shielding

The radiation source is divided into four axial zones. The bottom zone represents the lower end fittings, the middle zone the active fuel region and the upper zones represent the plenum and upper end fittings of the fuel assembly. The fuel, end fittings and plenum are homogenized within each assembly envelope and the axial length of their respective zones. The axial peaking factors described in Table 5-12 of the application and in Section 5.2.2 of this SER were applied to the active fuel region for gammas and for neutrons for their respective dose rate calculations.

The TN-40 package model is described in Section 5.3 and illustrated in Figures 5-3 through 5-6 of the application, with sample inputs included in Section 5.7. Sections 5.1 and 5.4 contain additional details regarding the tolerances (dimensional and material) and trunnions and surrounding area in the models for the design-basis dose rate calculation. The staff reviewed the model descriptions, including dimensional tolerances, and finds them to be consistent with the cask design as described in the licensing drawings. As already explained, the response function models differ with regard to these features.

As previously described, the neutron shield is composed of a borated polyester resin that is poured into aluminum containers. The walls of these containers are 0.12 in. thick. The applicant modeled the neutron shield as a homogeneous mixture of the resin and the aluminum containers. The applicant justified this approach by stating that measurements for TN-40 storage casks, as well as other TN casks with similar neutron shield designs, have not indicated any streaming effects. The applicant also argued that the neutrons will generally scatter and not travel a direct path through the aluminum for the full shield thickness. Staff considered the applicant's justification and the shield geometry. The containers are nominally 4.5 in. thick and 6 in. wide. The potential streaming path between sections of the resin compound is 0.25 in., based upon the aluminum wall thickness. Thus the potential streaming path is very narrow and the separation between neighboring paths is relatively small. Therefore, staff considers that any streaming would impact any measurement taken around the radial surface of the neutron shield so that the radiation streaming would not be detectable. Additionally, there is significant margin between the calculated 'package contact dose rates' and the applicable regulatory limit so that even if dose rates were tripled due to streaming, the regulatory limit would still be met. Thus, the staff finds the homogenization of the resin and aluminum tube materials to be acceptable.

5.3.2 Material Properties

As described previously, the package materials include stainless steel and aluminum for the basket, carbon steel for the gamma shielding, a borated polyester resin compound in aluminum containers for the neutron shielding, and stainless steel encased redwood and balsa wood for the impact limiters. Mass densities were used for these materials in MCNP with the materials' constituent fractions being specified in atom fractions or mass fractions. For those components that were homogenized in the model, such as the polyester resin and the aluminum tubes, the material density and constituent fractions reflect this homogenization. Staff reviewed the materials specifications input into MCNP and found them to be acceptable.

Section 8.1.5 of the application provides additional information regarding the neutron shielding material. In particular, the acceptable minimum density and ranges of weight percentages of the hydrogen and boron constituents are given. According to Section 8.1.5, density testing will be performed on every mixed batch of resin, and chemical analyses will be performed on the first batch with a given set of components as well as whenever a new lot of a major component is introduced. Staff finds this process acceptable to ensure proper composition of the neutron shield material mixture (see also Chapter 2 of this SER).

Staff notes that the shielding models use the nominal hydrogen and boron concentrations listed in Section 8.1.5. To demonstrate compliance with regulatory dose rate limits, the shielding model needs to account for the tolerances on these material specifications. The applicant justified using the nominal concentrations (or weight fractions) based upon the following:

1. for the hydrogen component, actual measurements of the packages' (which have already been fabricated) neutron shield constituents indicate that the weight fraction of hydrogen, accounting for measurement tolerance, is higher than the nominal specification (5.21±0.14 vs. 5.05 wt%), and

2. the nominal boron weight fraction is used because the weight fraction for which the boron effect was found to saturate is lower than the minimum acceptable amount (0.75 vs. 1.05 wt% ± 20%).

Based upon the foregoing statements, the staff finds the neutron shield material specifications used in the analysis models to be acceptable for the number of casks currently proposed to be covered by this CoC, which have been fabricated and are in use at PINGP. Since the acceptance criteria (Section 8.1.5 of the application) allow use of a neutron shield with less than the currently analyzed hydrogen content, the analysis models for any additional casks that the applicant may seek to include in the CoC at a future date need to use the minimum acceptable hydrogen content in the neutron shield, including the tolerance.

5.4 Shielding Evaluation

The applicant performed analyses with the SAS2H/ORIGEN-S modules of the SCALE code system, Version 4.4; MCNP, Version 4C2; and MCNP5 1.40. SAS2H/ORIGEN-S were used to generate the gamma and neutron source terms for the proposed contents' design-basis assembly type at the several burnup, cooling time, and minimum enrichment combinations described in the application. Based upon the descriptions of the fuel assemblies and the assumptions regarding features such as axial blankets and the limits on burnup, staff finds the use of this version of the SCALE code modules to be acceptable. As stated previously, staff finds the selected versions of MCNP acceptable for determining the dose rates for the proposed contents and package design. MCNP calculations were performed with the flux-to-dose rate conversion factors given in ANSI/ANS 6.1.1-1977, in accordance with staff guidance.

The applicant used MCNP to calculate the dose rates for the design-basis assembly at a minimum enrichment of 2.35 wt% U-235, a maximum burnup of 42,000 MWd/MTU, and minimum cooling times of 24.4 and 30 years. The BECT with 30 years of cooling time was found to result in the bounding dose rate at a location 2 meters from the vehicle's edge (assumed to be 10 feet wide) at the cask's axial mid-plane for the proposed contents described in Section 1.2.3 of the SAR. The NCT dose rates for these contents are given in Table 5-2. Additionally, Tables 5-18 and 5-19 give the axial dose rate profile for these BECTs. The calculation uncertainties were generally less than 5% for the majority of the dose rates; the uncertainties for the HAC end neutron dose rates were approximately 10%. The HAC dose rates in Table 5-2 were calculated for the 24.4-year minimum cooling time.

The applicant also provided dose rates at the ends of a railcar (vehicle) of varying lengths to show the package can comply with the limits for any normally occupied space to avoid the need by the carrier personnel to wear radiation dosimetry (see 10 CFR 71.47(b)(4)) for exclusive use shipments. However, no information is presented regarding the assumed positioning of the package on the vehicle. Yet, staff notes that in practice the package would be expected to be positioned on a rail car such that the car's axles bear equal proportions of the total load. Given the minimum rail car length is 40 feet and the length of the package, there would be a minimum of 9 feet between the end of the package and the rail car's edge. Additionally, there will be a buffer car placed between each end of the package's rail car and the next car in compliance with 49 CFR 174.85(b). Based upon these factors and the dose rates determined at the package's axial ends (see Table 5-2), staff finds reasonable assurance that the dose rates in any normally occupied space, for rail shipment, will not exceed 2 mrem/hr.

5.5 Response Function Method

MCNP provides information on the dose rate per source particle per second. This value is then multiplied by the source strength to obtain the dose rate for a given source. In the place of using MCNP to calculate dose rates for each BECT combination, the applicant used the dose rate per source particle per second data to create response functions for calculating dose rates at 2 meters from the vertical edge of the vehicle along the package's side. The various BECTs' source strengths (calculated using SAS2H/ORIGEN-S) were multiplied by the response functions to obtain their resulting dose rates. The MCNP calculation was run to ensure that the response functions are converged and can be used to estimate the dose rates for the different BECT source terms. Table 5-20 lists the response functions. For the primary gamma source, a separate response function was used for each energy bin of the gamma spectrum. The response functions cover gamma energies from 0.4 to 4.0 MeV. The applicant notes that gammas in this energy range account for over 99% of the gamma contribution to the dose rate. This energy range is greater than that stated in staff guidance to significantly contribute to external radiation dose and is therefore acceptable. These response functions include the assembly hardware regions' contributions. Since the Cm-244 spectrum was used to represent the neutron source, a single response function for the total neutron source is used to calculate neutron dose rates. The same is also true for (n,γ)-gamma dose rates. The response functions include an axial burnup normalization constant, derived from the bounding axial profile. These constants are listed in Table 5-12 and described in Section 5.2.4. The response functions were also increased by 5% to account for analytical uncertainties.

For the response function method, the applicant set the acceptance criteria at a dose rate limit of 9.8 mrem/hr and a decay heat limit of 525 W per assembly. The allowable contents would be defined by those BECT combinations that yielded results meeting these criteria. The BECT combination of 2.35 wt% enrichment, 42,000 MWd/MTU, and 24.4 years cooling time was determined to be bounding at the stated location though dose rates from many of the other BECT combinations resulted in dose rates that were within 0.1 mrem/hr of the design-basis value. The BECT combinations that meet these criteria are given in Table 5-9. Tables 5-10 and 5-11 show the resulting dose rates and cask decay heats. As can be seen from these tables, the dose rate criterion is limiting for many BECT combinations while the decay heat criterion is limiting for others. The cooling times in Table 5-9 were rounded up to the next full year to create the fuel qualification table (Table 5-8), with an absolute minimum cooling time being 15 years. The allowable BECT combinations calculated by the response function account for non-fuel hardware in the active fuel zone using the method previously described to represent the BPRA source. However, they do not account for the contribution from non-fuel hardware in the top end fitting and the plenum zone; the TPA source term is used for the non-fuel hardware source in these zones and was not included in the calculations.

Staff noticed that the dose rates for many BECTs in Table 5-10 were shown to be at the regulatory limit (10 mrem/hr), which is not the acceptance criterion of 9.8 mrem/hr. The applicant explained that this shows that these BECTs resulted in about the same dose rate. It also stated that the fuel qualification table was developed so that any BECT resulting in "an estimated dose rate of 10 mrem/hr (rounded up) can be used in the shielding calculations" but that the bounding source is the one already identified as such. Staff noticed that the source term listed in Table 5-6 resulted in a dose rate of 9.91 mrem/hr using the response functions given in Table 5-20. This source term includes the TPA contribution. Thus, the impact of non-fuel hardware in the top end fitting and plenum zones can be deduced. Removing the TPA contribution results in a dose rate of about 9.7 mrem/hr, meeting the applicant's stated 9.8 mrem/hr criterion. Staff notes, however, that neglect of the non-fuel hardware contribution in these zones of the assembly introduces uncertainty into the response function method.

As explained previously, the allowable contents are determined based upon a dose rate criterion at one location and a bounding BECT combination is determined from the dose rates at this single location. Staff notes that a single BECT combination does not necessarily result in bounding dose rates at every location around the cask. In its review of other approved packages, staff also found that the bounding dose rates at the package ends can be the more limiting dose rates unless conditions are assumed regarding the position of the package on the vehicle. Such conditions are then incorporated into the CoC. In the case of the current application, however, the proposed contents BECTs were modified so that the absolute minimum cooling time for all proposed contents was set to 30 years, instead of 15 years. Therefore, the minimum cooling time for all proposed contents is now 30 years. For this cooling time, the bounding burnup and enrichment combination is still 42,000 MWd/MTU and 2.35 wt% U-235 for dose rates at the 2-meter side location. Also, staff evaluations indicate that this BECT combination results in the greatest amount of Co-60 in the assembly hardware for the proposed contents BECTs (Section 1.2.3 of the application). Therefore, given that the location of the highest dose rate for this BECT is at the 2-meter side location, staff finds reasonable assurance that the selected location is the location of the highest bounding dose rate for the proposed contents. Also, staff finds reasonable assurance that contents that meet the dose rate limits at the selected location will meet the dose rate limits at all locations around the cask.

5.6 Uncertainties and Conservatisms

The applicant also considered the uncertainties and described the conservatisms included in the analysis. There are uncertainties associated with both the source term and dose rate calculations. The applicant stated that SAS2H/ORIGEN-S predictions of principal isotopes for neutron and gamma dose rates are within 10% of measured data. The MCNP calculation results have standard deviations less than 2%. Other sources of uncertainty include the package material and dimensional tolerances, modeling assumptions for the package ends and areas around the trunnions for the response function models, and other modeling assumptions used in representing features of the cask and the contents. The applicant also lists the difference in dose rates of 0.1 mrem/hr between the results for many BECTs in the response function method as another uncertainty. Additionally, uncertainties in the neutron dose rates for HAC were noted to be about 10%, and uncertainties were also large for neutron dose rates at the top and bottom impact limiters.

Several conservatisms were included in the analytical method. Source terms were calculated based upon conservative fuel parameters. Shielding models used a bounding burnup profile that is based on the profiles of assemblies with and without axial blankets. The models also used a neutron shield with a hydrogen content less than was measured in the fabricated casks' neutron shields. The design-basis, or bounding, dose rate calculation included the source terms from both TPAs and BPRAs (in the active fuel zone only) in the fuel assembly. This calculation also included package tolerances for radial shielding features. Though nominal dimensions and simplifying assumptions were used in the response function models, the response function values were increased by 5%. Staff notes, however, that the response function under-predicts the dose rate compared to the MCNP design-basis calculation by roughly 3%. Also, staff views inclusion of tolerances as a bounding approach to shielding evaluations and not as contributing conservatism whereas using nominal dimensions adds uncertainty since a package could be manufactured to those tolerances. Also, the minimum cooling times for the several BECTs were rounded up to the next full year. Additionally, the minimum cooling time for all proposed contents was set to 30 years (see Table 1-2 of the application). Based upon the differences in dose rates in Tables 5-18 and 5-19, this adds significant margin. Significant margins also exist for the dose rates at the top and bottom impact limiters and for HAC conditions. The applicant relies upon these and other conservatisms to account for uncertainties in the evaluation method.

Staff also performed an independent semi-quantitative evaluation of the conservatisms and uncertainties using the results of the MCNP calculated dose rates for the bounding BECT of the proposed contents from Table 1-2. Staff's evaluation included the dose rates at 2 meters from the vehicle along the package side and at 2 meters from the package's axial ends. Using the dose rates from this BECT would include all the conservatisms claimed by the applicant. Staff then included the effects of the uncertainties due to the calculation methods. These include the uncertainties and standard deviations in the SAS2H and MCNP results. Uncertainties arising from various evaluation assumptions, such as the representation of the in-core BPRA source with steel guide and instrument tubes and the assumed Cobalt impurity level in steel, were also included. For the Cobalt impurity level, staff also considered the maximum measured amount for assembly hardware steel across a range of assembly types (per publicly available data of which staff is aware). For end dose rates, the neglect of tolerances in the package model was also considered. These evaluations indicate that the margin is sufficient to cover the uncertainties. Based upon the foregoing, the staff finds reasonable assurance that evaluation uncertainties have been adequately considered and that sufficient conservatisms and margins exist to address these uncertainties such that the package dose rates, for the proposed contents as described in Section 1.2.3 of the application and the CoC, meet the radiation limits in 10 CFR Part 71.

5.7 Staff Evaluation of Method

The staff reviewed the applicant's analysis method and the results and finds, given the stated conservatisms and the bounding BECT for the proposed contents (in Table 1-2 of the application and included in the CoC), the applicant's evaluation to be acceptable. The staff finds that the applicant's evaluation provides a reasonable demonstration that the package will meet the external radiation requirements of 10 CFR Part 71 for exclusive-use transport. Using the information provided in the application, the staff performed confirmatory shielding analyses with the SCALE 5 code system, some of which have already been described.

The staff's analysis also included calculating estimated dose rates with the applicant's response function with source terms generated using SAS2H/ORIGEN-S. Source terms were generated for various BECTs from Tables 5-8 and 5-9. Staff's calculation showed relatively good agreement with the source term the applicant used as the bounding source in its analysis and development of the analysis method. Comparison of the dose rates for selected BECTs gave some indication that some BECTs in Table 5-9 resulted in dose rates up to around 0.5 mrem/hr higher than the applicant's bounding BECT. Staff notes, however, that minimum decay times were modified to an absolute minimum of 15 years with others being rounded up to the next full year for the fuel qualification table in Table 5-8. Thus, staff evaluated various BECTs from Table 5-8. Comparison of the dose rates from these BECTs indicated that there may still be some BECTs that yield dose rates that are higher than the applicant's bounding BECT; however, the staff found the difference to be on the order of 0.2 mrem/hr, which is only slightly larger than the variation noted by the applicant in its discussion of evaluation uncertainties. Based upon these results for Table 5-8 BECTs, the staff finds that the applicant's selected BECT provides a reasonably bounding package dose rate at 2 meters from the vehicle edge for contents with the BECTs in Table 5-8. Staff notes, though, that the MCNP calculations (Table 5-18) indicate that this BECT can't meet the dose rate limits at 2 meters from the vehicle edge. However, the proposed contents are limited to an absolute minimum cooling time of 30 years. As stated previously, the calculated dose rates for the bounding BECT at this 30-year cooling time, accounting for evaluation uncertainties, bound the dose rates for the other proposed contents and meet the dose rate limits.

Based upon reviews of other spent fuel transportation packages, staff is aware that, considering the BECTs in Tables 5-8 and 5-9, BECTs other than the evaluation design-basis BECT may result in bounding dose rates at 2 meters from the package ends and, given the assumptions about the package configuration in this application, these dose rates may be higher than those at 2 meters from the vehicle edge along the package side. However, staff finds reasonable assurance that the required minimum cooling time of 30 years (application Table 1-2, included in the CoC) ensures that the selected BECT's 2-meter side dose rate is bounding for the proposed TN-40 contents and that the bounding dose rate at this location will be the limiting dose rate (i.e., the dose rate closest to the regulatory limits) for the proposed contents. Thus, the staff finds there is reasonable assurance that the package design with the proposed contents can meet the dose rate limits for transportation.

5.8 Evaluation Findings

Based on its review of the statements and representations in the application and independent evaluations, the staff concludes that the design has been adequately described and evaluated and finds reasonable assurance that the package meets the shielding performance requirements of 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

The objective of this review is to verify that the TN-40 package design satisfies the criticality safety requirements of 10 CFR Part 71 under both NCT and HAC. The staff reviewed the description of the package design and criticality safety analyses presented in Chapters 1 and 6 of the Safety Analysis Report for the TN-40 transportation package and supplemental information provided by the applicant, including the applicant's responses to the Requests for Additional Information, proprietary calculation packages, and conference calls. The staff performed its review following the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." The staff's criticality safety evaluation for this package design is provided in the following sections of this report. The Safety Evaluation Report that involves proprietary information is provided in a proprietary version of this SER.

6.1 Description of Criticality Design

TN-40 is a burnup credit spent fuel transportation package. The packaging of the TN-40 system consists of a fuel basket and an overpack cask. The fuel baskets are made of stainless steel and aluminum alloy. Boral neutron poison plates are enclosed in fuel compartment walls. An area density of 10 mgrams of B-10 is used for the poison plates. Concentric cylinder shells are used to form the containment boundary and gamma shield. Outside the outer steel shell is the neutron shielding layer. Borated resin is used to provide shielding to neutron radiation. License drawings 10421-71-3, 10421-71-6, 10421-71-8, and 10421-71-9 show the structure design and the geometric dimensions of the TN-40 package that are important to criticality safety.

The package will be used exclusively for transporting up to 40 undamaged 14x14 class PWR spent nuclear fuels discharged from PINGP with or without Non-Fuel Assembly Hardware (NFAH). The TN-40 Prairie Island spent fuel packages are burnup-credit transportation packages. Burnup credits are taken for twenty seven (27) isotopes, including 12 actinides and 15 fission products. The bounding fuel parameters are 3.85% U-235 initial fuel enrichment, 31 GWd/MTU burnup, and 30 years of minimal cooling time for the purpose of criticality safety analysis.

The applicant's criticality safety analysis results demonstrated that a single package under NCT and HAC as well as an infinite array of undamaged or damaged packages remains subcritical. The applicant calculated the Criticality Safety Index (CSI) following the method described in 10 CFR 71.59. The CSI value is 0 for the TN-40 packages.

6.2 Spent Nuclear Fuel Contents

The TN-40 packaging is designed to load up to 40 intact 14x14 class PWR spent fuel assemblies with or without Non-Fuel Assembly Hardware (NFAH) or other spent fuel assemblies with equivalent geometry and neutronic characteristics. However, only fuels with BPRAs are evaluated in depletion analyses because BPRAs envelop all NFAHs as far as criticality safety calculation is concerned. The following tables list the authorized payloads for the TN-40 packages.

Parameters of Prairie Island PWR Assemblies for Shipment

Fuel Assembly Parameters

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

Non-Fuel Assembly Hardware Parameters

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

Fuel assemblies with control rod insertion during depletion are also allowable contents of the packages. Fuel assemblies discharged from Unit 1, with ID D-01 to D-40 are not authorized contents. The fuel assemblies with Integral Fuel Burnable Absorbers (IFBAs) are not authorized contents because evaluations of the criticality safety for packages with such types of fuel assemblies were not performed and the applicant states on page 6-8 of the SAR that none of the eligible spent fuel assemblies contains IFBA.

6.3 General Considerations for Criticality Evaluations

The packages are evaluated to verify that it satisfies the criticality safety requirements of 10 CFR Part 71, Subpart E [10 CFR 71.55, 71.59, 10 CFR 71.71, and 10 CFR 71.73]. The package designs are evaluated for criticality safety under NCT and HAC pursuant to the requirements of 10 CFR 71.55, 71.71, and 71.73. The packages are evaluated for criticality safety for arrays of undamaged and damaged packages in accordance to the requirements of 10 CFR 71.59.

Because TN-40 is a fissile materials transportation package, the criticality safety analysis is required to demonstrate the maximum reactivity of the package under NCT and HAC. The staff reviewed and evaluated the applicant's criticality safety analyses. The staff's review includes the evaluation of the methodologies, simplifications, assumptions, and the computer codes and computer models used in the criticality analyses. The adequacy of the benchmarking of the computer codes used in the criticality safety analyses is an integral part of the criticality safety review and evaluation.

The SAS2H module of the SCALE-4.4 computer code system is employed in the fuel assembly depletion analyses. The CSAS25 module of the SCALE-4.4 computer code system is used to determine the k_{eff} of the cask with the bounding parameters.

As TN-40 is a burnup credit package design, staff assesses the adequacy of the methodology used for burnup credit analyses as well as the correctness and adequacy of the parameters used in the fuel assembly depletion models. Benchmarking of the fuel depletion code and package criticality safety analysis code are the focuses of the review.

The criticality safety analysis of a burnup credit spent fuel transportation package consists of two major parts. The first part is to accurately determine the isotopic inventories of the spent nuclear fuel to be transported. The second part is to determine the k_{eff} of the cask loaded with the spent nuclear fuels. Since the isotopic inventory in a spent fuel assembly must be calculated with a fuel assembly depletion model and the package k_{eff} must be calculated with a criticality analysis model, the computer codes and models for these analyses must be benchmarked to experimental data to ensure the accuracy of these criticality safety analyses, identify the possible bias and uncertainties associated with these codes and models, and make adequate adjustments to the results.

The applicant performed benchmark analyses for the fuel depletion code SAS2H using the publicly available chemical assay data for spent fuels from a variety of publications. A combination of the "Correction Factor" method and "Direct Difference" method were used for determining the bias and uncertainties of the fuel assembly depletion code and model. The applicant performed benchmark for the criticality safety analyses code CSAS25 together with the selected nuclear cross section library using a combination of the critical experiments published in the International Handbook of Evaluated Criticality Safety Benchmark Experiments and Commercial Reactor Criticals. The discussion on burnup credit, including the methodologies and results of the burnup analyses, is presented in more detail in Section 6.8 of this SER.

One of the important control parameters for fuel qualification is the cooling time. Based on the study performed by Oak Ridge National Laboratory and published in NUREG/CR-6781, "Recommendations on the credit for cooling time in PWR burnup credit analyses," the fuel reactivity initially decreases as cooling time increases. However, at about 100 years cooling time, the reactivity of the spent fuels starts to increase. The reactivity of the cask continues to increase thereafter until it reaches a new plateau. At about 200 years of cooling time, the reactivity of the cask will reach approximately the same value as that of the fuel with 30 years of cooling time. For this reason, criticality safety must be reevaluated if the transportation of the TN-40 packages has not been completed when the cooling time of the fuels in the casks reaches 200 years.

6.3.1 Description of Calculation Models

The SAS2H module of the SCALE-4.4 computer code system is employed in the fuel assembly depletion analyses. The CSAS25 module of the SCALE-4.4 computer code system is used to determine the k_{eff} of the cask with the bounding parameters. The analyses used the 44-group cross section library developed using the ENDF/B-V data.

The SAS2H is a one-dimensional fuel lattice depletion analysis code. The output of the SAS2H model is the isotopic composition of the spent fuel for given initial enrichment, burnup, and cooling time. Correct and bounding fuel depletion parameters were used in all fuel depletion models, including the depletion benchmark calculations, to ensure accurate and conservative safety analysis results. The results of the SAS2H calculations for benchmark fuel samples are compared with the measured spent fuel chemical assay data to determine the bias and uncertainties associated with the fuel depletion calculation. Appropriate adjustments were made to ensure that the isotopic concentration data obtained from SAS2H and used in the criticality calculation are accurate and conservative for isotopes for which burnup credits are taken.

The criticality analyses were performed using the CSAS25 module of the SCALE-4.4 code system using the 44-group cross section library that was developed using the ENDF/B-V data. A series of calculations were performed to determine the relative reactivity of the various fuel assembly designs with the fresh fuel assumption and subsequently for the three most reactive designs evaluated at the highest credited enrichment and burnup combination. The most reactive configuration is the WE 14x14 standard fuel assembly with an initial enrichment of 3.85 wt% U-235 and a burnup of 31 GWD/MTU with a cooling time of 30 years, as demonstrated by the analyses.

The burnup credit analysis in this SAR evaluates all of the eligible PINGP specific fuel assemblies allowed to transport in the TN-40 packages to determine the most reactive cask loading configuration. The WE 14x14 standard fuel assembly is identified as the design basis assembly that bounds other assembly designs having the same initial enrichment, burnup, and cooling time. The minimum assembly burnup required to ensure sub-criticality as a function of the initial enrichment is listed in Table 6-1 of the SAR. Two polynomials have been developed based on the data in Table 6-1 to establish criteria for determining fuel assemblies eligible for loading; one for fuel assemblies with and the other is for fuel assemblies depleted without BPRAs. The following is a list of the assumptions used in the package criticality safety analysis models:

1. No burnable absorbers are included in the KENO models.
2. The fuel density is assumed to be 95.5% theoretical density. There is no allowance for dishing or chamfer of the fuel pellet stack.

3. Only 75% of the boron in the poison plates was credited in criticality safety analysis models.
4. All steel materials are modeled as SS304.
5. All zirconium based materials in the fuel are modeled as Zircalloy-4.
6. Nominal width of the poison plate is used.
7. Conservative and bounding axial burnup profiles are utilized to determine the isotopic concentrations of the burned fuel assemblies. The effects of natural uranium blankets when present and gadolinium based burnable absorbers are not included in the depletion.
8. All depletion calculations are performed assuming that the BPRAs are present in all the guide tubes for at least two-thirds of the depletion. BPRAs are included in these depletion models. Table 6-3 lists the major parameters and values of the fuel assembly super cell model used in the SAS2H models.
9. The light element composition utilized in the SAS2H input is representative and therefore the same composition is utilized for all fuel assemblies modeled.

The applicant discussed the major parameters that impact the depletion and consequently the isotopic concentrations of the spent fuel assemblies loaded in the casks. The following is a list of these parameters that are discussed in the analyses:

1. Axial burnup distribution
2. Fuel temperature
3. Moderator temperature and density
4. Soluble boron concentration
5. Specific power
6. BPRAs
7. Bounding burnup profiles
8. Control rod insertions
9. Horizontal burnup gradient
10. Cooling time

Bounding reactor operating parameters are applied in fuel assembly depletion analyses to ensure that the results are conservative in terms of criticality safety. Specifically, the limiting parameters and their values are: (1) 14 MW/Assembly for average specific power, (2) 0.705 g/cm^3 for moderator density, (3) 584 K (592°F) for moderator temperature, (4) 901 K (1,162°F) for fuel temperature, (5) 620 K (657°F) for cladding temperature, (6) 675 ppm for soluble boron concentration based on an average over the limiting actual non-linear boron letdown curve, (7) 30 years for minimum cooling time, and (8) bounding burnup profiles. An SAS2H model is developed for each axial segment of the fuel assembly.

The applicant performed design basis criticality safety analyses for the TN-40 package using 600 ppm constant soluble boron in the fuel depletion models. In order to assess the sensitivity of the k-eff against the soluble boron concentration used in the fuel depletion analysis models, the applicant also performed a calculation using 675 ppm boron in the depletion analyses. Based on the results of these two cases, the applicant concluded that using 675 ppm versus 600 ppm makes no statistically significant difference in the criticality results.

The staff reviewed the information presented in the SAR and determined that the applicant has not demonstrated that change of soluble boron in the depletion model from 600 ppm to 675 ppm makes no statistical difference in the system criticality analysis results. In addition, the conclusion is not consistent with the findings from the studies performed by Oak Ridge National Laboratory as published in ORNL/TM-1999/99, ORNL/TM-12973, and NUREG/CR-6665. NUREG/CR-6665 pointed out that "the effect of higher boron concentrations is more significant with higher burnup values, since more conversion occurs over the fuel cycle." However, the staff did not verify the result because it was presented as a sensitivity study. The applicant did not provide detailed information on this sensitivity analysis. Consequently, the staff makes no determination on the correctness, nor the accuracy, of the sensitivity calculations.

From the data published in the three reports mentioned above, the staff estimated that the difference in k-inf between the two scenarios, i.e., 600 ppm vs 675 ppm is about 0.0015 to 0.0020 Δk. Based on the results of the criticality safety analysis using 675 ppm in the depletion models, even if this additional penalty is applied to the criticality safety results, the packages with fuels discharged from cycles with average soluble boron of 675 ppm based on an average over the limiting actual non-linear boron letdown curve will still meet the subcriticality requirements. For these reasons, the staff determined that fuels discharged from cycles with average soluble boron of 675 ppm based on an average over the limiting actual non-linear boron letdown curve are acceptable. However, it is emphasized that the regulatory determination for this particular case does not constitute a basis for not using bounding soluble boron in the design basis criticality safety analyses. More detailed information could be needed to use higher soluble boron concentration as a design basis conclusion.

The criticality calculations are performed based on 30 years of cooling time. The applicant further restricts the minimal required cooling time to 30 years. Based on NUREG/CR-6781, "Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses," evaluating the criticality of a burnup credit cask with a cooling time shorter than the actual cooling provides an additional safety margin as long as the cooling time is greater than one year but less than 100 years. For a burnup credit cask that takes both actinides and fission products, the reactivity of the fuels starts to increase after discharge because of the decay of the short lived isotopes. The fuels start to become less reactive after one year of cooling time and this reactivity decrease continues until up to 100 years. Therefore, criticality evaluations performed at a cooling time shorter than the actual cooling time provides additional conservatism in terms of criticality safety. However, this conclusion is true only for fuel cooled less than 100 years. Spent fuel will become more reactive starting around 100 years of cooling time because the concentrations of some major absorbers, such as Am-241, will complete their buildup processes and start to decline from then on. This means that the criticality safety analyses based on 30 years of cooling time may become invalid at around 200 years of cooling time. Hence, the packages must be shipped before the spent fuels in the packages reach their 200 year cooling time. If not shipped by that time, a reevaluation of criticality safety must be performed.

In the SAR, the applicant recognizes that the burnup profile for the spent fuel assemblies is not generally uniform because of the axial neutron leakage from the ends of the assembly. The effect of axial burnup distribution is included in the criticality safety evaluation of this package. As the moderator temperature at the top of the fuel assemblies rises, spectrum hardening is considered in the depletion calculation to account for the reduced U-235 consumption and increased Pu-239 production.

The applicant divided the fuels into several zones along their axial direction in the cask. Different burnups were assigned to these zones based on the selected bounding burnup profile. Thus, each axial zone of the fuel assembly will have a unique material composition. The bounding profile was determined based on the actual operation history of the Prairie Island Nuclear Power Plant. The profile thus determined provides a high level of confidence for the adequacy of the results from modeling. Table 6-5 lists the burnup profiles used in the TN-40 PINGP spent fuel transportation package.

To obtain accurate results for burnup credit analysis, a depletion calculation with high level of fidelity in the depletion code with accurate fuel depletion history in the core(s) is essential. Bounding values are used for parameters that are important to assembly depletion. These parameters include moderator temperature and density, soluble boron concentration, specific power, and the presence of inserts such as BPRAs or control rods. The bounding values of these parameters are determined from PINGP operating history and spent fuel inventory data.

On page 6-12 of the SAR, the applicant indicated that IFBA fuel assemblies are not candidate payload for this design. Therefore, the discussions presented in Section 6.3.1.1 F of the SAR on the Integral Burnable Absorber is neither reviewed nor evaluated because these discussions are not relevant to this application.

The neutron flux at the peripheral locations of the reactor cores often have significant gradient across the horizontal cross section of the fuel assembly. The difference in neutron flux across a fuel assembly will result in uneven burnup. The spent fuel assemblies with significant burnup gradients, when loaded in the cask, may result in significant reactivity increase in the package. The SAR considered this factor in the criticality safety evaluation of the TN-40 package.

The applicant identified the most reactive fuel design among all authorized contents. The applicant also studied the reactivity impact of the fuel assembly and basket fabrication tolerances. The result of this study was used as an adjustment to the k_{eff} value of the cask to account for these tolerances. The staff reviewed the approach and the results of the analyses and found the approach is appropriate and the results are acceptable.

6.3.2 Package Regional Densities

All of the materials used in the criticality safety evaluation of the TN-40 packages are the standard material compositions from the SCALE-4.4 system standard material composition library. Materials that are not available in the library were created as either a compound or alloy, or mixture using free gas model. The applicant determined that the isotopes for which burnup credit was taken are not (1) soluble, (2) volatile, or (3) gaseous and are expected to stay in the packages under all conditions. The isotopes, particularly the fission products that are accounted for burnup credit, are relatively long lived for the duration of the transportation period. The reactivity for the burnup credit casks is not expected to change due to the change under NCT and HAC conditions. The staff reviewed the material properties of the isotopes and finds the applicant's conclusion acceptable.

In order to obtain reliable and accurate criticality safety analysis results, the calculated values of the isotopic concentrations for all of the 27 nuclides must be benchmarked to measured spent fuel assay data for various enrichments and burnups of a given fuel assembly design and depletion conditions, including core operating parameters and cooling time.

The applicant performed the code benchmark calculations by following the methodologies proposed in NUREG/CR-6811, “Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses.” Section 6.8 of this SER provides more detailed discussions and the staff’s evaluation on the applicant’s benchmark calculations of the fuel assembly depletion code.

6.4 Single Package Evaluation

The applicant analyzed the criticality safety for a single package under NCT and HAC. The package criticality model includes all of the components except the impact limiters. License drawings 10421-71-3 and 10421-71-6 show the structure design and the geometric dimensions of the packaging structure and license drawings 10421-71-8 and 10421-71-9 provide the structure of the fuel basket. The applicant calculated the k_{eff} of the package with axial eight zone burnup profile and bounding burnup. Table 6-11 provides the number densities of the isotopes used in the criticality safety calculations.

The criticality analyses were performed using the CSAS25 module of the SCALE-4.4 code system. The analyses used the 44-group cross section library that was developed using the ENDF/B-V data. Only 75% credit is taken for the B-10 in the poison plates. Based on the results of criticality safety analyses, the applicant determined the minimum burnup as a function of fuel enrichment necessary to meet the criticality safety requirements.

Table 6-1 provides the data that define the minimal burnup for given initial enrichment. Based on the data presented in Table 6-1, the applicant developed two loading curves, one for the fuel assemblies that had control rod or BPRA insertion history and the other one is for the assemblies that had no history of BPRA or control rod insertion. Two equations were obtained with least square regression analyses for these loading curves. The mathematical form of the loading curve for fuel assemblies that had BPRA or control rod insertion histories is:

$$B = -1259.8 \times E^2 + 20242 \times E - 23617$$

The mathematical form of the loading curve for fuel assemblies that had no BPRA or control rod insertion histories is:

$$B = -366.95.8 \times E^2 + 14770 \times E - 17200$$

where B is burnup in MWd/MTU and E is fuel initial percent enrichment. The valid range of U-235 enrichments is 2.0% to 4.0%. These two equations are included in the CoC for determining the criterion for fuel qualification and cask loading designs.

The applicant also performed criticality safety analyses for the TN-40 packages under HAC. The following are the major assumptions of the package under HAC:

1. Reactivity calculated at optimum moderator density (assuming full density water flooding into the cask cavity) with the fuel.
2. The pellet/cladding gaps are flooded with fresh water.

3. Only the active fuel length of each assembly type is explicitly modeled with water boundary conditions at both ends.
4. The neutron shield and steel outer shell of the cask are lost and the infinite array of casks is pushed close together with moderator in the interstitial spaces.
5. Temperature at 20^{o}C (293 K) is used for the criticality calculations.
6. The fuel density is assumed to be 95.5% theoretical density. There is no allowance for dishing or chamfer of the fuel pellet stack.
7. Nominal assembly compartment width is used.
8. Conservative and bounding axial burnup profiles are utilized to determine the isotopic concentrations of the burned fuel assemblies.
9. Full moderator density represents the most reactive condition because the power reactor fuel assemblies are all designed to be slightly under-moderated.

The staff reviewed these assumptions and finds they are acceptable according to the requirements of 10 CFR 71.55, 71.59, and 71.73 and the guidance provided in the Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (NUREG-1617).

The staff finds the conclusions of these sensitivity studies acceptable because the purpose of these studies is to determine the system sensitivity to changes in geometric configuration and cask fabrication tolerances rather than the absolute values of the system under study. Fresh fuel assumption used in the models will not significantly affect the result of cask system configuration sensitivity study.

The applicant evaluated the effect of using constant soluble boron concentration versus a soluble boron letdown curve to the fuel assembly depletion analyses. The applicant concluded that the effect is insignificant. The applicant also stated that the assumption of a constant boron concentration during depletion is adequate and representative of the operating parameters of the Prairie Island Nuclear Generating Plant.

The staff reviewed the applicant's justification; studies performed by researchers, such as Oak Ridge National Laboratory; and the actual plant operation records. Based on its review the staff concludes that the applicant's assumptions and evaluations are acceptable because the assembly depletion modeled with constant soluble boron produces slightly conservative results, i.e., higher k_{eff}.

The applicant discussed in Section 6.4.2.C of the SAR some additional criticality safety margins that were included in the criticality safety evaluations of the TN-40 packages. The additional criticality safety margins include: (1) additional cooling time, (2) conservative isotopic concentration scaling factors, (3) the negative reactivity contribution due to the loading of Non-Fuel Hardware or inserts such as BPRAs in the fuel assemblies, (4) the actual loadings of the casks on average have at least five assemblies with burnup greater than the design basis, (5) all calculations are performed based on the Westinghouse 14x14 assembly which is the most reactive among all the assembly designs, (6) the fuel assembly depletion analyses all assumed 16 BPRAs in the assemblies, and (7) fuel assemblies with natural uranium blankets were evaluated as without, i.e., the entire assembly has the same enrichment as the enriched fuel.

The applicant provides in Section 6.4.2, Subsection E of the SAR detailed discussions for these additional safety margins. The staff reviewed the discussions on these additional reactivity margins associated with the criticality safety analyses of the TN-40 PINGP spent fuel transportation package. The staff considers only the following item can be included as additional safety margin:

1. Use burnup lower than the actual burnup of the fuel assemblies. This strategy introduces some additional safety margins because using a lower burnup in depletion will yield higher fissile material and lower fission product concentrations than what are actually in the fuels. Consequently, the criticality calculation will give k_{eff} that is higher than the actual package has. Hence, this approach qualified for providing additional safety margin.

The staff finds some of the factors discussed in Section 6.4.2 of the SAR cannot be qualified as additional safety margins. The following are discussions for the reasons to exclude these factors from being considered as additional safety margins.

1. For item (ii), the effect is not considered qualified as additional safety margin. The rounding up of correction factors for fissile isotopes and truncating of correction factors for absorber isotopes are part of the methodology for the correction factors to account for the uncertainties associated with the calculated isotopic concentrations of the spent fuels rather than some additional conservatism.

2. For item (iii), the effect is not considered qualified as additional safety margin. The presence of non-fuel hardware inserts in the spent fuel assemblies in casks is not a design basis requirement. The users do not have to load any cask with inserts. If inserts are indeed loaded in a cask, it would lead to an additional safety margin. However, loading fuel assemblies with inserts in the cask is not required, nor guaranteed. The most reactive loading dictates the criticality safety margin of the cask design.

3. These additional safety margins cannot include the conservatism taken in determining the final correction factors for fission products for which burnup credits are taken. The truncation of the calculated correction factors for fission products is part of the methodology to account for the unqualified uncertainties in the calculations and therefore cannot be accounted as additional safety margin.

4. Neglecting natural uranium blankets in criticality calculation. This is not qualified as additional safety margin because not all fuel assemblies have natural uranium blankets. Although neglecting natural uranium blankets in criticality safety analyses for those casks loaded with spent fuels containing natural uranium blankets does provide some additional safety margin, this treatment, however, cannot be considered as additional safety margin for the general package design because this margin would not be available for casks that are loaded with regular fuel assemblies.

5. Use of WE 14x14 fuel assembly design as the design basis fuel. This assumption does not qualify as additional safety margin because the user can load one cask with all WE 14x14 fuel assemblies. However, this design does bound all casks

loaded with fuel assemblies at Prairie Island that are qualified for transportation using the TN-40 package design for this approval.

6. Use of eight instead of 18 zones in criticality analysis. This is a simplification. It serves the purpose of simplifying calculation rather than a factor for additional safety margin. The results shown in Table 6-16c demonstrate that the model that uses 18 zones gives slightly higher k_{eff} for both the cases presented in the table. The staff determined the simplification in modeling with seven zones instead of 18 zones as recommended by NUREG/CR-6811 because the difference is small.

7. Use of shorter cooling time than the actual fuel assembly cooling time can be claimed as additional safety margin because shorter cooling time over predicts the concentrations of short-lived absorber isotopes that have decreased due to decay. However, the applicant used 15 years as cooling time only in the sensitivity analyses. The design basis analyses are all based on 30 year of cooling time. Therefore, the extra safety margin due to longer cooling time is not available to this design.

The applicant also performed criticality safety analyses for the TN-40 packages under HAC. The following are the major assumptions of the package under HAC:

1. Reactivity calculated at optimum moderator density (assuming full density water flooding into the cask cavity) with the fuel.
2. The pellet/cladding gaps are flooded with fresh water.
3. Only the active fuel length of each assembly type is explicitly modeled with water boundary conditions at both ends.
4. The neutron shield and steel outer shell of the cask are lost and the infinite array of casks is pushed close together with moderator in the interstitial spaces.
5. Temperature at 20°C (293 K or 68°F) is used for the criticality calculations.
6. The fuel density is assumed to be 95.5% theoretical density. There is no allowance for dishing or chamfer of the fuel pellet stack.
7. Nominal assembly compartment width is used.
8. Conservative and bounding axial burnup profiles are utilized to determine the isotopic concentrations of the burned fuel assemblies.
9. Full moderator density represents the most reactive condition because the assemblies are designed to be slightly under-moderated.

The criticality safety analysis results for a single package under HAC are provided in Table 6-16a. The maximum k_{eff} value for a single package under HAC is 0.9344.

The staff reviewed these assumptions and finds they are acceptable according to the requirements of 10 CFR 71.55, 71.59, and 71.73 and the guidance provided in the Standard Review Plan for Transport Packages for Spent Nuclear Fuel (NUREG-1617).

Based on structural analyses, the applicant determined that fuel assembly would not experience plastic deformation under a 30-feet end-drop accident. As a result, criticality impact of fuel reconfiguration has been excluded from criticality safety analysis.

Based on structural analyses, the applicant determined that there will be a 0.023 inch change in fuel compartment due to deflection under 30-foot end-drop. The criticality impact of this change in the fuel compartment dimension is bounded by the fuel basket manufacturer tolerances and its effect has been included in the criticality safety analyses.

6.5 Evaluation of Array of Packages under Normal Conditions of Transport

The applicant analyzed the criticality safety of an infinite array of packages under NCT. The applicant demonstrated that an infinite array of packages under NCT remains subcritical and is under the Upper Sub-criticality Limit calculated in the code benchmark process as discussed in Section 6.7 of this SER.

6.6 Evaluation of Array of Packages under Hypothetical Accident Conditions

The applicant also analyzed the criticality safety of an infinite array of packages under HAC. The applicant demonstrated that an infinite array of packages under HAC remain subcritical and is under the Upper Sub-criticality Limit calculated in the code benchmark process as discussed in Section 6.7 of this SER. Since an infinite array of packages under both NCT and HAC is subcritical, the package Criticality Safety Index calculated in accordance with 10 CFR 71.59 is 0. The Criticality Safety Index (CSI) for the TN-40 Prairie Island Nuclear Power Station spent fuel transportation package is determined to be zero. Therefore, an infinite number of packages may be transported in a single conveyance as far as criticality safety is concerned.

6.7 Critical Benchmark Experiments

The applicant discussed the applicability of the SCALE-4.4 code system for the criticality safety evaluations of the TN-40 spent fuel packages. The applicant discussed the potential bias associated with the model for this specific application. One hundred forty two benchmark critical experiments were selected for this applicability and trend analyses. Some of these critical benchmark experiments were fresh UO_2 fuel and some were MOX fuel. The applicant used CRC data for criticality code benchmarking to supplement the fresh fuel and MOX fuel benchmark critical experiments which do not contain the fission products that were included in the burnup credit.

The applicant used the CSAS25 module of SCALE-4.4 code system in the TN-40 cask criticality safety analysis. In order to determine the biases and uncertainties introduced in the criticality safety analysis code and modeling methodology, the criticality safety analysis code must be benchmarked. Because the burnup credit cask includes actinides and fission products, the critical experiments selected for code benchmarking must include adequate representation of the material compositions with great similarities to that of the spent fuels. In addition, the critical experiments for spent fuel transportation cask should also have good representation of the material distributions, such as the basket geometry and poison plates, in the cask. However, critical experiments with good representation of spent fuel compositions and spent fuel transportation cask are not readily available.

To alleviate the problem of lacking critical experiments for benchmark, Oak Ridge National Laboratory developed a methodology that can use the Commercial Reactor Criticals as benchmark critical experiments for spent fuel package criticality safety analysis computer code. The applicant selected from the Oak Ridge National Laboratory Technical Report ORNL/TM-12294, V1, "SCALE-4 Analysis of Pressurized Water Reactor Critical Configurations: Volume 1 – Summary" some Commercial Reactor Critical records and used them in the code benchmark and USL calculations. The applicant followed the methodologies and examples presented in NUREG/CR-6951, "Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit."

The results of the benchmark are the Upper Sub-criticality Limit (USL). The applicant calculated the USL following the methodology published in NUREG/CR-6361 (Ref. 4). Table 6-21 of the SAR provides the results of the USL evaluation. Table 6-22 shows the bounding USL versus different control parameters. Based on the data in Table 6-22, the applicant determined that the bounding USL for the TN-40 is 0.9402.

Based on the results of the criticality analysis, the k_{safe} of the cask is 0.9360. This result demonstrated that the cask design provides a reasonable assurance for criticality safety. The maximum neutron multiplication factor for the cask is below the USL.

The model bias is determined based on these analyses. A summary of all of the pertinent parameters for each experiment is included in Table 6-20 along with the results of each run. The best correlation is observed for fuel assembly separation distance with a correlation of 0.64. All other parameters show much lower correlation ratios indicating no real correlation. All parameters were evaluated for trends and to determine the most conservative USL. The minimum USL is 0.9383.

6.8 Burnup Credit

The criticality safety analysis of a burnup credit cask must first adequately determine the isotopic concentrations of the nuclides for which burnup credit was taken. For this purpose, the fuel depletion analysis code must be benchmarked to measured chemical assay data for the fuel assemblies.

Based on the guideline provided in NUREG/CR-6811 (Ref. 4), there are three methods that can be used to determine the biases and uncertainties associated with the calculated isotopic concentrations of the isotopes for which burnup is taken. The first one is called "Correction Factor" method, the second one is called "Direct Difference" method, and the third one is called "Monte Carlo Uncertainty Sampling" method.

There are advantages and shortfalls in each of the methods. The correction factor method adjusts the calculated isotopic concentration using the correction factor calculated based on the average ratio of the measured versus the calculated isotopic concentrations for each isotope that is included in the burnup credit calculation. As described in NUREG/CR-6811, the best estimate correction factor method compares the calculated isotopic concentrations for the isotopes of interest with that of the calculated data to obtain a set of the ratios between the measured and the calculated. Following this approach, a measured/calculated ratio is obtained for each isotope with given initial enrichment, burnup, and cooling time. A statistical analysis is performed to determine the mean value and standard deviation of the ratio for each of the isotopes. A factor for correction of the calculated isotopic concentration is determined as the statistical mean plus two times of the standard deviation of the measured/calculated ratio. Thus, the calculated isotopic concentration with correction of the correction factor for each of the isotopes, with 95% confidence, will be correct 95% of the time. In order to add more conservatism, the method adds further restrictions to the determination of the correction factors. For the absorber isotopes, the calculated correction factors must be truncated to the nearest lower values if they are below 1.0 and the maximum value must not exceed 1.0. For the fissile isotopes, the correction factors must be rounded up to the nearest higher values. The estimate correction factor for each isotope is the statistical mean of the ratio of the measured over calculated isotopic concentration for each isotope. The result of the last step is the correction factor.

The final correction factors are applied to the calculated isotopic concentrations. The adjusted isotopic concentrations are then fed into the criticality safety analysis code to determine the k_{eff} of the cask. The shortfall of the correction factor method is that it is in some cases overly conservative to the burnup credit. However, for a scenario in which the uncertainties and biases cannot be accurately determined, it is prudent to take a conservative approach in treating the results of the criticality safety calculations.

In the direct difference method, the trends and biases associated with the selected isotopes are treated together as a lump sum of correction value to the k_{eff} of the cask. In this method, two criticality calculation models are made. The first one uses the measured isotopic concentrations of the selected isotopes for burnup credit and the second one uses the isotopic concentrations calculated by the fuel assembly depletion code. Two k_{eff} values are then determined by the criticality safety models. The difference between the two k_{eff} values is determined for each set of measured/calculated isotopic concentration data.

Because these isotopic radiochemical assay data were measured from a wide variety of fuel assemblies discharged from various reactors, there are several concerns on the quality of these data with respect to burnup credit. The following are major factors that may impact the results of code benchmark analyses:

1. There are many critical parameters that may affect the results of these measurements. The important parameters include enrichment, burnup, cooling time, H/U ratio, fuel assembly geometry, and fuel depletion history. Because the measured data were taken from the fuel samples that were irradiated under the influences of these factors, the models that are constructed for fuel assembly depletion analysis must take into account the influences from these factors.

2. The various available measurements were made for different purposes and the majority of them are not for burnup credit. As such, the isotopes included in the measured data vary from measurement to measurement and are probably not ideal for code benchmark.

3. The locations where the samples were taken, i.e., the middle of the rods, top tips of the rods, bottom parts of the rods, and the location of the rods in the fuel assemblies, i.e., in the locations surrounded by fuel rods, the locations close to water rods, water gaps, and the locations close to control rods or BPRA rods. The models for code benchmarking must be able to simulate the neutronic characteristics of the locals where the samples were taken.

4. The irradiation histories of these samples may not be accurately reflected in the data records. Major fuel depletion parameters, such as fuel temperature, moderator temperature, moderator density, soluble boron concentration, and power density data at the locations where the samples were taken may not always be available. Consequently, using the limited measurement data to perform benchmarks for the depletion models that are used to determine the isotopic concentrations in the spent fuels may be flawed.

5. Not all measurements include all isotopes that are desired for the burnup credit calculations; some of the isotopes have only a few data points. As a result, the measured data for some isotopes are very scarce; there are only a very limited number of experimental measurements for some isotopes. The results of the isotopic concentration calculations tend to have large uncertainties for some of the isotopes. The lack of data also makes it difficult to assess the distribution of the data in terms of normality. The users must test the normality of the measured data in order to avoid missing significant trending with regard to the control parameters.

6. The uncertainties in the sample data also create some challenges in using these data for burnup credit application because it is difficult to simulate the exact irradiation histories of the fuel samples in the depletion models for code benchmarking. The users must be aware of the deficiencies embedded in the measured data. To account for the unknown uncertainties associated with these data, the user must include additional safety margins when determining the burnup credit. Unless new sources of measured data become available to support more vigorous validation and benchmark of the model, sufficient safety margins must be assessed against the burnup credit.

6.8.1 Direct difference method for burnup credit benchmark

Oak Ridge National Laboratory developed the direct difference method for burnup credit analysis with the support of the US NRC. This method is published in NUREG/CR-6811. In essence, the direct difference method first computes the k_{eff} values of two casks; one uses the measured isotopic concentrations and the other uses the isotopic concentrations calculated by a fuel assembly depletion analysis code. Thus a pair of k_{eff} values is obtained. A Δk_{eff} value is thus obtained for each set of measured data. The statistical means and standard deviations of these differences in Δk_{eff} are determined using the following equations:

$$\Delta k_{eff} = k_{eff}^{m} - k_{eff}^{calc}$$

$$\overline{\Delta k_{eff}} = \frac{1}{n}\sum_{i=1}^{n}(k_{eff}^{m,i} - k_{eff}^{calc,i})$$

$$\sigma_m \approx \sqrt{\frac{1}{n}\sum_{i=1}^{n}(\sigma_m^i)^2}$$

$$\sigma_{calc} \approx \sqrt{\frac{1}{n}\sum_{i=1}^{n}(\sigma_{calc}^i)^2}$$

$$\sigma = \sqrt{(\sigma_m^2 + \sigma_{calc}^2)}\ ,$$

where k_{eff}^{m} and k_{eff}^{calc} are the k_{eff} values of the cask models using measured and calculated actinide concentration values respectively. σ_{m}^{i}, σ_{m}, σ_{calc}^{i}, and σ_{calc} are the uncertainties associated with calculated k_{eff} value using the i^{th} measured data set, the average uncertainty associated with the calculated k_{eff} value using the measured actinide data set, the uncertainty associated with calculated k_{eff} value using the i^{th} calculated actinide data set, and the average uncertainty associated with the calculated k_{eff} value using the calculated actinide data set respectively.

The staff reviewed the methodology employed in the analyses and the results presented in the SAR and determined that the conclusion is acceptable. The calculation method used by the applicant is consistent with the approach described in NUREG/CR-6811 for bias and uncertainty calculations as part of the direct difference method and NUREG/CR-6361 for trend analysis.

6.9 Misload Analysis

The applicant established measures in the operating procedures to provide assurance that the packages have been loaded correctly, i.e., loading the correct fuel assemblies into designated basket locations. However, cask misloads have occurred in the past in other facilities as well as Prairie Island Nuclear Generation Station. The applicant discussed in Appendix 6A of the SAR the administrative procedures that have been established to provide protection against the loading a fuel assembly with a burnup value less than that required by the loading curve. The discussion is broken down into the three most likely causes that a misloading could occur: 1) incorrect selection of the fuel assemblies during loading, 2) incorrect calculated burnup values from the core depletion calculation, and 3) incorrect burnup value assigned to a fuel assembly during data transposition.

The applicant outlined measures to prevent the first type of misload from happening:

1. Verifying each fuel assembly to be loaded satisfies the loading requirements listed in Section 1.2.3. This verification shall be performed by two independent individuals.
2. The assigned burnup loading value for each fuel assembly shall be from the TOTE or BURNUP computer codes. The burnup value from the output of these codes shall be reduced by a factor of 1.04 to account for the uncertainties of the calculated burnup values.
3. Once prior to inserting into cask and once prior to closure of cask, verify the identity of each fuel assembly. This verification shall be performed by two independent individuals.

The applicant also performed analyses for the TN-40 package misloading and the corresponding impact to the criticality safety of the packages based on the specific spent inventory of the Prairie Island Nuclear Generating Plant. The applicant demonstrated that in the worst case scenario the packages remain subcritical. Based on these analyses, the staff determined that the package design meets the criticality safety requirements of 10 CFR 71.55 and 59.

6.10 Evaluation Findings

The staff reviewed information presented in the criticality safety analyses for the TN-40 spent fuel transportation packages with confirmatory analyses, the applicant's responses to the staff's Requests for Additional Information, and supplemental calculation packages. The staff developed criticality analysis models to perform confirmatory analysis. Based on the review of the information presented in the SAR and supplemental calculation packages and the results of confirmatory analysis, the staff finds that criticality safety analyses as well as the results are acceptable and package design meets the criticality safety requirements in accordance to the requirement of 10 CFR Part 71 and the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel", and ISG-8, Rev. 2, "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," with the following conditions:

1. The boron-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,
2. IFBA is not an authorized content,
3. Cooling time equal or greater than 30 years but less than 200 years, and fuel assemblies were irradiated with the following parameters:
   (1) The minimum average specific power shall be 14 MW/Assembly,
   (2) The minimum hot leg average moderator density shall be 0.705 g/cm^3,
   (3) The maximum hot leg average moderator temperature shall be 584 K (592°F),
   (4) The average fuel temperature shall be 901 K (1,162°F), and
   (5) The maximum average soluble boron concentration shall not exceed 675 parts per million based on an average over actual non-linear boron letdown curve.

These conditions are the design basis of the package and should be captured in the CoC of this package.

References:

1. "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, Oak Ridge National Laboratory, September 1996.
2. Strategies for Application of Isotopic Uncertainties in Burnup-Credit," NUREG/CR-6811 (ORNL/TM-2001/257), U.S. Nuclear Regulatory Commission, June 2003.
3. "Three Mile Island Unit 1 Radiochemical Assay Comparison to SAS2H Calculations," Office of Civilian Radioactive Waste Management, U.S. Department of Energy, April 2002.
4. "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," NUREG/CR-6665, Oak Ridge National Laboratory, February 2000.
5. "Strategies for Application of Isotopic Uncertainties in Burnup Credit," Oak Ridge National Laboratory, June 2003.

7.0 PACKAGE OPERATION

Chapter 7 of the SAR specifies operating procedures for the TN-40 package. The chapter includes sections on the receipt of an empty TN-40 package, determination of eligible fuels and non-fuel hardware contents, loading the package, preparing it for transport or moving it to a storage area, converting it from the storage configuration to the transport configuration, unloading, and shipping it as an empty package. As part of the approval, the users of the TN-40 packaging system are required to perform a number of additional activities during the loading of or preparing a loaded TN-40 package for transport in order to satisfy the design limitations in the areas of structural, thermal, containment, shielding, and criticality.

7.1 Package Loading

Section 7.1 of the SAR lists the sequence of steps in preparing an empty TN-40 for loading, loading contents, and preparing the loaded package for direct transport instead of placing it in storage.

Chapter 7 of the SAR describes the operation procedures and requirements for the TN-40 package. The applicant also indicates in Chapter 7 that a separate Operations Manual will be prepared for the TN-40 transportation package to describe in greater details the operational steps. The operations required to convert the TN-40 cask from its storage configuration to the transport configuration are also described in Chapter 7 (see Section 7.4).

In Steps 7.1.1.1 through 7.1.1.4 and 7.1.2.5 of the SAR, the applicant outlined the steps in verifying the spent fuel assemblies to be loaded. This is done to reduce the probability of misload with respect to burnup based on which TN-40 criticality control design is approved. The burnup verification described in the loading procedure combined with misload analyses performed by the applicant is part of the basis for approval of the TN-40 package using burnup credit. Furthermore, the core conditions to which the fuel assemblies were exposed should be also verified. As indicated in Chapter 6 of this SER the following fuel assembly irradiation parameter values should be verified prior to loading spent fuel assemblies in the TN-40 cask:

1. The minimum average specific power shall be 14 MW/Assembly,
2. The minimum hot leg average moderator density shall be 0.705 g/cm
3. The maximum hot leg average moderator temperature shall be 584 K (592°F),
4. The average fuel temperature shall not exceed 901 K (1,162°F),
5. The maximum average soluble boron concentration shall not exceed 675 parts per million based on an average over the limiting actual non-linear boron letdown curve, and
6. The maximum cooling time is 200 years. If the cooling time of the spent fuel assemblies in the cask exceeds 200 years at the time of shipment, a reassessment of criticality safety of the cask shall be performed prior to transport.

The cask will be placed in a "cask loading pool" which is the term used to describe the area where the cask is to be loaded. The applicant also listed steps in Chapter 7 of the SAR for loading the cask with the pre-selected spent fuel assemblies. Additionally, as part of the structural approval with respect to the maximum gap requirement, an aluminum spacer plate with nominal dimensions of 0.75” thickness and 71.75” diameter must be placed on the top of the basket as described in Step 7.1.2.6. The spacer will limit the amount of stresses imposed to the closure lid bolts and the spent fuel assemblies under HAC. To secure the closure lid, 48 SA-540 Grade B23 Class 1 bolts shall be used.

Steps 7.1.3.4 to 7.1.3.6 of the operating procedures provide a detailed description for vacuum drying, moisture removal, and helium backfilling operations and requirements. The users shall follow these step by step operational instructions to make sure the cask is correctly dried to meet the moisture content limit and backfilled with helium to the maximum pressure limit.

Once the fuels have been verified with correct control parameters, loaded in the basket, the spacer plate has been installed in the cask, and metal seals have been replaced, the users shall follow the steps listed for draining, closing, vacuum drying, pressurizing, and leak testing the entire containment boundary of the package with a revised acceptance criterion of 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less). Steps 7.1.3.7 to 7.1.3.9 of the operating procedures provide a detailed description for the leak testing. The users shall follow these steps to complete leak tests to assure the package meets the containment requirements. The user shall perform a leak rate test prior to shipment, with an acceptance criterion of 1×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less).

Step 7.1.3.15 of the TN-40 operating procedures requires the user to perform a comprehensive neutron and gamma dose rate survey over the entire surface of the cask to demonstrate the adequacy of the shielding design, to meet the intent of the shielding tests (see Chapter 8 of this SER), and to check if the surface dose rates are within the regulatory limits. Surface contamination levels are checked to verify that levels are within the regulatory limits. Also, an external temperature survey is performed as described in Section 3.4.7 for monitoring thermal performance.

Prior to releasing the loaded cask for shipment, the user shall perform a final radiation and contamination survey to assure compliance with 10 CFR 71.47 and 71.87.

As part of the loading procedure, in lieu of a thermal acceptance test that has not been performed on already loaded casks, a temperature survey must be performed on each loaded cask and the results to be compared to calculated outer shell temperatures from SAR thermal model analyses with appropriate adjustments for decay heat and ambient temperature. Prior to releasing the loaded cask for shipment, the temperature on all accessible surfaces will be checked to make sure it is less than 85°C (185°F).

7.2 Package Unloading

Section 7.2 of the TN-40 SAR provides steps for unloading the package. The applicant has listed necessary steps to visually check for irregularities that may have occurred and to perform a radiation survey and leak test to ensure seals have not been degraded. A cavity gas sample is also required to ensure fuel has not degraded.

The cask may be filled with demineralized water before lowering it into the pool if the total weight (cask filled with water) does not exceed the lift capacity of the crane. The water pumping rate must not exceed 1 gallon per minute while continuously monitoring the exit pressure. Continue pumping water at 1 gpm for at least 80 minutes. The flow rate can then be gradually increased but the outlet pressure must not exceed 55.3 psig. If the outlet pressure exceeds

55.3 psig, the inlet value must be shut off to allow the pressure to recede to 50 psig before resuming reflooding the cask.

When the contents are ready to be removed, the cask is placed in the cask pit area inside the spent fuel pool and the pressure conditions are monitored as the cask is flooded. While flooded, the cavity spacer plate is removed and the spent fuel assemblies are unloaded. Upon removal of the fuel assemblies, the cask is drained and decontaminated. Steps 7.2.1.1 to 7.2.3.10 of the SAR provide step-by-step guidance for unloading the TN-40 cask.

7.3 Preparation of Empty Package for Transport

Section 7.3 of the SAR provides steps to prepare the empty package for transport. Empty packages will be shipped in accordance with the requirements of 49 CFR 173.428. Staff has reviewed the package operations described in Section 7.3 of the SAR and finds them to be acceptable.

7.4 Other Procedures

Because the TN-40 cask is designed for both storage and transport, the user must convert the TN-40 from the storage configuration to the transport configuration if the cask is in the storage configuration prior to transport. Section 7.4 provides steps to convert the TN-40 cask from a storage configuration to the transport configuration. This procedure assumes all TN-40 casks that are in storage have lid bolts that are to be replaced with higher strength bolts, shown on SAR drawing 10421-71-1, and that a cavity spacer plate is installed to minimize the gap between fuel assembly top nozzles and the inner surface of the lid prior to shipping. The nominal dimensions of the spacer plate are 0.75 in. thick x 71.75 in. O.D.

In some cases, casks in the storage configuration may not have impact limiter bracket mounts. For these cases, the impact limiter bracket mounts must be welded to the outer shell for transport. In addition, some casks that were configured for storage may not have the transportation regulatory name plates. In these cases, appropriate nameplates must be installed.

During the conversion, the fuel assemblies in the casks will be checked by two independent individuals to ensure they satisfy the content requirements specified in the CoC. The burnup loading value for each fuel assembly shall be obtained from a source controlled by the site's QA program and traceable TOTE or BURNUP computer codes and are reduced by a factor of 1.04 to account for uncertainties of the calculations.

The user shall review the cask maintenance and operational records to identify situations where air may have leaked into the cask while on the storage pad. If air in-leakage has occurred, prior to transportation, the user must perform an evaluation to verify that the fuel is not damaged using the methodology provided in ISG-22 for potential rod split due to exposure to an oxidizing atmosphere.

While in the loading area, a cavity gas sample is taken to check for any necessary precautions based on the sample results. If degraded fuel is suspected, additional measures, appropriate for the specific conditions, must be developed, reviewed, and approved by appropriate site personnel and implemented to minimize worker exposure and release of radioactive material to the environment.

Section 7.4 provides operating procedures for converting the cask storage configuration into the transport configuration. Steps 7.4.1.10 through 7.4.1.33 are very similar to cask loading procedures as described in steps 7.1.2.1 to 7.1.3.26. When ready, the cask is lowered into the spent fuel pool cask pit where it is flooded as the lid, vent port, drain port, and OP port seals are replaced and the cask cavity spacer plate is installed. The cask is then drained and the cask cavity is evacuated to remove any remaining moisture and backfilled with helium. The containment leakage test is then performed on the entire containment boundary of the package to ensure the maximum acceptable leak rate is not exceeded.

The cask is placed on the transport vehicle and the user performs a comprehensive neutron and gamma dose rate survey over the entire surface of the cask to demonstrate the adequacy of the shielding design, to meet the intent of the shielding tests (see Chapter 8 of this SER), and to check if the surface dose rates are within the regulatory limits. The surface contamination levels are also checked to verify compliance with the regulatory limits.

A temperature survey is performed on each loaded cask and the results are compared to calculated outer shell temperatures from SAR thermal model analysis with appropriate adjustments for decay heat and ambient temperature. Additionally, the temperatures on the accessible surfaces are checked to be sure they are below 85°C (185°F).

Prior to releasing the loaded cask for shipment, a final radiation and contamination survey to assure compliance with 10 CFR 71.47 and 71.87 is performed and the appropriate DOT labels and placards are placed in accordance to 49 CFR 172.

7.5 Evaluation Findings

The staff reviewed the operating procedures for the TN-40 spent fuel transportation package for the PINGP. The staff followed the guidance provided in the "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617 in its review. Based on the statements and representations in the application, the staff concludes that the operating procedures have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71 with the following conditions:

1. 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;
2. As part of the preparation for transport, the 48 as-installed 1.375-in. diameter SA-320 Grade LA43 closure lid bolts are replaced by the SA-540 Grade B23 Class 1 bolts of the same configuration;
3. 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 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-5} ref-cm^3/sec or less) in compliance with ANSI N14.5;
4. As part of the preparation for transport, the user shall perform a leak rate test of the entire containment boundary prior to shipment, with an acceptance criterion of 1.0×10^{-4} ref-cm^3/sec (at a sensitivity of 5.0×10^{-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;

5. 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 SAR, with appropriate adjustments for decay heat and ambient temperature. The temperature difference between calculated and measured values shall not exceed ±25°F;
6. For casks previously loaded under 10 CFR Part 72 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 cask. Total dose rates from these surveys must meet the regulatory limits in 10 CFR 71.47. This comprehensive measurement requirement is necessary to meet the intent of the shielding acceptance and periodic tests; and
7. For casks that are configured for storage, the operating procedures prescribed in Section 7.4 must be used to convert the storage configuration to the transportation configuration of the package.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the SAR specifies various acceptance tests that the packaging will undergo and describes the maintenance program in place to assure packaging performance during its service life will be adequate.

8.1 Acceptance Tests

Section 8.1 of the SAR specifies various acceptance tests that shall be performed prior to the first use of the package. These tests are designed to examine the acceptability of the packaging design and fabrication as well as the expected performance during operations. These acceptance tests include visual inspections, structural and pressure tests, and containment boundary leakage tests.

Visual inspections will be done to ensure that the packaging conforms to the drawings and specifications. Visual inspections include visual check of the packaging with regard to the cleanness of the product, quality of the welds per the requirement of ASME Boiler and Pressure Vessel Code Section III, 1989 Edition, sealing surface finish, and conformance of dimensions to design drawings, which are included by reference as part of the CoC. Specifically, the visual inspections are to verify that all specified coatings are applied and the packaging is free of defects that could reduce its effectiveness or result in unacceptable leakage.

Structural and pressure tests will be performed on the structural materials to ensure that the packaging can perform its design function. The structural materials are chemically and physically tested to confirm they meet the required properties. The component welds and basket will be designed, fabricated, and inspected in accordance with the ASME B&PV Code. The containment welds are tested and inspected in accordance with ASME B&PV Code Subsection NB. The fuel baskets are inspected in accordance with ASME B&PV Code Subsection NB. The impact limiter attachment bolt material and tie rod material will have their Charpy values tested.

A pressure test will be performed on the cask assembly in accordance with ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6200 or NB-6300. A bubble leak test will be performed on the neutron shield enclosure to identify any potential leak passages in the enclosure welds.

A load test of 1.5 times the design lift load is applied to the trunnions for a period of ten (10) minutes. The impact limiters will undergo load tests with 1.5 times of the weight and examined for defects and permanent deformations after the tests.

Prior to shipment of the cask used in the storage for the transport, the helium leakage rate test should be performed on the entire containment boundary of the package (using a test envelope) at the storage/loading area to meet the containment requirements in Part 71. The leakage test will be performed in accordance with ANSI N14.5 where the acceptance criterion is 1.0×10^{-4} ref-cm^3/sec.

The acceptance tests of the TN-40 components include tests of gasket and impact limiter leakage. After all the seal welds are completed on the impact limiter and prior to initial use the limiters will be pressurized and will undergo a soap bubble test to test the weld seams for leakage. This test will verify that the impact limiter wood will be protected from any moisture exchange with the environment.

Functional tests will be performed for installation and removal of lid, penetration covers and other fittings. Each component shall be checked for operational difficulties, indication of deformation, galling, improper function etc. Defects shall be corrected. After the installation of the basket, each basket compartment will be checked to ensure that the fuel assemblies will fit in the basket.

Section 8.1.5 of the application describes the neutron shielding material and the tests to which the material is subjected. The neutron shield is comprised of a proprietary borated polyester resin compound. This resin is mixed and poured into long aluminum containers, as described in the application and discussed in Chapter 5 of this SER. Density tests are performed on every mixed batch to ensure a minimum mixture density of 1.547 g/cm^3. Chemical analyses are performed on the first batch mixed with a given set of components and any time a new lot of one of the major components is introduced to ensure proper composition.

The staff finds that, in addition to the foregoing tests, an acceptance test of the as-fabricated casks is needed to ensure proper performance (such as uniformity of pour, meaning lack of voids and streaming paths in the material, there is no settling out of components, etc.) of the shielding in the as-fabricated cask configuration. This is typically done with a dose measurement survey over the entire cask surface over which the neutron shield extends, the results of which are compared to calculated/predetermined dose rate values for the test radiation source with the as-designed shielding. A periodic test to ensure continued proper neutron shield performance is also needed. This test would be performed in a manner similar to the acceptance test, though not needing to be as extensive in coverage of the cask surface.

Due to the casks that are the subject of this certificate being already loaded under a 10 CFR Part 72 license, the applicant proposed using the pre-shipment dose rate measurements to demonstrate that the as-fabricated neutron shield is performing as designed. The applicant also noted the dose rate measurements done on the cask to comply with the 10 CFR Part 72 license requirements as demonstrating the adequacy of the as-fabricated shielding. In general for transportation packages, pre-shipment measurements only demonstrate that a particular packaging together with its contents for a particular shipment meets the limits of 10 CFR 71.47 at the time of shipment and can therefore be shipped. Acceptance tests, which are needed to comply with 10 CFR 71.85(a), are designed to verify the as-fabricated shield's performance by means of comparing measured dose rates for a given radiation source(s) in the package with pre-determined, or calculated, dose rates for the same source in the as-designed package. Additionally, measurements for 10 CFR Part 72 license requirements do not cover the entire cask surface and are based upon design-basis contents (as defined for the storage license application) and not the contents for which the measurements are made. Thus, they only give assurance that the cask and its contents do not exceed the design-basis dose rates at selected locations and thus provide reasonable assurance that 10 CFR Part 72 limits are met.

Thus, to address shielding tests in the case of the current application, the applicant modified its proposed operations descriptions to explicitly state that the pre-shipment measurements are taken over the entire cask surface. Additionally, it is proposed that the casks be limited to a single use. That is, they will only be used to ship the contents they currently contain (as they were loaded for use in storage). Along with these conditions, a condition that limits this CoC to only the limited number of casks currently loaded at PINGP is added to the CoC. Based upon these conditions, the staff finds that the pre-shipment measurements will meet the intent of the acceptance shielding tests (complying with 10 CFR 71.85(a)) and periodic shielding tests and are thus acceptable for those purposes. Staff does note, however, that any changes that may be sought under future amendment/revision requests for this CoC that modify the foregoing conditions (e.g., include additional casks, allow contents changes in the casks) will necessitate performance of neutron shielding acceptance and periodic tests of the as-fabricated shielding for the affected casks (e.g., casks added later to this CoC will need these tests). The shielding effectiveness of the other cask features is ensured by the other acceptance tests described in Chapter 8 of the application, such as the verification of dimensional and fabrication compliance of the cask's steel components with the specifications of the licensing drawings referenced in the CoC. Based upon its review and the foregoing considerations, staff finds the acceptance and periodic tests for shielding to be acceptable to demonstrate (continuing) package shielding effectiveness.

Wet chemical analysis and/or neutron attenuation testing will be performed to verify the minimum 10 mgram/cm^2 of B-10 is met for the neutron poison plates in the TN-40 basket. The staff reviewed Section 8.1.6 of the SAR and finds this requirement is not clearly stated.

To ensure the adequacy of cask thermal performance, the thermal survey of the TN-40 cask will be performed as described in Chapter 3, Section 3.4.7, prior to shipment. This section describes comparing the result of a temperature survey performed on each loaded cask to calculate outer shell temperatures from SAR thermal model analysis with appropriate adjustments for decay heat and ambient temperature.

The staff reviewed the descriptions and acceptance criteria of the acceptance tests, containment boundary leakage tests, and component tests. Based on its review, the staff finds that the acceptance tests and acceptance criteria are acceptable and that they provide a reasonable assurance that the requirements of 10 CFR 71.87 are satisfied.

8.2 Maintenance Program

Section 8.2 of the SAR specifies a maintenance program for the package. The TN-40 casks are used as a storage cask prior to their use as a transport cask. If a loaded cask is taken from storage and prepared for transport, no load testing beyond the initial fabrication load test is required prior to shipment.

After lid or vent/drain port cover removal, the affected metallic containment seals shall be replaced and the entire containment boundary shall be leak tested prior to shipment to show a leak rate less than $1x10^{-4}$ ref-cm^3/sec. Because these seals are used only once, the pre-shipment leak tests may be used to fulfill the ANSI N14.5 requirements for maintenance and periodic testing. The metallic seals may be reused for transport of an empty TN-40 packaging and therefore no leak tests are required prior to shipment of an empty TN-40 packaging.

All fasteners, seal surfaces, impact limiters, and fuel control structures will be visually examined before each shipment for any damage, and, if necessary, shall be removed from service for repair or replacement.

No periodic tests or inspections are required for the TN-40 shielding or heat transfer components. However, radiation and thermal surveys will be performed prior to transport as previously described in Chapter 7 of this SER. The comprehensive pre-shipment radiation surveys fulfill the intent of periodic shielding tests given the considerations described in Section 8.1 of this SER.

The CoC has been conditioned to specify that the package be acceptance tested and maintained in accordance with Chapter 8 of the SAR, as supplemented.

8.3 Evaluation Findings

The staff reviewed the TN-40 spent fuel storage and transportation package for the PINGP. The staff reviewed the information and commitments for acceptance tests and the maintenance program for the TN-40 package provided in the Safety Analysis Report. The staff followed the guidance provided in the "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617, in its review. Based on the statements and representations in the application, the staff concludes that the acceptance tests and maintenance program have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

CONDITIONS

Based on staff evaluation in Chapter 2-8 of this SER, the following conditions are highlighted:

1. 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,
2. As part of the preparation for transport, the 48 as-installed 1.375-in. diameter SA-320 Grade LA43 closure lid bolts are replaced by the SA-540 Grade B23 Class 1 bolts of the same configuration,
3. 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,
4. 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 5.0×10^{-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,

5. 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 SAR, with appropriate adjustments for decay heat and ambient temperature. The temperature difference between calculated and measured values shall not exceed ±25°F,
6. 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, and
7. For casks that are configured for storage, the operating procedures prescribed in Section 7.4 must be used to convert the storage configuration to the transportation configuration of the package.

CONCLUSION

Based on the statements and representations in the application, as supplemented, and the conditions listed above, the staff concludes that the design has been adequately described and evaluated and the package meets the requirements of 10 CFR Part 71.

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