UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION REPORT
Docket No. 71-9389
Model No. TN-40HT Package
Certificate of Compliance No. 9389
Revision No. 0

SUMMARY

By letter dated November 30, 2021 (Agencywide Documents Access and Management System [ADAMS] Package Accession Nos. ML21334A171, ML21334A172, ML21334A173, ML21334A174, ML21334A175), TN Americas LLC (TN or the applicant) submitted an application for approval of the Model No. TN40-HT as a transport package. The storage application of the TN-40HT had been previously approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a site-specific license, under Docket No. 72-10.

On August 1, 2023, TN responded to staff's request for additional information (RAI) (ML23213A163, ML23213A165, ML23213A167, ML23213A169, ML23213A170) and subsequently submitted on December 4, 2023, a supplemental response (ML23338A320, ML23338A321, ML23338A322). TN also provided a revised and updated application on December 7, 2023 (ML23341A089).

NRC staff reviewed the application, as supplemented, using the guidance in NUREG-2216. The package was evaluated against the regulatory standards in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, including the general standards for all packages and the performance standards specific to fissile material packages under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

The analyses performed by the applicant demonstrate that the package provides adequate protection to meet structural, thermal, containment, shielding, and criticality requirements after being subject to the tests for NCT and HAC.

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

EVALUATION

1.0 GENERAL INFORMATION

The Model No. TN-40HT package, designed to transport up to 40 pressurized water reactor (PWR) spent fuel assemblies, consists of a basket assembly, a containment vessel, a package body which also functions as the gamma shield and neutron shield, and impact limiters. A transport frame, which is not part of the packaging, is used for tie-down purposes.

The basket structure consists of an assembly of stainless-steel cells joined by a fusion welding process and separated by aluminum and poison plates which form a sandwich panel.

The panel consists of two aluminum plates separated by a poison plate. The aluminum plates provide the heat conduction paths from the fuel assemblies to the cask inner plate. The poison material provides the necessary criticality control. The opening of the cells is 8.05-inch (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 vessel components consist of the inner shell and bottom inner plate, shell flange, lid outer plate, lid bolts, penetration cover plates and bolts (vent and drain), and the inner metallic seals of the lid seal and the vent and drain seals. The containment vessel prevents leakage of radioactive material from the cask cavity and also maintains an inert atmosphere (helium) in the cask cavity. The overall containment vessel length is approximately 175.0 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a nominal diameter of 72.0 in. and a length of 163 in.

The carbon steel packaging body, which also functions as the gamma shielding, is around the inner shell and the bottom inner plate of the containment vessel. The 7.25 in. gamma shield completely surrounds the containment vessel shell and bottom plate, respectively. A 5.50 in. thick shield plate is also welded to the inside of the 4.50 in. thick lid outer plate.

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

Radial neutron shielding is provided by a borated polyester resin compound surrounding the gamma shield shell. The total radial thickness of the resin and aluminum is 5.25 in. The array of resin-filled containers is enclosed within a 0.50 in. thick outer steel shell. The aluminum container walls also provide a path for heat transfer from the gamma shield shell to the outer shell. A pressure relief valve is mounted on top of the resin enclosure to limit the possible internal pressure increase under HAC.

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

The nominal external dimensions of the package, with impact limiters, are 261 in. long by 144 in. wide. The total weight of the package is 275,000 pounds (lb.).

The package is constructed and assembled in accordance with the following TN Americas LLC, Drawing numbers:

Drawing No	Title
TN40HT-71-1, Rev. 0	TN-40HT High Burnup Transport Cask Parts List and Notes (1 sheet)
TN40HT-71-2, Rev. 0	TN-40HT High Burnup Transport Cask Transport Configuration (2 sheets)
TN40HT-71-3, Rev. 0	TN-40HT High Burnup Transport Cask Shell Assembly (2 sheets)
TN40HT-71-4, Rev. 0	TN-40HT High Burnup Transport Cask Lid Assembly and Details (1 sheet)
TN40HT-71-5, Rev. 0	TN-40HT High Burnup Transport Cask Lid Details (1 sheet)
TN40HT-71-6, Rev. 0	TN-40HT High Burnup Transport Cask Shell Assembly Details (1 sheet)
TN40HT-71-7, Rev. 0	TN-40HT High Burnup Transport Cask Basket Assembly and Details (6 sheets)
TN40HT-71-8, Rev. 0	TN-40HT High Burnup Transport Cask Basket Rails (8 sheets)
TN40HT-71-9, Rev. 0	TN-40HT High Burnup Transport Cask Impact Limiter Radial Spacer (1 sheet)
10421-71-7, Rev. 3	TN-40 and TN-40HT Transport Packaging Impact Limiter Spacer Details (1 sheet)
10421-71-40, Rev. 2	TN-40 and TN-40HT Transport Packaging Impact Limiters General Arrangement (1 sheet)
10421-71-41, Rev. 2	TN-40 and TN-40HT Transport Packaging Impact Limiters Parts List and Notes (1 sheet)
10421-71-42, Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Assembly (1 sheet)
10421-71-43, Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Details (1 sheet)
10421-71-44, Rev. 1	TN-40 and TN-40HT Transport Packaging Impact Limiters Parts (1 sheet)

The staff verified that the drawings include the information described in NUREG-2216 on the:

(1) materials of construction,
(2) dimensions and tolerances,
(3) codes, standards, or other specifications for materials, fabrication, examination, and testing
(4) welding specifications, including location and nondestructive examination (NDE),

(5) coating specifications and other special material treatments that perform a safety function and
(6) specifications and requirements for alternative materials.

The Criticality Safety Index of the package is 0.

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

2.0 STRUCTURAL AND MATERIALS EVALUATION

2.1 STRUCTURAL EVALUATION

The applicant submitted an application for the TN-40HT package, a new Type B fissile transportation package (reference 1), with an initial safety analysis report (SAR), Rev. 0A, as Enclosure 2 of the application. The objective of the structural evaluation is to verify that the structural performance of the package meets the regulatory requirements of 10 CFR Part 71.

2.1.1 Description of Structural Design

The applicant provided the description of the TN-40HT package in section 1.2, “Package Description,” of the SAR. The TN-40HT package consists primarily of the cask body, fuel basket, and impact limiters. The dimensions, materials and tolerances of the components are provided in the licensing drawings in section 1.6.4, “TN-40HT Packaging Drawings,” of the SAR. Table 1-1 of the SAR tabulates overall dimensions and weights of the components and table 1-2 of the SAR presents the nominal design dimensions and specifications for the fuels.

The containment vessel within the cask body is comprised of inner shell, bottom inner plate, shell flange, lid outer plate, lid bolts, cover plates for the vent and drain, and seals for the lids and cover plates. Surrounding the containment vessel is the shield shell which provides both neutron and gamma shielding and is made of aluminum, resin, and steel. The gamma shield is provided around the inner shell and inner bottom plate of the containment vessel by an independent carbon steel shell, surrounding the containment vessel. The neutron shield is comprised of resin contained in aluminum alloy tubes.

The closure system of the package is comprised of the lid outer plate and associated bolts to provide necessary closure. Metal seals are energized in turn by the lid when lid bolts are pre-tightened. This system is used to prevent leakage into the containment vessel, while maintaining a pressurized helium atmosphere within. The outer lid plate contains two penetrations for the vent and drain which have their own metal seals and covers. The maximum normal operating pressure of the TN-40HT is 14.7 pounds per square inch gauge (psig) and is cooled passively.

The fuel basket is an assembly from several plates in an egg crate structure. Its main function is to transfer heat while providing neutron absorption to maintain criticality requirements. The basket structure is mainly comprised of stainless-steel plates joined by a proprietary fusing process. Aluminum rail inserts surround the exterior basket cells and provide the heat conduction paths from the fuel assemblies to the cask inner plate, while poison material furnishes key criticality control.

The impact limiters consist of thin stainless-steel shells that encase balsa and redwood which attach to the cask body. They are designed to fit over the trunnions and are connected with 13 tie-rods to keep the impact limiters attached to the cask. Each impact limiter has fusible plugs that melt and relieve excessive internal pressure during the thermal test. A top spacer is placed between the top lid cask plate and a radial spacer exists between each impact limiter and cask body.

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (reference 3) is used for both design and fabrication of most components of the TN-40HT such as the containment vessel and basket. The containment vessel is designed, fabricated, examined, and tested in accordance with the requirements of subsection NB of the ASME Code. Subsection NG is used to design the basket structure, while Subsection NF is used to design the trunnions.

Two upper and lower trunnions are used to load the TN-40HT package in the vertical configuration and transported in the horizontal orientation on a specially designed shipping frame which is not considered to be part of the transportation package. The lower trunnions are only used to upend the cask while the upper trunnions are designed for lifting. The cask lid has threaded holes for lifting but cannot be accessed when the impact limiters are attached to the transportation package. Each impact limiter has a lifting lug.

Centers of gravity of components such as the cask body, fuel basket, and impact limiters are provided in table 2-1 of the SAR. The maximum payload of the TN-40HT is 51,000 lb.

The staff reviewed the structural design description of the package and determined that the contents of the application satisfy the requirements of 10 CFR 71.31(a)(1)(c), 10 CFR 71.31(a)(2), 10 CFR 71.33(a), and 10 CFR 71.33(b).

2.1.2 Identification of Codes and Standards for Package Design

The applicant used the ASME B&PV Code (reference 3) to design and fabricate most components. Specifically, the components of the cask containment vessel (i.e., inner shell, flange, bottom inner plate, lid, lid bolts, lid seals, drain and vent port cover plates, cover plate seals, and bolts, etc.) are designed with the ASME B&PV Code, section III, subsection NB. Additionally, the components are designed to meet the requirements of Regulatory Guides (RG) 7.6 (reference 4) and RG 7.8 (reference 5). Alternatives to the ASME Code are listed in appendix 1.6.3 of the SAR.

In addition, structures such as the shield shell and neutron shield are designed and fabricated according to the ASME Code, subsection NF, while welding follows section IX of the ASME Code (reference 6). The basket is designed according to the ASME Code, subsection NG. Trunnions are designed according to 10 CFR 71.45 and ANSI N14.6 (reference 7). The staff reviewed the codes and standards used for the package design and found them acceptable.

The staff determined that the package satisfies the regulatory requirements of 10 CFR 71.31(c).

2.1.3 General Requirements for all packages

Minimum Package Size

The applicant indicated that the smallest dimension of the package is greater than 4 in., which is larger than the requirement of 10 CFR 71.43(a).

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.43(a).

Tamper-Indicating Feature

The staff reviewed the package description and confirmed that the only access path into the package is through the closure lid and associated lid closure bolts, but they are completely covered, and the path is prevented by the presence of the front impact limiter during transport. In addition, a wire security seal is installed in the front impact limiter. The presence of this seal indicates that unauthorized opening of the package has not occurred. This tamper-indication feature meets the requirement of 10 CFR 71.43 (b).

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.43(b).

Positive Closure

The staff reviewed the package closure description and found that the positive closure of all openings through the containment vessel is accomplished by bolted closures.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.43(c).

Package Valve

10 CFR 71.43(e) requires that a package valve must be protected against unauthorized operation. The staff reviewed the package description and found that the TN-40HT does not have any valves or other device whose failure would allow for the escape of radioactive material.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.43(e).

2.1.4 Lifting and Tie-Down Standards

Lifting Devices

The package has two upper trunnions and two lower trunnions which are welded to the cask body. The lower trunnions are used to upend and rotate the cask while the upper trunnions are used for lifting.

ANSI N14.6 requirements do not apply to the lower trunnions because they do not transmit the load to the hook of an overhead hoisting system. The lower trunnions were evaluated with a dynamic load factor of 1.15, and an applied load that is 5 times the maximum intended load; the applicant examined combined stresses in the trunnions for shear and bending and found them to be less than the allowable stress of the material for the lower trunnions and found a margin of safety to be 0.31 for the ultimate condition, and the minimum margin of safety to be 0.39 (yield condition).

The applicant evaluated the potential failure of a trunnion to determine if such a failure would prevent the cask from meeting other requirements of 10 CFR Part 71. Based on a stress evaluation, the applicant determined that a higher margin of safety exists in the surrounding cask body (table 2.11.13-1 of the SAR) as compared to the trunnion weld or trunnion material. Thus, the trunnion or its weld will fail prior to the cask body, leaving the cask body unimpaired from performing its other requirements.

Based on the applicant's analysis for lifting, the staff concluded that the calculated safety factor is greater than 3 with respect to yielding, thereby, the TN-40HT package meets the requirements of 10 CFR 71.45(a) for lifting.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.45(a).

Tie-Down Devices

The applicant described tie-down of the package in sections 1.2.1.4 and 2.3.2 of the SAR. The applicant stated that there are no tie-down devices that are structural part of the cask. The longitudinal forces experienced by the transport cask are resisted by steel end restraints with the impact limiters to prevent movement in the direction of travel. The vertical and lateral forces that act on the transport cask are restrained by a dual saddle/strap tie-down system. Specifically, the tie-down straps resist uplifting and lateral overturning forces whereas the saddles react downward and strap reaction forces. This restraint system is designed to preclude yielding in the load bearing material of the transport cask during NCT.

The staff reviewed the applicant's statement and confirmed that there are no tie-down devices that are structural part of the cask and concluded that the tie-down requirements for the package are according to 10 CFR 71.45(b).

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.45(b).

2.1.5 General Considerations for Structural Evaluation

The applicant performed evaluations for the TN-40HT package using the finite element (FE) method with the computational modeling tools (ANSYS and LS-DYNA). Physical testing was not done on the package. However, the applicant made several comparisons to the licensed TN-40 package to validate computational models of the TN-40HT, where the licensed TN-40 package was previously evaluated by the physical model testing and the FE method using the ANSYS and LS-DYNA computer programs (reference 8).

LS-DYNA Model

The applicant created an LS-DYNA model to calculate the maximum strain of a fuel rod under the HAC end drop impacts, as discussed in section 2.11.9.3 of the SAR. Validation of the LS-DYNA model for the drop analysis was achieved through a comparison of the results provided in reference 9. Additional details of benchmarking are based on work previously done for the TN-40 package (reference 8). The LS-DYNA model used in this TN-40HT package and the differences from the benchmark model are described in section 2.11.9.3.4 of the SAR.

The acceptance criteria for this model was that the fuel rod cladding had to have maximum principal strains that were less than elastic strains for irradiated fuel cladding properties. Additional details of benchmarking are based on work previously done for the TN40 package and an internal pressure of 1060 pounds per square inch (psi) was assumed with fuel cladding mechanical properties described in NUREG 2224, and also assuming a 1 percent (%) post yield hardening slope.

The staff reviewed the model descriptions and found that the LS-DYNA model is adequately developed to analyze the performance of the TN-40HT package under HAC and concluded that the LS-DYA model is acceptable.

ANSYS Model

The applicant created a ANSYS FE model consisting of a three-dimensional (3-D), 180 degrees (°) symmetry sector with appropriate symmetry boundaries based on the licensing drawings and used for the structural analyses. The model only contained structural components (i.e., lid shield plate, outer plate, shell flange, inner shell, bottom inner plate, shield shell, and the bottom shield, etc.). The solid element types were used for linearly elastic materials, which are temperature dependent. The lid bolts were modeled using the beam elements. A prestress of 50 kilo per square in. was applied to the primary lid bolts. Temperature distributions from NCT thermal evaluations in chapter 3 of the SAR were mapped into a structural model node configuration. Buckling analyses were performed assuming a non-linear elastic plastic behavior.

For side drop analyses, bilinear elastic-plastic material properties of the structural basket were used when evaluating stresses for 0°, 30°, 45°, 60°, and 90° drop orientations. For the corner 30-ft. drop, an elastic FE analysis was performed to determine the status of the lid/cask seal.

In addition, single fuel rod was modeled using pipe elements where the fuel weight was incorporated by adjusting the density of the cladding and was modeled as being constrained laterally at grid spacers. A modal analysis of the fuel was also performed to support the dynamic analyses.

The applicant performed dynamic analyses in ANSYS for drop scenarios by applying dynamic load factors to quasi-static stress analyses based on the natural frequency of the cask. The dynamic load factors are described in section 2.11.8.3.2 and section 2.11.8.4.1.3 for NCT and HAC respectively. these factors and are based on impact duration and modal analyses.

The staff reviewed the model descriptions and found that the ANSYS model is adequately developed to analyze the closure device under NCT and HAC and concluded that the ANSYS model is acceptable.

Conclusion

The staff reviewed the information provided in the TN-40HT SAR and associated appendices for package modeling and analyses. The staff concluded that the LS-DYNA and ANSYS models adequately presented the package geometry and the mass distribution of the components for performing NCT and HAC drop analyses.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.41(a).

2.1.6 Normal Conditions of Transport

The applicant evaluated the TN-40HT package for NCT of heat, cold, reduced external pressure, increased external pressure, vibration and fatigue, water spray, free drop, corner drop, compression, and penetration as required by 10 CFR 71.71.

Heat

The applicant performed thermal analyses for the TN-40HT package and presented the evaluation findings in chapter 3, "Thermal Evaluation," of the SAR.

The applicant used steady state environmental conditions corresponding to the maximum daily averaged ambient temperature of 100 degrees Fahrenheit (°F) and the 10 CFR 71.71(c)(1) insolation averaged over a 24-hour (hr.) period to determine the maximum fuel cladding temperature, TN-40HT component temperatures, and containment pressure. These temperatures were applied to the ANSYS finite element model for the evaluation of thermal stresses which considered spatial temperature gradients and differences in material thermal expansion rates and showed there is adequate clearances between components with respect to thermal expansion. This case incorporates maximum ambient temperature, maximum fuel decay heat, and maximum insolation.

The applicant applied the temperatures to the ANSYS FE model as described in appendix 2.11.1 of the SAR. Table 2.11.1-5 of the SAR provides a summary of the calculated load combination stresses for the structural components under NCT. The table also provides calculated factors of safety above 1.0 when they are compared with the allowable stresses indicating that the heat requirements for the package are met.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.71(c)(1).

Cold

10 CFR Part 71.71(c)(2) requires that the package is subjected to an ambient temperature of -40°F in still air and shade.

The applicant performed thermal analyses for the TN-40HT package subjected to cold environment conditions (ambient temperature -40°F). Temperatures from the thermal analyses of chapter 3 of the SAR were applied to the ANSYS FE model for the calculation of thermal stresses. Table 2.11.1-5 of the SAR tabulates the stresses and presents the calculated factors of safety above 1.0 indicating that the components of the TN-40HT package are safe and operational.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.71(c)(2).

Reduced External Pressure

10 CFR Part 71.71(c)(3) requires that the package is subjected to a reduced external pressure of 3.5 pounds per square inch absolute (psia).

The applicant evaluated the TN-40HT package subjected to a reduced external pressure of 3.5 psia. The ANSYS FE model was used to calculate the stresses of the components and the evaluations are documented in appendix 2.11.1 of the SAR. Table 2.11.1-5 of the SAR tabulates the stresses and presents the calculated factors of safety above 1.0 indicating that the reduction pressure as specified in 10 CFR Part 71.71(c)(3) will not affect the performance of the TN-40HT.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.71(c)(3).

Increased External Pressure

10 CFR Part 71.71(c)(4) requires that the package is subjected to an external pressure of 20 psia.

The applicant evaluated the package with a 20 psia ambient pressure using the ANSYS FE model. The evaluations are documented in appendix 2.11.1 of the SAR. Table 2.11.1-5 of the SAR tabulates the stresses and presents the calculated factors of safety above 1.0 indicating that an external pressure of 20 psia will not affect the performance of the TN-40HT.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.71(c)(4).

Vibration and Fatigue

The applicant performed vibration and fatigue analyses for the TN-40HT package based on the study of the vibration and shock effects in NUREG 766510 (reference 13). Table 2.11.1-5 summarizes the calculated factors of safety which are shown to be above 1.0.

The applicant also calculated the fatigue cycles on the containment boundary from a combination of various sources (rail car shock, test pressure, lifting etc.) and evaluated their cumulative effects. Table 2.11.7-1 of the SAR provides the calculated total damage factor, which is less than 1.0 indicating that the containment boundary of the TN-40HT is adequate with respect to the fatigue.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.71(c)(5).

Water Spray

10 CFR Part 71.71(c)(6) requires that the package must be subjected to a water spray test that simulates exposure to rainfall of approximately 2 in./hr. for at least 1 hr.

The applicant stated that all exterior surfaces of the TN-40HT body are metal and, therefore, not subject to soaking or structural degradation from water absorption including the impact limiters which are fully encased in metal. The staff reviewed the statement and agreed that the water spray will not impair the package.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71. 71(c)(6).

Free Drop

The applicant examined two 1-foot drops: end drop and side drop. Appendices 2.11.4, 2.11.8, and 2.11.9 examine the lid bolts, basket, and fuel assemblies, respectively. Table 2.11.1-5 tabulates these loads cases and shows the calculated factors of safety to be above 1.0 for the shell flange, inner shell and bottom, lid outer plate, lid shield, shield shell and bottom shield plate.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.71(c)(7).

Corner Drop

The applicant stated that the corner drop test does not apply since the TN-40HT has a mass in excess of 220 lb. As a result, 10 CFR 71.71(c)(8) is not applicable.

The staff determined that the regulatory requirements of 10 CFR 71.71(c)(8) are not applicable to the TN-40HT package.

Compression

The applicant stated that the compression test does not apply since the TN-40HT has a mass of in excess of 11,000 lb. As a result, 10 CFR 71.71(c)(9) is not applicable.

The staff determined that the regulatory requirements of 10 CFR 71.71(c)(9) are not applicable to the TN-40HT package.

Penetration

10 CFR Part 71.71(c)(10) requires that impact of a hemispherical end of a vertical steel cylinder of 1.25 in. diameter and 13 lb. mass, dropped from a height of 40 in. onto the exposed surface of the package, is expected to be most vulnerable to puncture.

The applicant stated that due to lack of external protuberances, the 40 in. drop of a 13 lb. bar has a negligible effect on the TN-40HT. The staff agrees that the TN-40HT package is not susceptible to the 13 lb. bar and concludes that the package meets the regulatory requirements of 10 CFR 71.71(c)(10).

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.71(c)(10).

2.1.7 Hypothetical Accident Conditions

The applicant evaluated the TN-40HT package for HAC of free drop, crush, puncture, thermal, and water immersion as required by 10 CFR 71.73.

Free Drop

The applicant analyzed four 30-foot drops using the LS-DYNA model: (i) 90^{o} end drop, (ii) 0^{o} side drop, (iii) center of gravity over corner drop, and (iv) 20^{o} slap-down. The applicant provided the results of the stress analyses for the containment boundary and its bolts in appendices 2.11.2 and 2.11.4 of the SAR. The applicant also provided the calculated stresses for the basket and fuel cladding in appendices 2.11.8 and 2.11.9. Table 2.11.2-3 of the SAR tabulates a list of the load combinations and table 2.11.2-5 shows the calculated stresses for the load cases for the shell flange, lid outer plate, shield shell, inner shell, bottom shield, and lid shield plate. Table 2.11.2-5 presents the calculated factors of safety, which are larger than 1.0.

Additionally, the applicant calculated principal strains of the components using the LS-DYNA model for the 30 feet (ft.) end drop scenario. The calculated maximum principal strain was found to be 0.571%, which is less than the yield strain of 0.94%. For the side drop scenario, the applicant found the fuel cladding to have an axial stress of 78,017 psi, which is lower than the yield strength of 95,800 psi for Zircaloy-4 cladding. Based on the review of the applicant's analyses, the staff concluded that the TN-40HT package meets the requirements of 10 CFR 71.73(c)(1) for free drop.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.73(c)(1).

Crush

The applicant stated that the crush test does not apply since the mass of the TN-40HT is in excess of 1,100 lb. As a result, 10 CFR 71.73(c)(2) is not applicable.

The staff determined that the regulatory requirements of 10 CFR 71.73(c)(2) are not applicable to the TN-40HT package.

Puncture

The applicant evaluated the puncture drop in appendix 2.11.5 of the SAR for the TN-40HT shield shell between the impact limiters since the impact limiters protect the ends of the package. The LS-DYNA FE drop analyses showed that there is no damage to the shield shell. The staff finds that the TN-40HT package meets the requirements of 10 CFR 71.73(c)(3) for puncture.

The applicant also considered an additional puncture scenario in response to a RAI, where an impact limiter of the transportation package strikes a long puncture bar (18 ft. +) in a near end drop configuration. This was done to prove that the impact limiter could not be stripped off the package which would make it susceptible to the subsequent fire scenario as part of cumulative damage.

The applicant considered that the impact limiter striking the puncture bar was also damaged previously from the 9m slap down scenario, and that several tie rods and bolts would be rendered ineffective as a result. The applicant demonstrated that the remaining tie rods and bolts that attach the impact limiter to the package remain effective and thus the impact limiters remain attached to the package.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.73(c)(3).

Thermal

The applicant described the thermal analyses of the TN-40HT subjected to thermal fire accident case in chapter 3 of the SAR and incorporated them into the structural analyses in appendix

2.11.2 of the SAR. The applicant assumed material properties used in the analyses are taken at 750°F, which bounds the maximum temperatures of the containment boundary for HAC as per Table 3-3 in chapter 3 of the SAR.

The applicant conservatively used an internal pressure of 125 psig instead of the actual 70.2 psig as shown in section 3.4.3.2 of the SAR. The staff reviewed the evaluations provided in the SAR and concluded that the TN-40HT package meets the thermal requirements of 10 CFR 71.73(c)(4).

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.73(c)(4).

Immersion - Fissile Material

The applicant stated that the immersion test for fissile material requirements of 10 CFR 71.73(c)(5) is covered by the requirements of 10 CFR 71.73(c)(6) because the cask body stresses for this immersion condition with a head of water at 3 ft. (1.3 psi external pressure) are bounded by the immersion condition for all packages (water head at least 50 ft., 21.7 psig external pressure) as described in section 2.6.6 of the SAR. The staff agreed that the requirements of 10 CFR 71.73(c)(5) will be covered by the requirements of the10 CFR 71.73(c)(6) for the TN-40HT package.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.73(c)(5).

Immersion - All Packages

The applicant noted that the TN-40HT according to 10 CFR 71.61 for Type B packages contains more than 10^5 A_2. Thus, the applicant applied a pressure of 290 psi to the package body as described in appendix 2.11.2/2.11.11 of the SAR using an ANSYS finite element model and hand calculations. Load combination A2 represents this case and is shown to have a minimum factor of safety above 1 in table 2.11.2-4 of the SAR. Buckling was also evaluated for this scenario using an ANSYS nonlinear elastic-plastic finite element model.

The applicant also used ASME Code Case N-284 to demonstrate that the inner shell will not buckle assuming conservatively that the outer shield shell will not contribute as described in 2.11.11 of the SAR. Specifically, the applicant determined that the plastic amplified stress for hoop compression under the immersion accident condition to be 41,054 psi which is less than the theoretical buckling stress of 45,771 psi. With respect to axial compression, the plastic amplified stress is 13,308 psi which is less than the theoretical buckling stress of 659,327 psi.

The applicant demonstrated that no interaction checks between axial and hoop compression is required. Thus, the applicant determined that the inner shell can withstand the combined external pressure due to fabrication stress and immersion pressure without buckling.

The staff reviewed the structural evaluations of the immersion test and concluded that the TN-40HT package meets the requirements of 10 CFR 71.73(c)(6).

Air Transport Accident Conditions for Fissile Material

The applicant stated that this test does not apply to the TN-40HT package since the package will not be transported by air.

The staff determined that the regulatory requirements of 10 CFR 71.55(f) are not applicable to the TN-40HT package.

Special Requirements for Type B Packages Containing More than 10^5 A_2

The applicant stated that the TN-40HT is a Type B package and contains more than 10^5 A_2 according to 10 CFR 71.61. The applicant evaluated the package for an external pressure of 290 psi as evaluated in Appendix 2.11.2 of the SAR. The staff reviewed the evaluation in Section 2.6.6 of this safety evaluation report (SER) above. Based on the review, the staff concluded that the TN-40HT package meets the requirements of 10 CFR 71.61.

The staff determined that the application satisfies the regulatory requirements of 10 CFR 71.61.

Air Transport of Plutonium

The applicant stated that the test does not apply to the TN-40HT package since the package will not be transported by air.

The staff determined that the regulatory requirements of 10 CFR 71.64 and 71.74 are not applicable to the TN-40HT package.

2.1.8 Conclusion for Safety Evaluation

The staff reviewed and evaluated the applicant's statements and representations in the application. Based on the review and evaluations, the staff concludes that the TN-40HT transportation package is adequately described, analyzed, and evaluated to demonstrate that its structural capability and integrity meet the regulatory requirements of 10 CFR Part 71.

References

1. "TN Americas LLC Application for Approval of the TN-40HT Transportation Package (Docket No. 71-9389)," Orano TN Letter E-59464, November 30, 2021.

2. 10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

3. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code, Section III, 2004 with 2006 addenda."

4. U.S. Nuclear Regulatory Commission (NRC), Regulatory Guide 7.6, Revision 1, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," 1978.

5. U.S. NRC, Regulatory Guide 7.8, Revision 1, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material," 1989.

6. ASME, B&PVC, Section IX, Division 1, "Welding and Brazing Qualifications, 2004 through 2006 addenda."

7. American National Standards Institute (ANSI) N14.6-1993, “Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or More,” January 1993.

8. Orano TN, “TN-40 Transportation Packaging – Safety Analysis Report,” Revision 16. Docket No. 07109313.

9. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Volumes I-III, NUREG/CR-0200, Revision 5 (ORNL/NUREG/CSD-2/R5), March 1997 (Standard Composition Library, Table M8.2.4).

10. Harold E. Adkins, Jr., Brian J. Koeppel and David T. Tang, “Spent Nuclear Fuel Structural Response when Subject to An End Impact Accident,” PVP-Volume 483, Transportation Storage and Disposal of Radioactive Materials, July 25 through 29, 2004, San Diego, CA, USA.

11. U.S. NRC, NUREG-1864, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant,” March 2007.

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

13. U.S. NRC, NUREG 766510, “Shock and Vibration Environments for Large Shipping Containers on Rail Cars and Trucks,” 1977.

2.2 MATERIALS EVALUATION

The staff evaluated the material characteristics of the TN-40HT design for transportation of up to 40 Westinghouse (WE) and Exxon 14 x 14 PWR spent fuel assemblies.

2.2.1 Drawings

The drawings for the TN-40HT components are provided in SAR chapter 1, appendix 1.6.4. The drawings include a Bill of Materials that provides the material specification and the safety category of each component. Material alternatives, fabrication instructions, and additional material property requirements are provided in the drawing details and notes. The staff reviewed the drawings using the guidance in NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material,” issued August 2020, NUREG/CR-5502, “Engineering Drawings for 10 CFR Part 71 Package Approvals,” issued May 1999, and Regulatory Guide 7.9, “Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material,” for the recommended content of engineering drawings. In addition, the staff used NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” issued February 1996, and the NRC Regulatory Guide 7.10, "Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material," appendix A, “A Graded Approach to Developing Quality Assurance Programs for Packaging Radioactive Material,” for guidance on safety classification of transportation packaging components. The staff verified that the drawings included design features considered in the package evaluation, including:

- the containment system
- closure device
- internal supporting or positioning structures

- gamma shielding
- outer packaging
- heat-transfer features
- energy-absorbing features
- lifting and tie-down devices
- personnel barriers

The staff verified that the drawings include the information described in NUREG-2216 on the (1) materials of construction, (2) dimensions and tolerances, (3) codes, standards, or other specifications for materials, fabrication, examination, and testing (4) welding specifications, including location and NDE, (5) coating specifications and other special material treatments that perform a safety function and (6) specifications and requirements for alternative materials. The staff determined that the drawings for the package provide the necessary information identified in the NRC guidance documents and the engineering drawings provided by the applicant are consistent with the design and description of the package, in accordance with 10 CFR 71.33, "Package Description." Therefore, the staff determined that the drawings provided by the applicant were acceptable.

2.2.2 Codes and Standards

The materials codes and standards are described in SAR section 7.2.2. The TN-40HT is designed and constructed to ASME BPV Code, 2004 Edition including the 2006 Addenda. The containment boundary is designed and fabricated to Class 1 in accordance with the ASME BPV Code, section III, Division 1, subsection NB. Shielding components are designed, fabricated, and inspected to ASME BPV Code, section III, Division 1, subsection NF. Criticality components are designed, fabricated, and inspected to ASME BPV Code, section III, Division 1, subsection NG. The alternative provisions to the ASME Code are described in SAR appendix 1.6.3. The TN-40HT transport cask is designed and constructed with Code Case N-284-4 in SAR appendix 2.11.11.

Non-code components such as impact limiters, neutron shielding, and non-pressure retaining cover plates are designated non-code in the drawing parts list.

The staff reviewed the applicable codes and standards and material specifications for the TN-40HT transportation packaging components. The staff determined that the use of ASME BPV Code, section III, Division 1, subsection NB, NF, and NG for the containment boundary is consistent with the NRC guidance in NUREG-2216 which references NUREG/CR-3854, "Fabrication Criteria for Shipping Containers." Table 4.1 of NUREG/CR-3854 provides guidance for use of ASME B&PV Code section III, Division 1, criteria for the fabrication of containment, criticality, and other safety components for Category I containers such as those that transport SNF. The staff reviewed the Code Alternatives identified in SAR appendix 1.6.3. The staff determined that the applicant described and provided a basis for the code alternatives for the TN-40HT transportation package in SAR appendix 1.6.3. The staff has previously determined ASME Code Case N-284-4 to be acceptable as documented in the NRC Regulatory Guide 1.84 Revision 39 issued December 2021 (ML21181A225). Therefore, the staff determined that the description of the codes and standards applicable to the TN-40HT package provided by the applicant was acceptable.

2.2.3 Weld Design and Inspection

The welding design and inspections are described in SAR section 7.3 and appendix 1.6.1 and 1.6.3. The welding procedures, welders, and welding operators are qualified in accordance with ASME section III, NB-4000, NG-4000, or NF-4000 for the confinement boundary, basket, and shield and trunnion. NDE specifications and procedures are in accordance with section V of the ASME Code. Personnel performing NDE will be qualified in accordance with ASME Section III.

The staff evaluated the welding practices and NDE specifications and found them to be in accordance with ASME Code section III and section V.

Confinement welds are inspected in accordance with ASME Code subsection NB, including the alternatives to the ASME Code as specified in SAR appendix 1.6.3.

Non-confinement welds are inspected in accordance with ASME Code subsections NF or NG including alternatives to the ASME Code as specified in SAR appendix 1.6.1.

The TN-40HT containment boundary components are shown in Drawings TN-40HT-71-3, "Cask Shell Assembly, " TN-40HT-71-4, "Cask Lid Assembly and Details," and TN-40HT-71-5, "Cask Lid Details."

The staff reviewed the confinement welds and finds they are performed in accordance with ASME Code NB. The staff reviewed the non-confinement welds and finds they are performed in accordance with ASME Code NF or NG. Therefore, the staff finds the confinement and non-confinement welds to be acceptable.

The TN-40HT basket is designed and fabricated to ASME Code NG-3200, including alternatives to the ASME Code specified in SAR appendix 1.6.3. The staff reviewed the basket and found it to be designed and fabricated in accordance with ASME Code NG-3200 and is therefore acceptable.

2.2.4 Mechanical Properties

The applicant provided a description of the mechanical properties of the packaging materials in SAR Section 7.4, which included tensile properties, fracture resistance, tensile properties and creep of aluminum alloys at elevated temperatures, and the properties of impact limiter materials. The applicant provided tables in chapter 7 of the SAR listing the mechanical properties for the containment boundary, shielding materials, impact limiter materials, and materials used for the structural components of the TN-40HT transportation packaging. The applicant included material properties including elastic modulus, yield strength, ultimate strength, allowable stress, and thermal expansion as a function of temperature. For ASME BPV Code materials, the applicant cited the material property values included in the ASME BPV Code, section II, Part D and provided properties as a function of temperature.

The staff reviewed the material properties provided by the applicant and determined that the applicant provided the mechanical properties of the materials used for the containment boundary, shielding materials, impact limiter materials and materials used for the structural components of the TN-40HT transportation package. The staff reviewed ASME BPV Code materials properties as a function of temperature and determined that the properties provided by the applicant were acceptable because the applicant used the values provided in ASME BPV Code, section II, Part D. The staff determined that the temperature ranges for the mechanical properties provided by the applicant bound the range of the packaging component temperatures provided in chapter 3 of the SAR for NCT and HAC conditions. Therefore, the staff determined that the mechanical properties of the materials for the TN-40HT transportation package provided by the applicant were acceptable.

2.2.5 Thermal Properties

The applicant provided thermal properties of the materials in table 7-1 to 7-11 and table 7-13 and table 7-15 including thermal expansion coefficients, thermal conductivity, thermal diffusivity, and heat capacity. The applicant provided values of the thermal properties as a function of temperature obtained from the ASME BPV Code, section II, Part D. The staff determined that the thermal properties provided by the applicant were acceptable because they were supported by information included using ASME BPV Code, section II, Part D.

2.2.6 Radiation Shielding

The staff reviewed information provided by the applicant regarding neutron shielding materials for the TN-40HT transportation package. The applicant states that neutron shielding is provided by a borated polyester resin compound cast into long, slender aluminum tubes placed around the cask shell and enclosed within a smooth outer shell and is presented in table 7-13 and 7-17. In addition to serving as neutron shield containers, the aluminum also provides a conduction path of heat transfer from the cask body to the outer shell.

The staff reviewed information provided by the applicant regarding gamma shielding materials for the TN-40HT transportation package. The applicant states that gamma shielding SA-105 is primarily provided by the lid shield plate designed in accordance with subsection NF of the ASME Code and is presented in tables 7-11, 7-12, and 7-17. Additional gamma shielding is provided by the steel shell surrounding the resin layer and by the stainless steel and aluminum/steel structure of the basket.

2.2.7 Criticality Control

The staff reviewed the TN-40HT package to determine if the packaging design and the contents specified such that the package is subcritical under the design-basis conditions, NCT, and HAC, in accordance with 10 CFR 71.55(b), (d), and (e) and 10 CFR 71.59.

Neutron – Absorbing (Poison) Material Specification

The applicant described the poison material as providing the criticality control, which consists of two cladding layers and one core. The core material conductivity is significantly lower than the cladding material conductivity. The cask basket uses a borated-aluminum alloy, aluminum/B4C metal matrix composite or Boral® as the fixed neutron poison material. The staff verified that the applicant included the absorber material chemical composition, physical and mechanical properties, fabrication process, and minimum poison content.

Computation of Percent Credit for Boron-Based Neutron Absorbers

The applicant stated that a credit of 90% is taken for the borated-aluminum alloy or the aluminum/B4C metal matrix composite materials. The applicant stated that a credit of 75% is taken for the presence of neutron poison for Boral plates. The qualified neutron poisons are the systems or processes that meet ASTM standard that specifies how a product will be made and has successfully completed a set of qualification tests for durability and homogeneity. All three neutron poisons have been previously qualified as neutron poisons for storage casks. Also, in the case of burnable poison rod assemblies (BPRAs), no credit is taken for absorber in the inserts and is modeled as B4C with 100% boron-11. This conservatively underestimates the amount of boron-10 poison, both displaced in borated moderator and in the BPRA itself and includes carbon moderator. Any other potential absorbers in the fuel material are conservatively omitted from the criticality model.

Qualifying Properties Not Associated with Attenuation

The applicant stated that neutron absorbers for criticality control are described in SAR section 5.1.1, with testing for properties described in the Prairie Island ISFSI SAR. In the Prairie Island ISFSI SER dated August 2010, ML101590797, the staff reviewed the testing of Boral, borated aluminum, and boron carbide/aluminum metal matrix composite plates. The staff noted that all three neutron absorbers had been previously qualified as neutron absorbers for storage casks. The thermal conductivity and specific heat of the neutron absorber plates, and neutron shield resin are the same values that were previously used in the analysis of TN-40, TN-32, and TN-68 casks. The thermal properties of Boral were used to bound the properties of the metal matrix composite. The staff finds that the applicant mechanically qualified the neutron absorber material in the TN-40HT package.

2.2.8 Corrosion Resistance Environment

The applicant stated that the TN-40HT is loaded in the spent fuel pool and stored on the independent spent fuel storage installation pad. It is prepared for transport in a dry environment, so there is no water immersion under normal conditions. During storage, the TN-40HT is in direct exposure to precipitation and is exposed only to the humidity and aerosols in the cooling air that flows on the ISFSI pad. Some exposure to road salt is possible during transfer from the ISFSI or during transport. The loading procedures in SAR section 8.1.2 state that the cask is vacuum dried and backfilled with helium during loading for storage.

The staff reviewed the description of the environments provided by the applicant in SAR section 7.8.1 and the operating procedures in SAR chapter 8. The staff determined that the applicant has accurately described and addressed corrosion from the environments for the TN-40HT transportation package, and therefore the applicant’s description is acceptable.

Carbon and Low Alloy Steels

The applicant stated that all exposed surfaces of carbon and low-alloy steels are coated to prevent corrosion. The applicant described the coatings used to protect the carbon and low alloy steel components in SAR section 7.9. The staff review of the coatings is described in SER section 7.9 below.

Austenitic Stainless Steel

The applicant stated that the internal components of the TN-40HT transportation package are made of aluminum and stainless steel.

The staff has previously reviewed a number of hardware components and materials to ensure that there are no significant chemical, galvanic, or other reactions as a result of exposure of these various contents to the wet loading and the package’s internal environment. These include components encased in stainless steel and aluminum alloys such as neutron-source assemblies, BPRAs, thimble-plug devices, and other types of control elements. The staff found the following components to be acceptable for transportation when the canister is constructed of stainless steel with stainless steel and aluminum basket components:

- neutron-source materials encased in stainless steel or zirconium alloy cladding containing antimony-beryllium, americium-beryllium, plutonium-beryllium, polonium-beryllium, and californium
- control elements encased in zircaloy or stainless-steel cladding containing B_4C, borosilicate glass, silver-indium-cadmium alloy, or thorium oxide

Therefore, the staff determined that the internal components made of aluminum and stainless steel will not produce any significant chemical or galvanic reactions due to the package's environment.

2.2.9 Protective Coatings

The applicant stated that an aluminum or zinc-aluminum thermal spray is used on several parts. Cask cavity surfaces receive a sprayed metallic coating of Zn/Al or Al for corrosion protection.

Carbon steel cask exterior components are painted (epoxy, acrylic urethane, or equivalent enamel coating) for corrosion protection, which may be thermally sprayed prior to painting.

Cask interior is protected from corrosion by thermal spray during loading and by helium backfill during storage and transport.

The applicant stated that the Zn/Al thermal spray has been previously used to coat the interior of TN-32 and TN-40 storage casks. SAR chapter 8 described the visual inspection of the coatings that are included in periodic maintenance, which includes inspection of all accessible surfaces of the cask for evidence of cracks, corrosion, dents, or other degradation. The staff noted that however, that the thermal spray does not appear in the drawings, nor is mentioned in periodic maintenance in SAR chapter 8 or 9. The applicant responded to the RAI and updated the drawings and maintenance programs to reflect the coatings used by the TN-40HT package. The staff verified that the TN-40HT package meets the requirements of 10 CFR 71.35(a) and 10 CFR 71.43(d) by properly identifying the package components coated and by assessing the effects of corrosion, chemical reactions, and radiation effects.

2.2.10 Content Reactions

Flammable and Explosive Reactions

The applicant states that this is not applicable as TN-40HT is backfilled with helium after loading.

In the Prairie Island ISFSI SER dated August 2010, ML101590797, the staff reviewed the Zn/Al thermal spray coating for reactions with the borated pool water and found that it did not present a safety issue. The cask is bolted shut and the interior of the cask is vacuum-dried, which would remove any generated hydrogen prior to backfilling with helium for storage. Analysis also has shown that galvanic action and hydrogen generation are insignificant in the TN-40HT cask.

Content Chemical Reactions, Outgassing, and Corrosion

The applicant states that this is not applicable as TN-40HT is vacuum-dried and backfilled with helium before storage.

The staff reviewed SAR section 7.10 and verified that the applicant has demonstrated that the contents will not lead to potentially flammable or explosive conditions and that the applicant has evaluated the potential formation of and has employed methods to prevent eutectic reactions by vacuum-drying and backfilling the TN-40HT package with helium. Therefore, the staff finds that the TN-40HT package meets the requirements of 10 CFR 71.33(b), 10 CFR 71.35(a), and 10 CFR 71.43(d).

2.2.11 Radiation Effects

The applicant states that the TN-40HT is made up of systems, structures and components that are not comprised of any rubber, elastomer, or polymers materials (other than the polymer used in the neutron shield), and thus there is no concern that any cask components will be degraded by exposure to gamma radiation.

The bounding neutron flux calculated for the basket structural metal plates is below the measurable neutron fluence range, 10^{17} n/square centimeter, for effects on mechanical properties. Therefore, the staff determined that the TN-40HT package will meet the requirements of 10 CFR 71.35(a) and 10 CFR 71.43(d).

2.2.12 Package Contents

TN-40HT contents are discussed in SAR chapter 1, section 1.2.3. Section 1.2.3 states that the fuel shall be unconsolidated and be limited to WE 14 x 14 standard, WE 14 x 14 OFA, Exxon 14 x 14 standard (including high burnup standard), and Exxon 14 x 14 TOPROD. Requirements for the BPRAs and thimble plug assemblies in cooling time, average cumulative exposure, weight, number of assemblies, initial enrichment, average burnup, uranium loading per assembly, no damaged fuel assembly, max heat load per cask and per assembly, and burnup credit (SAR table 8-2) are also given.

The staff determined that the applicant provided a description of the TN-40HT transportation package contents in SAR chapter 1 including the description of TN-40HT in SAR section 1.2.3, the drawings in SAR appendix 1.6.4 and the technical information on TN-40HT and the SNF contents in SAR section 1.6. The staff determined that the applicant provided an acceptable description of the contents of the TN-40HT transportation package. The staff determined that the applicant provided information with respect to the drawings for the TN-40HT contents because the applicant provided information that was in accordance with the guidance in NUREG/CR-6407 and NUREG/CR-5502.

The staff determined that the technical information on the TN-40HT contents provided in SAR section 1.6 was acceptable because the applicant identified the physical characteristics of the TN-40HT, descriptions of the neutron poison materials and the descriptions of the SNF approved for the TN-40HT transportation package. Therefore, the staff determined that the information provided by the applicant was acceptable.

The staff determined that the applicant has identified the contents of the package in accordance with 10 CFR 71.33(d) and as discussed in sections 7.8, 7.10, and 7.11, demonstrated that the package meets the requirements of 10 CFR 71.35(a), and assessed the effects of corrosion, chemical reactions, and radiation effects in accordance with 10 CFR 71.43(d).

2.2.13 Fresh (Unirradiated) Fuel Cladding

Not applicable.

2.2.14 Spent Nuclear Fuel

The applicant stated that the mechanical properties of spent nuclear fuel are documented and analyzed in SAR chapter 2. The spent nuclear fuel is qualified for storage and is not altered during storage so as to prevent transport. The fuel authorized for transport is undamaged fuel and damaged fuel confined within its fuel compartment by damaged fuel end caps.

In the SER for NUHOMS-61BT, ML040640934, dated March 3, 2004, the staff had reviewed and approved damaged fuel as an approved cask content material if the fuel bundle is confined with the addition of end caps. The end caps prevent loose particulates from the damaged fuel migrating into other cells, thus alleviating criticality, thermal, or shielding concerns that could result from such an occurrence. The staff concluded that if the applicant adheres to the guidance in ISG-1, the staff finds the aforementioned approach to classification for damaged fuel acceptable.

The SAR section 2.11.9 contains the structural evaluation of the fuel rod cladding. In SAR section 2.11.9.3.3, the applicant states that NUREG-2224 is utilized for the guidance for fuel cladding and mechanical behavior and hydride orientations in fuel cladding in dry storage systems and transportation packages. The applicant concluded that for side and end drops, the maximum axial stress in the fuel cladding was lower than the yield strength of the cladding and the maximum principal strain was less than the yield strain of the cladding and less than the strain acceptance criterion. The staff reviewed these results and determined that the fuel cladding maintains its structural integrity and since no plastic deformation was observed, there is no effect on criticality analysis.

The applicant stated that in section 4.2.4.2 of NUREG-2224, “Dry Storage and Transportation of High Burnup Spent Nuclear Fuel,” provides guidance that the impact of Scenarios 1(a) and 1(b) of Category 1 on pressure and temperature can be assumed as 3% fuel rod failure for NCT and 100% fuel rod failure for HAC thermal evaluations. For Scenarios 2(a) and 2(b) in Category 2, and Scenario 3 in Category 3, the impact of fuel reconfiguration is not expected to be significant because the fuel rods are assumed to remain intact. Therefore, the applicant considers 3% and 100% fuel rod failures of reconfigured fuels under NCT and HAC, respectively. The staff reviewed and determined this approach is consistent with the guidance in NUREG-2224, and therefore is acceptable.

The applicant stated that per section 4.2.3 of NUREG-2224, supplemental safety analyses are not expected for high burn up fuel (HBU) fuels that have been in storage for less than 20 years. The staff reviewed and determined that this is in accordance with the guidance in NUREG-2224, and therefore is acceptable.

The staff reviewed the mechanical properties of the spent fuel to confirm that the cladding materials were adequate to ensure that the SNF remains in the configuration analyzed in the application over the ranges of conditions associated with the tests in 10 CFR 71.71 and 10 CFR 71.73 and to confirm that the applicant has identified the contents of the package in accordance with 10 CFR 71.33(b). The staff determined after reviewing the results in SAR chapter 2 that the SNF will remain in the configuration analyzed in accordance with the tests in 10 CFR 71.71 and 10 CFR 71.73 and that the contents of the package were appropriately identified in accordance with 10 CFR 71.33(b).

2.2.15 Bolting Material

The staff reviewed SAR section 7.15 which provides the bolting material used. The applicant described the bolt material and its minimum yield strength at room temperature.

The applicant also described the lid and flange materials. The bolts, lid and flange have the same coefficient of thermal expansion.

The applicant described the vent and drain bolt material, and the vent and drain cover materials.

Torque values on drawings in chapter 1 are derived from minimum preload values calculated in chapter 2.

The staff reviewed the material properties tables in chapter 7 for the bolting materials specified in the SAR, describing the strength of the material, young's modulus, and thermal expansion coefficients. The staff confirmed that the applicant identified the materials to be used in bolted connections in accordance with 10 CFR 71.33(a)(5). The staff reviewed SAR section 8.1.3 and verified that visual inspections of the bolts were part of the periodic maintenance, which would allow for identification of damage or degradation and allow for rework or replacement prior to use. Therefore, the staff considers that the applicant has assessed the effects of corrosion, chemical reactions, and radiation effects on the bolting materials, in accordance with 10 CFR 71.43(d).

2.2.16 Seals

Elastomeric Seals

The applicant states that there are no elastomeric seals for TN-40HT in its transport configuration.

Metallic Seals

The applicant states that the metallic seals have a minimum and maximum temperature rating of 40°F (40 degrees Celsius [°C]) and 644°F (340°C), respectively. The maximum metallic seal temperatures under HAC are 422°F (217°C) for intact fuel and 435°F (224°C) for HBU fuel.

The staff verified that the metallic seals reported maximum temperatures do not exceed the stated maximum temperature limit of 644°F as reported by the applicant. The staff further confirmed that the minimum operating temperature limit specified for the seals is 40°F, which is in compliance with 10 CFR 71.71 (c)(2).

3.0 THERMAL EVALUATION

The objective of the review is to verify that the thermal performance of the Model No. TN-40HT package has been adequately evaluated for the tests specified under both 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

The applicant described in section 3.1.1 of the SAR the following thermal design features in order to provide adequate heat removal for the Model No. TN-40HT package by passive mechanisms only:

- The Model No. TN-40HT includes a basket that is a welded assembly of stainless-steel fuel compartment boxes separated by aluminum and poison plates, forming a sandwich panel. The aluminum provides heat conduction paths from the fuel assemblies to the basket peripheral plates.
- The aluminum basket rails are bolted to the inner shell and provide a conduction path from the basket to the inner shell.
- The aluminum boxes, designed to fit tightly against the steel shell surfaces, that contain the neutron shielding material also create a conduction path and improve the heat transfer across the neutron shield.
- The steel-encased wood impact limiters act as a thermal insulator during the HAC thermal event.
- A personnel barrier, which consists of a stainless-steel mesh attached to stainless steel tubing, encloses the package body between the impact limiters and prevents access.

3.1.2 Codes and Standards

The staff verified that established codes and standards for material properties and components, and verification and validation in computation fluid dynamics and heat transfer (e.g., ASME B&PV Code) are referenced by the applicant throughout the SAR, excluding the seals and impact limiter wood material that have properties from the manufacturer's catalog or a U.S. Department of Agriculture, Forest Service handbook.

3.1.3 Content Heat Load Specification

The staff verified the total package and per assembly decay heat are specified in section 3.1.3 of the SAR. Section 3.1.3 of the SAR specifies that the total heat load is limited to 32 kilowatts (kW) from 40 fuel assemblies with a maximum of 0.80 kW per fuel assembly; the staff verified the total decay heat applied to the thermal model was the same as the value in section 3.1.3 of the SAR.

3.1.4 Summary Tables of Temperatures

The applicant provided the Model No. TN-40HT NCT maximum temperatures in table 3-1 of the SAR. The components include the fuel cladding, outer shell, radial neutron shield, inner shell, basket rail, basket fuel compartments, gamma shield shell, impact limiter wood, bottom inner plate, lid, lid seals, as well as the average cavity gas temperature and the accessible surface temperature.

The maximum temperature of the aluminum alloy neutron shield boxes is bounded by the maximum temperature of the radial neutron shield to perform its intended safety function. Similarly, the maximum temperatures of the basket aluminum and poison plates are bounded by the basket fuel compartments to perform their intended safety function. The staff confirmed that all of the components and the package surface in table 3-1 of the SAR remain below the maximum allowable NCT temperature limits provided in table 3-1 of the SAR. The package surface temperature is evaluated in section 3.4 of this SER.

The applicant provided the Model No. TN-40HT HAC maximum temperatures during the fire or during the post-fire in table 3-3 of the SAR. The components include the impact limiter outer surface, outer shell, bottom plate, lid, lid seals, gamma shield shell, basket rail, inner shell, basket fuel compartments, fuel cladding, and the average cavity gas temperature. The maximum temperatures of the basket aluminum and poison plates are bounded by the basket fuel compartments to perform their intended safety function.

The impact limiter wood was not included in table 3-3 of the SAR. This is because the structural integrity of the impact limiter wood material is not relied upon during or after the fire, which the staff finds to be consistent with the order indicated of the tests specified in 10 CFR 71.73; therefore, the staff finds this to be acceptable. The radial neutron shield material was also not included in table 3-3 of the SAR; the staff confirmed this was consistently described in section 5.3.4.1.1 of the SAR for the HAC shielding evaluation. The staff confirmed that all components in table 3-3 of the SAR remain below the maximum allowable HAC temperature limits provided in table 3-3 of the SAR. The staff verified that the lid, and vent and drain port cover metallic seals have a minimum and maximum temperature rating of -40°C (-40°F) and 280°C (536°F), respectively.

3.1.5 Summary Tables of Pressures in the Containment System

The applicant provided the Model No. TN-40HT maximum normal operating pressure (MNOP) for the package cavity in section 3.3.4 of the SAR. The staff confirmed the pressure with assumed 3% rods ruptured in section 3.3.4 of the SAR, 14.7 psig, (29.4 psia) was below the design pressure limit, 100 psig (114.7 psia), in sections 2.11.1.4.6 and 3.2.3 of the SAR.

The applicant provided the Model No. TN-40HT maximum hypothetical fire accident pressures for the package cavity in section 3.4.3.2 of the SAR. The staff confirmed the pressures with assumed 100% rods ruptured in section 3.4.3.2 of the SAR, 76.1 psig (90.8 psia) with BPRAs, and 60.7 psig (75.4 psia) without BPRAs, were below the HAC design pressure, 125 psig (134.7 psia) in sections 2.11.2 and 3.2.3 of the SAR.

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

The applicant provided material thermal properties such as thermal conductivity, density, specific heat, and surface emissivity for the thermally modeled components of the package. The staff reviewed these properties in the SAR and in the thermal models and finds the properties to be acceptable.

3.2.2 Technical Specifications of Components

The applicant described in section 3.2.2 of the SAR that the containment boundary metallic seals have a maximum temperature limit of 280°C (536°F). The applicant also described in section 3.3.2 that the metallic seals and the package components will continue to function at a temperature of -40°C (-40°F). The staff reviewed, and therefore, finds these component specifications to be acceptable.

3.2.3 Thermal Design Limits of Package Materials and Components

The Model No. TN-40HT package materials and components design criteria are summarized in section 3.2.3 of the SAR and these materials are required to be maintained below the maximum pressure and temperature limits. The staff verified that the maximum pressure and temperature limits of package materials and components, as provided by the applicant in section 3.2.3 of the SAR, are used consistently in the SAR.

The staff reviewed and confirmed that the maximum allowable temperatures limits for components critical to the package containment, radiation shielding, and criticality are specified in section 3.2.3 of the SAR. The staff verified that the fuel cladding temperature limits in section 3.2.3 of the SAR are in compliance with the guidance in NUREG-2216.

3.3 General Considerations for Thermal Evaluations

3.3.1 Evaluation by Analyses

The applicant described the one-quarter symmetry three dimensional (3-D) thermal model of the Model No. TN-40HT transportation package that was developed using the ANSYS computer code in section 3.3.1 of the SAR. The applicant described the thermal modeling of the packaging that included the basket and impact limiters, and the associated gaps between components.

The applicant described in section 3.3.1.4 of the SAR, heat is transferred from the outside surface of the package to the environment by natural convection, and thermal radiation. Within the package body component materials, heat is transferred by conduction. Inside the package cavity and between component surfaces such as the fuel assemblies, heat is transfer by conduction and thermal radiation; internal convection is not thermally modeled.

The staff finds the overall analysis approach and assumptions acceptable because the description satisfies NUREG-2216.

3.3.2 Evaluation by Tests

The TN-32B, Docket No. 72-1021, has been validated by TN and DOE/EPRI as described in the references 16 and 19, respectively, that are listed in section 3.5 of the TN-40HT SAR. The applicant showed on page 3-25 of the SAR that the TN-32B and the TN-40HT are similar in: design and key dimensions, the use of heat transfer in the radial direction, and the use of conservative radial gaps in the thermal model that bound the fabrication gaps. The TN-32B has a slightly higher decay heat of 32.934 kW as compared to 32 kW for the TN-40HT.

The TN-40HT thermal evaluation also demonstrates that there is adequate margin for the fuel cladding and the seals. The applicant concluded that the thermal performance of the as fabricated TN-40HT package will be bounded by the thermal evaluation in chapter 3 of the TN-40HT SAR; the staff finds this to be acceptable based on the staff's review of the applicant's evaluation in section 3.3.5 of the SAR.

Therefore, the staff finds it to be acceptable that a thermal acceptance test, as listed in section 9.1.8 of the SAR, is not required for the TN-40HT package, as discussed in this paragraph based on references 16 and 19 and the similarity between the TN-32B and TN-40HT designs.

3.4 Evaluation of Accessible Surface Temperature

The applicant described in section 3.3.3 of the SAR that with the personnel barrier installed, the accessible surfaces are the impact limiters and the personnel barrier surface. In the shade and in still air at 100°F ambient, the applicant calculated the impact limiter temperature and the personnel barrier surface temperature to be 41.1°C (106°F) and 64.5°C (148°F), respectively. Therefore, the package surface temperature is below the 85°C (185°F) limit specified in 10 CFR 71.43(g) and the staff finds the accessible surface temperature limit criterion is satisfied. Staff also confirmed that a temperature survey is performed as part of the chapter 8 operating procedures in section 8.1.3 of the SAR.

3.5 Thermal Evaluation under NCT

As described in section 3.3.1 of this SER, the applicant performed the thermal evaluation using the ANSYS computer code. The applicant described in section 3.3 of the SAR, that the solar heat input is averaged over 24 hours (hrs.) rather than as is specified in 10 CFR 71.71(c)(1) for a 12-hr. period. Due to the large thermal inertia of the TN-40HT package and the margin to the maximum allowable temperature limits, this assumption is justified. This assumption to average the solar heat input over 24 hrs. was also used and approved for the TN-40 transportation package and the TN-40HT storage cask in Docket Nos. 71-9313 and 72-0010, respectively.

The staff finds the approach to perform the NCT evaluation acceptable because the developed thermal model, as described in section 3.3.1 of the SAR and section 3.3.1 of this SER, is adequate to capture the heat transfer characteristics expected for this design.

3.5.1 Heat

Under a 38ºC (100°F) ambient temperature, in still air, with solar heat input as described in section 3.5 of this SER, and maximum decay heat, the applicant predicted the maximum temperatures of the fuel cladding, outer shell, radial neutron shield, inner shell, basket rail, basket fuel compartments, gamma shield shell, impact limiter wood, bottom inner plate, lid, lid seals, as well as the average cavity gas temperature and the accessible surface temperature. These temperatures are listed in table 3-1 of the SAR.

The staff confirmed that these maximum temperatures are below the material temperature limits with sufficient margin and finds the temperatures to be acceptable.

3.5.2 Cold

With no decay heat, no solar heat load, and an ambient temperature of -40°C (-40°F), the entire package uniformly approaches the ambient temperature of -40°C (-40°F). Package components, including the seals, are not adversely affected by exposure to cold temperatures.

The staff finds these arguments acceptable because the materials of construction are designed to operate at this low temperature.

3.5.3 Temperatures

See section 3.1.4 of this SER.

3.5.4 Maximum Normal Operating Pressure

The MNOP is determined by the ideal gas law from the initial helium gas backfill at the maximum average cavity gas temperature and also includes the gases released from 3% of the fuel rods. The MNOP is 14.7 psig, (29.4 psia) for normal conditions. The maximum average cavity gas temperature considers the bounding NCT ambient temperature of 100°F, in still air, insolation, and the maximum decay heat.

The staff determined that the MNOP is below the design pressure of 100 psig (114.7 psia), as reported in sections 2.11.1.4.6 and 3.2.3 of the SAR, and therefore is acceptable.

3.5.5 Thermal Stresses

The applicant calculated the maximum NCT thermal stresses in chapter 2 of the SAR, specifically appendix 2.11.1 of the SAR. The thermal radial and axial expansions are calculated in appendix 2.11.10 of the SAR. The results in appendix 2.11.1 show adequate margin to allowable stresses to exclude safety concern. The results in appendix 2.11.10, as summarized in table 2.11.10-7 of the SAR demonstrate that the hot gaps after thermal expansion for NCT are larger than zero and therefore permit free thermal expansion and are acceptable.

3.6 Thermal Evaluation for Short Term Operations

The chapter 8 loading procedures in section 8.1.2 of the SAR describe loading procedures assuming a TN-40HT is loaded and in use for storage under a 10 CFR Part 72 license; therefore, there is no thermal evaluation for short term operations in the TN-40HT SAR.

3.7 Thermal Evaluation under HAC

The applicant performed the regulatory fire analysis using a 3-D ANSYS model for the 30-minute engulfing fire and post-fire cooldown. The accident scenario considers the structural damage from the 30-ft. drop and puncture tests for the TN-40, which is similar in design and weight (1% difference) to the TN-40HT impact limiters, and therefore can be used for the TN-40HT. The applicant described that the impact limiters remain in place after the 30-ft. drop. The applicant's description of the conservative impact limiter drop damage and possible wood char during the fire is described in section 3.4.2.2 of the SAR.

The staff finds the thermal evaluation under HAC to be acceptable because the description and analysis satisfy NUREG-2216.

3.7.1 Initial Conditions

The initial conditions of the package, prior to the start of the fire accident are 38ºC (100°F) ambient temperature, in still air, that the staff confirmed is prescribed in 10 CFR 71.71(c)(1), with insolation as described in section 3.5 of this SER. The applicant used the NCT temperature distribution for the Model No. TN-40HT as the initial conditions for the fire accident evaluation. The staff finds the initial conditions acceptable because the description results in bounding temperature results and satisfies NUREG-2216.

3.7.2 Fire Test

The applicant performed a transient thermal analysis to evaluate the package under HAC. During the fire, the applicant evaluated the package at 802ºC (1475ºF) for 30 minutes using the maximum total decay heat specified in section 3.3.1.3 of the SAR.

The applicant described that during the fire, the surface emissivity of the package is assumed to be 1.0, the staff confirmed that value bounds the prescribed value in 10 CFR 71.73(c)(4) and the surface absorptivity of the package is assumed to be 0.8, which the staff confirmed is the value that is prescribed in 10 CFR 71.73(c)(4). After the fire, an emissivity of 0.9 and an absorptivity of 1.0 is used for the external surfaces, which the staff finds is bounding. During the post-fire the insolation is as described in section 3.5 of this SER, which is averaged over 24 hrs. rather than 12 hrs.

The applicant described in the response to RAI 3.1 that there are large margins between the maximum temperatures and their allowable ranges as shown in tables 3.3 and 3.6.1-1 of the SAR and the impact of the insolation values averaged over 24 hrs. rather than 12 hrs. during the HAC post-fire analysis on the integrity of the fuel cladding and the containment boundary is expected to be minimal, which the staff finds to be acceptable based on the large thermal inertia of the TN-40HT package and the aforementioned large temperature margins.

The applicant described in section 3.4.2.2 of the SAR the HAC crushed impact limiter damage and char damage to the impact limiter portions of the HAC thermal models; the staff verified that the damage described in the SAR was included in the HAC thermal models.

The applicant also described in appendix 3.6.4 of the SAR a thermal evaluation of damaged aluminum inserts in the event of a HAC drop event, in the subsequent fire accident and post-fire. The aluminum inserts provide a conduction path for heat dissipation from the basket to the cask.

The applicant demonstrated through the results in table 3.6.4-1 that while the post-HAC drop configuration with damaged inserts had an adverse impact on the basket assembly, there was a minor thermal impact on the cask assembly. The applicant then applied those temperature increases to HAC thermal results in table 3-3 for intact fuel and table 3.6.1-1 HBU reconfigured fuel, which staff discusses in section 3.8 of this SER. The applicant demonstrated that the bounding component temperature results shown in table 3-3 with damaged aluminum inserts remain below the allowable limits, which the staff finds to be acceptable.

The analysis simulates the engulfing fire by prescribing a combination of radiation and convection heat transfer on the package surface. The Sandia National Laboratory fire experiment convection heat transfer coefficient is adopted for the calculation from the Sandia Report SAND85 – 0196 TTC – 0659 UC 71, 1987, "Thermal Measurements in a Series of Large Pool Fires," which the staff finds to be consistent with the guidance in NUREG-2216 and is also applicable to this package. During the fire, gaps are removed and replaced with the adjacent material, during the post-fire the gaps are restored; the staff finds this to be conservative as it allows for a greater transfer of heat into the package during the fire, and for less transfer of heat out of the package during the post-fire.

Similarly, during the fire, the thermal conductivity of the wood impact limiters is increased to maximize heat transfer into the package, and then during the post-fire the thermal conductivity of the wood impact limiters is reduced to minimize the heat transfer out of the package. In addition, the wood impact limiters have a specific heat and a density conservatively set to zero, therefore, they do not have thermal mass during the fire analysis.

After the 30-minute fire, the 38ºC (100°F) ambient temperature is restored and the damaged package is allowed to proceed through a post-fire cooldown phase in still air. The applicant used the ending condition of the 30-minute fire analysis as the initial condition for the post-fire cooldown that is of sufficient duration to allow the package and its contents to reach maximum temperatures. The staff finds the fire test conditions to be acceptable because the description satisfies NUREG-2216.

3.7.3 Maximum Temperatures and Pressure

The maximum temperatures calculated by the applicant are listed in table 3-3 of the SAR. The post-fire cooldown accident temperatures in the table reflect the peak temperature of a specified component from the time the fire was extinguished to the time the package reached steady-state conditions. For both normal and accident conditions, the package cavity was assumed to be filled with helium. The staff confirmed that all components in table 3-3 of the SAR remain below the maximum allowable HAC temperature limits provided in table 3-3 of the SAR.

The applicant calculated the maximum package cavity pressure during HAC assuming that 100% of the fuel rods ruptured. The maximum package cavity pressure under HAC is 76.1 psig (90.8 psia), based on the transient average cavity gas temperature of 323ºC (614°F) and with BPRAs. The maximum package cavity pressure is lower than the HAC design pressure, 125 psig (134.7 psia), listed in sections 2.11.2 and 3.2.3 of the SAR and therefore, the staff finds this to be acceptable.

The applicant concluded in section 3.4.4 of the SAR that the TN-40HT maintains the containment during the drop, puncture, and fire test sequence because the maximum seal temperatures and fuel cladding temperatures are below their respective maximum allowable temperature limits. The neutron shield will off-gas; however, there is a pressure relief valve in the outer shell to prevent pressurization. As noted in section 3.1.4 of this SER, the shielding is assumed to be lost during the fire, and accident dose rates are calculated without that shielding in chapter 5 of the SAR.

3.8 Thermal Evaluation of High Burnup Reconfigured Fuel Assemblies

In appendix 3.6.1, the applicant provided a thermal evaluation of the reconfigured high burnup fuel assemblies during NCT and HAC. The staff reviewed the applicant’s thermal model with high burnup reconfigured fuel. The applicant’s evaluation in section 3.6.1.2 concluded that the NCT thermal evaluation in section 3.3 of the SAR is bounding for the high burnup reconfigured fuel for NCT. Based on the staff’s review of the applicant’s evaluation in section 3.6.1.2, the staff finds this conclusion to be acceptable.

The applicant's evaluation in section 3.6.1.3 showed that the lid seals, and vent and drain port cover plate seals temperatures increased during the HAC reconfigured high burnup fuel assemblies' evaluation, while also considering damaged aluminum inserts, the applicant concluded that the HAC component temperatures in table 3.6.1-1 of the SAR remain below the maximum allowable temperature limits.

The maximum temperatures of the basket aluminum and poison plates are bounded by the basket fuel compartments to perform their intended safety function. The maximum HAC pressure in sections 3.1.5 and 3.7.3 of this SER were determined using the maximum average cavity gas temperature from the HAC reconfigured high burnup fuel assemblies' analysis, which is below the HAC design pressure, 125 psig (134.7 psia), listed in sections 2.11.2 and 3.2.3 of the SAR and therefore, the staff finds this to be acceptable.

3.9 Mesh Sensitivity Analysis

The applicant calculated the grid convergence index (GCI) in appendix 3.6.2 and determined that the GCI could be added to all calculated fuel and component temperatures for NCT and HAC and they would still remain within the maximum allowable temperature limits.

3.10 Evaluation Findings

The staff reviewed the package description and evaluation and concludes that they satisfy the thermal requirements of 10 CFR Part 71.

The staff has reviewed the material properties and component specifications used in the thermal evaluation and concludes that they are described in sufficient detail to permit an independent review of the package thermal design.

The staff reviewed the accessible surface temperatures of the package, as it will be prepared for shipment, and concludes that they satisfy 10 CFR 71.43(g) for packages transported by exclusive-use vehicle.

The staff has reviewed the package design, construction, and preparations for shipment and concludes 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.

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

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

4.0 CONTAINMENT EVALUATION

The objective of this review is to verify that the TN-40HT package design satisfies the containment requirements of 10 CFR Part 71 under NCT and HAC.

4.1 Description of Containment System

Containment Boundary

The containment boundary components of the TN-40HT package consist of the welded inner shell and bottom inner closure, steel lid outer plate, welded shell flange, vent / drain port covers, and any metallic seals (inner seal of tandem seal) and bolts associated with the lid outer plate and vent / drain port covers. The components of the containment system are well defined in chapter 1 (specifically section 1.2.1.1) and chapter 4 of the SAR and are displayed in SAR figures 1-1 and 4-1. Detailed dimensions of the containment boundary are provided in the applicant's SAR and specifically in the Drawing Nos. TN-40HT-71-3, through -5 (provided in appendix 1.6.4 of the SAR).

The applicant states that the sealed "containment vessel prevents leakage of material from the cask cavity" as well as "maintains an inert atmosphere (helium) in the cask cavity" to assist heat removal and protect fuel assemblies against fuel cladding degradation, in compliance with 10 CFR 71.43(d).

The containment vessel, specifically the lid, has two penetrations: a drain port and vent port in the lid. Port covers and metallic seals are utilized to seal these penetrations.

The staff confirmed that the TN-40HT 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 of all package materials, and

(2) the package is made of non-reactive materials and is filled with an inert, non-explosive gas mixture of helium, and fission product gases (including 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).

Seals and Welds

As described in section 4.1.1.3 of the SAR, the TN-40HT package employs tandem metallic Helicoflex® (or equivalent) seals on the lid and two port covers. These seals, which feature a helical spring with an inner lining and an outer jacket, are shown with their nominal dimensions and materials of fabrication in SAR figure 4-2, which indicates that the spring is made of a high strength alloy, such as Nimonic 90 (or equivalent), while the lining and the jacket of the seal may be different materials.

The seals are placed in grooves, machined into the package lid and both port covers, the dimensions of which have been optimized based on the cross section of the seals, to provide the ideal compression of the jacket of the seal when the lid or port cover has been fully installed and tightened. The deformable jacket of the seal and the helical spring within the seal allows the seal to conform to any irregularities in the flange surface while allowing for metal-to-metal contact of the lid and shell flange surfaces for normal transport conditions.

As described by the applicant in SAR section 4.1.1.3, the preload applied to the bolts used to secure the lid and port covers exceeds the force required to compress the seals. The applied preload will ensure that any seal decompression that could potentially occur as a result of the HAC structural or thermal tests would not exceed the useful elastic recovery range (nominally 10% of the design compression) for seal compression. This means that there is enough “elasticity” in the seals to prevent the seals from losing contact with the sealing surfaces during the HAC tests.

This is demonstrated by the applicant in appendix 2.11.4 of the SAR, where the “separation of the lid from the shell flange, calculated for HAC,” will not exceed the useful elastic recovery of the seal as it is installed. In addition, the outer jacket of the seal provides a “connection” between the inner and outer seals, and the “tandem” seal is secured via screws through the seal jacket.

The applicant has stated, in section 4.1.1.3 of the SAR, that leak testing of the containment boundary seals of the TN-40HT is conducted during cask/package fabrication, after any package maintenance, periodically within 12 months prior to shipment, and prior to shipping. The applicant indicated that any leak testing conducted is done in accordance with the ANSI N14.5-1997, “American National Standard for Radioactive Material – Leakage Tests on Packages for Shipment.”

The TN-40HT lid and vent/drain port design features an interspace between the tandem metal seals. In its storage configuration, the TN-40HT features an overpressure (OP) system port in the lid to monitor seal leakage and provides a connection to the tandem lid metallic seal interspace for leakage testing of the lid seals. This OP system port is displayed on Drawing No. TN40HT-71-4 (Items 32 - 34), which is not part of the containment boundary.

The applicant evaluated the temperatures of the seals at the location of the closure lid seal grooves for both the NCT and HAC analyses as presented in chapter 3 of the SAR. The applicant’s calculation of the maximum seal temperatures is discussed in SAR sections 3.3.5 (for NCT) and 3.4.4 (for HAC). The applicant did not explicitly model the seals in their thermal evaluations; therefore, the temperatures reported for the seals are from the seal groove location(s) of the model, the location(s) where the seals would reside if the package was loaded and prepared for transportation. This is generally considered a conservative approach for reporting seal temperatures.

The temperatures calculated are reported by the applicant in tables 3-1 and 3-3 of the SAR, for NCT and HAC respectively, as 224°F (107°C) and 344°F (173°C) for the lid seal, which are both below the allowable limit of 536°F (280°C) for Helicoflex® metallic seals under NCT and HAC and also bounds the temperatures calculated by the applicant for the vent and drain port seals.

The staff confirmed that the temperature of the 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, in SAR section 4.1.1.1, states that the welding of the containment boundary includes: (i) circumferential welds that attach the shell flange and the bottom inner plate to the inner shell, and (ii) both longitudinal and circumferential seam welds on the inner shell.

The applicant, in SAR section 4.1.1.1, describes that the containment boundary components including the inner shell, bottom inner plate, shell flange, and lid outer plate base metal and all circumferential and longitudinal seam welds are leak tested to leak tight criteria found in ANSI N14.5-1997, during fabrication.

Closure

The applicant describes the TN-40HT containment vessel closure in SAR section 4.1.1.4 and a drawing of the closure (Lid Assembly) is provided in Drawing No. TN40HT-71-4. The containment vessel includes “an integrally welded bottom closure and a bolted, flanged lid” assembly, which includes the lid outer plate and the attached shield plate.

The outer lid plate is secured to the shell flange with forty-eight (48) lid bolts using the bolt torque specified in Drawing No. TN40HT-71-1 and utilizing the torquing pattern in figure 8-1 of the SAR, to establish the appropriate sealing surface pressures so that the metallic seals placed in the lid will maintain containment under normal and accident conditions.

A closure bolt analysis is provided by the applicant in appendix 2.11.4 of their SAR. This analysis was reviewed and found acceptable as part of the NRC staff’s structural review, as discussed in sections 2.1.1 and 2.6.4 of this SER.

Containment Penetrations

The applicant, in SAR section 4.1.1.2, describes two lid penetrations (mentioned above) for access to the containment cavity vent and drain ports. Each penetration has a flanged cover plate secured to the lid using eight bolts and employs dual metallic seals to provide a “leak tight” seal.

Similar to the lid outer plate discussed above, the required bolt torque to properly seal the metallic seals of the vent and drain port cover plates is specified in Drawing No. TN40HT-71-1. Finally, the impact limiters installed prior to transportation provide protection to the penetrations from the structural or thermal tests under 10 CFR 71.71 and 71.73 during transportation.

The applicant further states that continuous venting of the TN-40 HT’s containment vessel, which is prohibited under 10 CFR 71.43(h), is not possible, as no devices employed in the design of the TN-40HT would allow it.

Summary of Containment System Review

The staff reviewed the containment system design features of the TN-40HT, as presented by the applicant, in SAR chapters 1, 4, 8, and 9, and verified that the application defines the containment boundary of the package, which includes the containment inner shell, lid outer plate, closure bolts, lid outer plate metallic seals (inner seal of tandem seal) , shell flange, vent / drain port cover plates, bolts at vent / drain ports, metallic seals (inner seal of tandem seal) at vent / drain ports (refer to SAR figure 4-1 for a visual depiction of the containment boundary).

The applicant also adequately described the penetrations for the vent and drain ports and their method(s) of closure. The description of the containment system provided by the applicant complies with 10 CFR 71.33. The staff ensured that components and associated welds of the TN-40HT containment system are adequately described and/or depicted in SAR Drawing Nos. TN40HT-71-3 through -5.

4.2 Codes and Standards

The applicant, in section 4.1.2 of the SAR, indicates that the containment boundary of the TN-40HT is designed, to the maximum extent practicable, as an ASME Class I component in accordance with the rules of section III of the B&PV Code.

ANSI N14.5 specifies the allowable leakage rates and leakage test methods for the containment boundary of the TN-40HT.

4.3 Special Requirements for Damaged Spent Nuclear Fuel

The applicant has not requested damaged fuel as a content for this package.

4.4 General Considerations for Type B Packages

As described in section 4.2.1 of the applicant's SAR, the TN-40HT transport package is designed, constructed and prepared as a Type B package for shipment to assure no loss or dispersal of radioactive contents, as demonstrated by an allowable activity release rate of 10^{-6} A_2 per hour under the tests specified in 10 CFR 71.71 for NCT, and less than an A_2 per week (10 A_2 per week for Kr-85) under the tests specified in 10 CFR 71.73, for HAC, in accordance with 10 CFR 71.51(a)(1) and (2), respectively.

4.4.1 Gas Generation

The applicant specifies, in section 4.2.2 of the SAR, that the TN-40HT cask cavity is vacuum-dried and backfilled with helium at the end of fuel loading operations, providing a helium atmosphere within the cask for NCT. For HAC, fuel rod failure is assumed by the applicant and, therefore, the cavity gas mixture includes fission product gases (mostly xenon, krypton, and iodine); however, neither of the gas mixtures present during NCT or HAC are explosive, and the applicant indicates that: "there is no mechanism for combustible gases to be generated in the TN-40HT." The staff agrees with this assessment by the applicant.

4.4.2 Pressurization of Containment Vessel and Maximum Normal Operating Pressure (MNOP)

In section 4.2.3 of the SAR, the applicant discusses pressurization of the containment vessel, stating that the TN-40HT is backfilled with helium gas (after vacuum drying) and, in section 3.3.4 of the SAR, calculating MNOP as 14.7 psig (29.4 psia).

The applicant reported that, for accident conditions, the pressure inside the cavity increases to 76.1 psig (90.8 psia) due to increased temperatures and the release of fission gases from the fuel rods. The accident pressure is below the maximum design cavity pressure of 125 psig (134.7 psia), therefore containment integrity of the TN-40HT is maintained for accident pressures.

4.4.3 Source Terms for Spent Nuclear Fuel Assemblies

The applicant, in determining the total activity available for release, considered three source activities that would be available for release, as described in section 4.3 of the SAR. The sources considered included: crud (spalled from the outer surface of fuel cladding), radioactive fuel fines, and fission product gases or volatiles (both potentially released through cladding breaches).

The applicant calculated the concentration of activity available for release, using the approach described in NUREG/CR-6487, "Containment Analysis for Type B Packages used to Transport Various Contents," as described in SAR section 4.3, by assuming uniform distribution of the activity available for release and then dividing that value by the free volume of the TN-40HT fuel compartment. The applicant stated that this approach neglects any residual contamination (such as what may be found on the interior surfaces of the cask) as such contamination would be negligible when compared with crud that might be found on fuel cladding surfaces. The staff agrees with the assessment provided by the applicant.

The applicant further summarizes how the ORIGEN-ARP code was used to develop a preliminary Fuel Qualification Table (FQT), presented in table 5-24 of the SAR to determine the design basis fuel for the shielding evaluation. The calculation considers the proposed contents for the TN-40HT: spent 14 x 14 WE standard fuel assemblies (WE14 FA) with up to 5.00 weight (wt).% U-235 maximum assembly average initial enrichment, an average burnup of 60 MWD/MTU, with at least 12 years of cooling time (which meets the specific fuel acceptance criteria provided by the applicant in SAR section 1.2.3).

The applicant then uses the ORIGEN-ARP cases in calculating bounding source activities for the containment evaluation. The input data, provided in table 4-2 of the SAR, is used along with the rod breach and release fractions, shown in table 4-4 of the SAR, to, first, calculate the source activities for the WE14 FA contents and then, the source term activity available for release from containment.

The applicant states that table 5-24 of chapter 5 of the SAR was then adjusted to include the minimum required cooling time before a FA may be loaded into the TN-40HT of 12.0 years, and those results were reported in a "final version" of the FQT, provided in table 8-1 of the SAR.

In table 4-3 of the SAR, the applicant records the burnup, initial average enrichment, and minimum cooling time ranges used in their calculations. Further, the maximum activities (A_2) for fines, gases, and volatiles for a single WE14 FA are also shown in SAR table 4-3. The applicant goes on to state that, in order to simplify the calculations on source term and utilize a conservative approach, they assume that: "crud spallation and cladding breaches occur instantaneously after fuel loading and container closure operations. Therefore, the source term is time-independent, and all the radioactivity in the cavity gas that is available for release from the containment vessel is assumed to be present at the beginning of transportation".

4.4.4 Determination of Allowable Leakage from the TN40HT Package

The applicant, in section 4.4 of the SAR, describes, in detail, the application of the methodologies found in NUREG/CR-6487 and ANSI N14.5, using the appropriate inputs as presented in table 4-1 of the SAR, to determine the permissible volumetric gas release rate from the TN-40HT transportation package, the equivalent helium leakage rate, a hole diameter for the postulated leak, and then the equivalent air leakage rate for both NCT and HAC conditions.

The staff verified the radionuclide inventory and A_2 values for all radionuclides in the TN-40HT package by conducting a confirmatory calculation using Microsoft Excel and following the guidance of ANSI N14.5. The staff also validated the calculations of the effective A_2 value (total activity concentration), the results of which were reported by the applicant in table 4-5 of the SAR, and the calculation of standard reference leakage rate of 2.7×10^{-5} ref-cm^3/sec, as reported in section 4.5 of the SAR.

As part of the review of the overall containment calculations, the staff also confirmed that the data in SAR table 4-1, “TN-40HT Cavity Gas Temperatures, Pressures and Properties,” table 4-2, “WE14 FA Input Data,” table 4-3, “Bounding Activity Values for Fission Gases, Volatiles, and Fines for a Single FA (A_2),” and table 4-4, “Source Term Release Fractions,” were consistent and appropriately applied in the containment calculations that were completed in accordance with ANSI N14.5 for fabrication, maintenance, periodic, and pre-shipment verification to determine the NCT and HAC leakage rates of the TN-40HT package, as described below.

4.4.5 Allowable Leakage for Normal Conditions of Transport

In section 4.4.1 of the SAR, the applicant details the determination of the maximum allowable release rate, as well as the equivalent air leakage rate for NCT. Using the allowable release rate for NCT of 10^{-6} A_2 per hr., the applicant calculated the maximum allowable leakage rate using the average activity concentration for the helium inside the cask (as described in section 4.3.4 of the SAR). Calculating the hole diameter (equivalent hole size) of the postulated leak, the applicant then determines the air leakage rate at standard conditions. The results of the applicant’s calculations are summarized in table 4-1 of this SER, below.

4.4.6 Allowable Leakage for Hypothetical Accident Conditions

Similar to the calculations presented for NCT, in section 4.4.2 of the SAR, the applicant details the determination of the maximum allowable release rate, as well as the equivalent air leakage rate for HAC. Using the allowable release rate for HAC of an A_2 per week, the applicant calculated the maximum allowable leakage rate using the average activity concentration for the helium inside the cask (as described in section 4.3.4 of the SAR). Calculating the hole diameter (equivalent hole size) of the postulated leak, the applicant then determines the air leakage rate at standard conditions. The results of the applicant’s calculations are summarized in table 4-1 of this SER, below.

4.4.7 Compliance with Containment Criteria

As noted in section 4.5 of the SAR, the applicant applies the NCT reference air leakage rate as the most restrictive of the values calculated for both NCT and HAC. The staff confirmed that both HAC with Kr-85 and without Kr-85 are limited to the permissible standard leakage rate of 2.7×10^{-5} ref-cm^3/s (air) to meet the allowable release rates of 10 A_2 per week for

HAC with Kr-85 and an A_2 per week for HAC without Kr-85, in compliance with 10 CFR 71.51(a)(2).

The staff finds that the containment criteria and its allowable leakage rate for NCT, as described by the applicant, satisfies the containment requirements for both normal and accident conditions and demonstrates that the containment boundary performs acceptably under NCT and HAC.

Table 4-1 Permissible Leakage Rates of TN-40HT under NCT and HAC

Case	*Allowable Release Rate*	*Effective A_2 (Total activity concentration) (A_2/cm^3)*	*Allowable Release Rate (A_2/sec)*	*Permissible Leakage Rate (cm^3/sec)*	*Permissible Standard Air Leakage Rate (ref-cm^3/sec)*
NCT	*10^{-6} A_2/hr*	*$6.11x10^{-6}$*	*$2.77x10^{-10}$*	*	*2.726×10^{-5}*
HAC	*A_2/week*	*$6.45x10^{-3}$*	*$1.65x10^{-6}$*	*$2.56x10^{-4}$*	*7.342×10^{-5}*
Notes	**Value is Proprietary*				

4.5 Leakage Rate Tests

As described by the applicant in section 4.5 of the SAR, the applicant determined that NCT provided the more restrictive reference air leakage rate. The applicant will use helium for fabrication, maintenance, periodic, and pre-shipment leak rate tests (if required). As noted by the applicant in SAR section 8.1.3, “Preparation for Transport”:

A periodic leakage rate test shall be performed within 12 months prior to shipment, and a pre-shipment leak test is not required if there has been no assembly of the lid, vent port cover, or drain port cover, or loosening of any of the closure bolts since the periodic leakage rate test was last performed.

The following is a summary of the results of allowable leakage rates calculated by the applicant and presented in section 4.5 of the SAR:

- Leakage rate testing with a resulting acceptance criterion of 2.7×10^{-5} ref-cm^3/sec and a sensitivity of 1.3×10^{-5} ref-cm^3/sec or less to meet the intent of the fabrication, maintenance, and periodic leakage tests.

- Leakage rate testing with a resulting acceptance criterion of 2.7×10^{-5} ref-cm^3/sec and a sensitivity of 1.3×10^{-5} ref-cm^3/sec or less or no detected leakage when tested to a sensitivity of 1×10^{-3} ref-cm^3/sec to meet the intent of pre-shipment leak rate tests.

The staff confirmed that the allowable leak rate of 2.7×10^{-5} ref-cm^3/sec for fabrication, maintenance, periodic, and pre-shipment leak tests (if required) of TN-40HT meets the requirements of 10 CFR 71.51(a)(1) and 71.51(a)(2).

4.6 Evaluation Findings

The staff has reviewed the applicant's description and evaluation of the containment system and concludes that: the application identifies established codes and standards for the containment system, the package includes a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package.

The staff has reviewed the applicant's evaluation of the containment system under NCT and concludes that the package is designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71, "Normal Conditions of Transport," the package satisfies the containment requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for NCT with no dependence on filters or a mechanical cooling system.

The staff has reviewed the applicant's evaluation of the containment system under HAC and concludes that the package satisfies the containment requirements of 10 CFR 71.51(a)(2) for HAC, with no dependence on filters or a mechanical cooling system.

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

5.0 SHIELDING EVALUATION

The objective of the shielding review is to ensure that there is adequate protection to the public and occupational workers against direct radiation from the contents of the TN-40HT transportation package, and to verify that the package design meets the external radiation requirements of 10 CFR 71.47 and 10 CFR 71.51 under NCT and HAC, respectively.

The analysis performed by the applicant uses a bounding source high burn-up WE 14x14 standard (WE14 STD) fuel assemblies (FA) containing a hybrid design basis source fuel insert. The shielding analysis maximum total dose rate results are compared to the NCT and HAC dose rate regulatory criteria of 10 CFR 71.47 and 10 CFR 71.51 to demonstrate limit compliance.

5.1 DESCRIPTION OF SHIELDING DESIGN

5.1.1 Shielding Design Features

The principal design features of the TN-40HT packaging with respect to radiation shielding consists of a stainless steel and aluminum basket structure containing borated aluminum poison plates that are gamma and neutron shielding materials. A steel gamma shield shell, a steel containment vessel inner shell and bottom inner plate, and a steel lid are high Z shielding material appropriate for attenuating high energy gamma radiation. A radial neutron shield assembly is shielding material appropriate for attenuating high energy neutron radiation. The impact limiters placed on each end of the cask are constructed of wood and steel that provide some shielding material and protect the shielding design features. The staff reviewed the dimensions, tolerances, configurations, and densities of materials for neutron and gamma shielding components that can affect package shielding performance.

The TN-40HT general dimensions, materials of construction, and arrangement are described in chapter 1, section 1.2.1 of the application. The TN-40HT materials of construction are summarized in table 5-1 of the application.

Detailed drawings for the TN-40HT are provided in appendix 1.6.4 of the application. The materials used to fabricate the cask are shown in the Parts List on Drawing TN40HT-71-1 and the materials used to fabricate the impact limiters are shown in the Parts List on Drawing 10421-71-41. The staff verified the engineering drawings, which provide the necessary details for the features of the package and configurations of components that are important in assessing the shielding performance of the package and demonstrating compliance with 10 CFR Part 71 regulations.

5.1.2 Summary of Maximum Radiation Levels

Summary of TN-40HT Exclusive Use Open Transport NCT and HAC Maximum Total Dose Rates are presented in table 5-2 of the application. The table shows: primary gamma radiation, secondary gamma, or neutron capture gamma (n, γ) radiation, and Neutron (N) radiation. Results reported in table 5-2 of the application are determined from an array of case scenarios, which are described in detail in section 5.3.3.2 of the application for NCT and in section 5.3.4.2 of the application for HAC.

The maximum calculated dose rate on the surface of the package for NCT was on the cask side surface between the impact limiters. The total dose rates resulted to be 155.26 mrem/hr. This dose rate shows the package shielding design meets the regulatory requirements of 10 CFR 71.47(b) and 71.51(a)(1) which is 200 mrem/hr. under NCT.

The maximum calculated dose rate on the vehicle on contact was determined at the vehicle rear for NCT. The total dose rates resulted to be 42.89 mrem/hr. This dose rate also shows the package shielding design meets the regulatory requirements of 10 CFR 71.47(b) and 71.51(a)(1) which is 200 mrem/hr. under NCT.

The applicant's maximum calculated dose rate at 2 meters from the package with the TN40-HT fuel basket is 8.39 mrem/hr. under NCT. This dose rate shows the package shielding design meets the regulatory requirements of 10 CFR 71.47(b)(3) which is 10 mrem/hr. under NCT.

The applicant's maximum calculated dose rate with uncertainty in the vehicle cab for the package with the F-69L fuel basket is 1.92 mrem/hr. under NCT. This dose rate shows the package shielding design meets the regulatory requirements of 10 CFR 71.47(b)(4) which is 2 mrem/hr. under NCT.

For the package under HAC, the applicant's maximum calculated dose rate at 1 meter from the surface side of the TN40-HT fuel basket is 829.47 mrem/hr. This dose rate demonstrated that the package shielding design meets the regulatory requirements of 10 CFR 71.51(a)(2) which is 1000 mrem/hr. under HAC.

The staff reviewed the dose rates presented in these tables and finds that the applicant has correctly identified the location of the maximum dose rates for the package under NCT and HAC pursuant to the regulatory requirements of 10 CFR 71.47 and 71.51.

5.2 Source Specifications

Section 1.2.3 of the application describes the fuel assemblies authorized for transport in the TN-40HT. The WE14 STD is used in the shielding analyses because it contains the maximum amount of heavy metal mass at 410 kgU compared to all the other FAs authorized for transport in the TN-40HT. The staff finds acceptable WE14 STD FAs because these fuel assemblies contain maximum uranium mass resulting in the generation of bounding neutron and gamma source terms. Non-fuel assembly hardware is also authorized for transport in the TN-40HT. The non-hardware is described also in section 1.2.3 of the application. The compositions of FA structural materials, including impurities in the fuel matrix, are from table 5.1 and table 5.4 of "Standard and Extended-burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018. Table 5-3 reports cobalt impurities in stainless steel of 800 part per million (ppm) and in Inconel of 4694 ppm.

The radioactive decay of fission products, actinides, and activated structural materials and impurities are the major contributors for the WE14 STD FA source term. The only isotope that contributes appreciably to the gamma source for the FA hardware in the top nozzle (TN), plenum (PL), and bottom nozzle (BN) regions is Co-60, which arises primarily from cobalt activation of the stainless steel and Inconel structural materials.

The applicant used a flux scaling factor for each region (TN, plenum and BN) to correct for the spatial and spectral changes of the neutron flux outside of the active fuel (AF) region. These flux scaling factors, obtained from "*Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal, Volume 1- Activation Measurements and Comparison with Calculations for Spent Fuel Assembly Hardware,"* PNL-6906, are shown in the last column of table 5-6 of the application.

For both NCT and HAC, only a single uniform heat load zoning configuration (HLZC) is incorporated for the TN-40HT transportation configuration. The TN-40HT shielding analysis uses the WE14 STD FA design basis for the source term analyzes. The design basis fuel assembly is chosen based on the burnup, enrichment, and cooling times (BECT) using a FQT approach which is explained in greater detail in section 5.4.1.1.4 of the application.

For HAC, it is assumed that the cask radial neutron shield and the neutron attenuating front (top) and rear (bottom) impact limiters are lost in the accident, using a BECT which maximizes the neutron component of the dose rate will produce bounding HAC dose rate results.

In terms of creep, the applicant cited NUREG-2223 *"Dry Storage and Transportation of High Burnup Spent Nuclear Fuel."* This NUREG specified that creep is not expected to result in FA cladding gross rupture if cladding temperatures are maintained below 400 °C (752 °F) for dry storage of high burnup SNF for a period of up to 20 years.

The applicant used ORIGEN-ARP module of the SCALE 6.0 modular computer code system to generate the design basis source for the WE14 STD fuel assembly. ORIGEN-ARP is a control module for the ORIGEN-S computer program that can rapidly compute source terms and decay heat with a simplified input file compared to a full two-dimensional SCALE 6.0/TRITON calculation. The applicant used two depletion cycles with shutdown between the two cycles. The depletion cycles were performed with effective full power day and a few days for shutdown. Staff finds this approach acceptable because when using a full day power cycle, with a few days for shutdown between cycles are assumed resulting in the final burnup values being conservative.

The assemblies to be qualified for transportation in the TN40-HT contains WE14 STD FA. The applicant provided a description of the design basis fuel assemblies for the source term calculations in table 5.6 of the application. The steel and Inconel hardware masses for the design basis PWR assembly are listed in table 5-5 of the application.

The staff reviewed the source-term calculations and found acceptable based on the facts that the applicant considered the nuclide importance changes in high-burnup fuels as a function of burnup and cooling time. The staff ensured that the applicant provided appropriate descriptive information, including validation and verification status, and reference documentation. Also, the staff determined that the ORIGEN-ARP code is suitable for determining the source terms and if it has been correctly used.

Staff reviewed and performed confirmatory analysis similar depletion code as the applicant for the source terms and finds them acceptable based on the information provided by the applicant and by staff's confirmatory calculations.

5.2.1 Gamma Source

The applicant modeled the design basis source WE14 STD FA as containing a design basis source thimble plug assembly (TPA) fuel insert in the plenum (PL) and TN axial regions of the FA and as containing a design basis source BPRA fuel insert in the AF region of the FA. The design basis TPA and BPRA fuel insert sources were obtained from the TN-40HT storage safety analysis of "Prairie Island Independent Spent Fuel Storage Installation Safety Analysis Report," Revision 19. For the TPA fuel insert, the depletion calculation was performed with an initial planar average U-235 enrichment of 3.85 wt.% as having undergone 125 GWd/MTU of irradiation and then having a post-irradiation cooling time of 16 years. For the BPRA, the depletion calculation was performed with an initial planar average U-235 enrichment of 3.85 wt.% as having undergone 30 GWd/MTU of irradiation and then having a post-irradiation cooling time of 18 years. The design basis source for these fuel inserts is reported in table 5-8 of the application.

The source terms for the WE14 STD FA design basis source, used for NCT intact (not reconfigured) fuel conditions and all HAC fuel conditions including the table 5-8 bounding fuel insert gamma source, is reported in table 5-9 of the application.

Staff reviewed the gamma source methodology and found it acceptable. The staff verified that the key parameters were described in the application and listed in the input file provided by the applicant. The staff performed confirmatory analysis similar depletion code as the applicant for the gamma source terms and finds them acceptable based on the information provided by the applicant and by staff's confirmatory calculations.

5.2.2 Neutron Source

For the neutron source terms analysis, the applicant generated a raw neutron source with an ORIGEN-ARP depletion simulation, and then adjusting that unmodified total neutron source with a neutron multiplication correction factor (NMCF). The equation for the correlations is presented in section 5.2.4 of the application. In the NMCF correlation, the two terms are BPF and k_{eff}. The BPF is the FA Burnup peaking factor, and the k_{eff} is the dry TN-40HT system effective neutron multiplication factor.

The applicant also performed analysis for neutron source peaking and the burnup peaking factor. The staff recognize from previous applications that FA will exhibit an axial burnup profile in which the fuel is more highly burned near the axial center of the FA and less burned near the ends. Since the gamma source term varies proportionally to the gamma axial burnup profile, while the neutron axial burnup profile varies exponentially with the gamma axial burnup profile by a power of 4.0 to 4.2, the applicant used the gamma axial burnup profile to obtain the neutron axial profile raised to the power of 4.0. The average value of the neutron source distribution is 1.152, as shown in table 5-15 of the application.

The neutron source results for the post-irradiation cooling time HBU WE14 STD FA design basis source, used for NCT intact (not reconfigured) fuel conditions and all HAC fuel conditions is reported in table 5-12 of the application. The FA unmodified total neutron source from the ORIGEN-ARP depletion simulation output for this source is also report in table 5.12 of the application.

Staff reviewed the neutron source methodology and found it acceptable. The staff verified that the key parameters were described in the application and listed in the input file provided by the applicant. The staff performed confirmatory analysis using a similar depletion code as the applicant for the neutron source terms and finds them acceptable based on the information provided by the applicant and by staff's confirmatory calculations.

5.3 Shielding Model and Model Specifications

5.3.1 Shielding Model

The applicant performed its shielding analysis of the TN-40HT modeled explicitly with MCNP 5, version 1.4, computer code. The applicant also used a MCNP auxiliary code called ADVANTG. According to the applicant, the neutron cases were first run using the MCNP auxiliary code ADVANTG, which generates weight window files that reduce the variance in the dose rate tally results. Using the weight window file approach for the neutron runs results in a dose rate result tally convergence of less than 3%.

The TN-40HT transport configuration MCNP model is based on the TN-40HT storage configuration MCNP model. The applicant modified the model from the storage configuration into the transport configuration by first removing some storage features. The storage features are presented in section 5.3 of the application. To complete the transformation from the storage model to a transportation model, the applicant added some transportation features like a dual zone HLZC is modified to be a single, uniform HLZC with an allowable 0.80 kW of decay heat per FA including fuel inserts in each storage compartment, and an aluminum front (top) IL spacer is placed onto the TN-40HT on top of the cask top lid (at the front of the horizontally positioned cask). Some other transportation features are presented in section 5.3 of the application.

5.3.2 Computer Codes

The applicant performed shielding analysis calculations using the Monte Carlo computer code MCNP 5, version 1.4. In addition to using MCNP, the neutron models were first run using the MCNP auxiliary code ADVANTG, which generates weight window files that reduce the variance in the dose rate tally results.

Staff finds this acceptable because Monte Carlo transport code offers a full three-dimensional combinatorial geometry modeling capability including complex surfaces, such as cones and tori. This means that no gross approximations were required to represent the TN-40HT in the shielding analysis. The staff also found acceptable the use of AVANTG computer code since this code can extract problem geometry, material composition, source, and tally information from MCNP 5 models. It also provides a powerful, efficient, and fully automated alternative to traditional methods for generating variance reduction parameters.

5.3.3 Materials

The applicant presented the basic structural materials as used in the TN-40HT transport shielding analysis models in table 5-16 of the application. Some of the materials used by the applicant are 304 stainless steel, carbon steel, and aluminum obtained from the SCALE Standard Composition Library. The composition of the cask radial neutron shield resin is reported in table 5-17 of the application. The composition of the structural materials used in the TN-40HT impact limiters are obtained from the SCALE Standard Composition Library and are summarized in table 5-18 of the application.

The staff found acceptable the description of the materials used in the TN-40HT based on the facts that the applicant used appropriate material properties (e.g., composition, mass densities, and atom densities) in the shielding models for all packaging components, package contents, and the conveyance. Also, the applicant calculated correct atom densities and correctly input these densities into the analysis models.

5.3.4 Model Specification

NCT Model

The shielding model for the NCT configuration shielding model was complete in three-dimensional simulation of the TN-40HT transportation package in the exclusive use open transport configuration with all systems, structures, and components intact and fully functional and the FA material and geometry are unmodified. The TN-40HT basket structure, cask body, containment shell, lid assembly, radial neutron shield assembly, impact limiters, and fuel assembly are presented in sections 5.3.3.1.1, 5.3.3.1.2, 5.3.3.1.3, 5.3.3.1.4, and 5.3.3.1.5, respectively.

The applicant used the drawings presented in appendix 1.6.4 of the application to create the MCNP models used in the radiation transport calculations. For the basket structure, the TN-40HT NCT shielding model radial basket structure cross-section is shown in figure 5-2 of the application. For the shield shell, containment shell, and lid assembly structure, the applicant presents the NCT and HAC intact fuel shielding model axial cross-sections in figure 5-4 of the application.

The staff verified that the dimensions and material properties of the packaging components used in the models maximize the package radiation levels. The radial neutron shield assembly dimensions as used in the shielding models were summarized in figure 5-5 of the application, which is an excerpt taken from the appendix 1.6.4 Drawing TN40HT-71-3. The IL dimensions were summarized in figure 5-7 of the application, which is an excerpt taken from the appendix 1.6.4 Drawing 10421-71-42. For the fuel assembly, the applicant modeled as four homogeneous rectangular regions such as lower end fitting in the bottom region, the AF, the plenum and upper end fitting at the top region.

Configuration of Source and Shielding

The applicant analyzed both intact and reconfigured fuel scenarios for NCT. Based on NUREG-7203, dose rates from the package can vary with respect to a variety of source relocation scenarios. The applicant analyzed the NCT reconfigured fuel analysis relocating the active fuel region source into the BN, plenum, and TN axial regions and analyzed increasing the AF region peak node source. Material is not relocated in these investigations. The staff finds this approach to be conservative because placing additional mass into these regions could enhance self-shielding. The NCT analyses calculate maximum total dose rates in five case configurations by modifying the FA source strength and distribution physics parameters. Description of the modification of the FA source strength are documented in section 5.3.3.2 of the application.

Fuel Assembly Axial Distribution

Table 5-9 of the application shows the four axial zone gamma source energy spectrums. Table 5-6 of the application shows the intact FA distributed into the FA axial zones of the shielding model FA using axial source distribution cards defined using the BN, AF, PL, and TN axial zone region heights and spans (width and depth) reported in table 5-6.

HAC Model

The applicant performed HAC configuration using a complete 3-D simulation of the TN-40HT transportation package in the exclusive use open transport configuration. Fuel reconfiguration was also examined under HAC configuration. The applicant replaces the neutron shield with a void and removes the impact limiters from its MCNP model under HAC.

Staff finds this acceptable because replacing the neutron shield with a void and removing the impact limiters from its MCNP model under HAC makes the model conservative since it is expected that those parts remain in place under HAC. For HAC reconfiguration analysis, the applicant used NUREG-7203 *"A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages,"* to discuss scenarios where the FA is compressed, and gross cladding failure occurs causing the uranium oxide pellets to form into rubble. Axial relocation of this rubble in the package storage compartment can result in exceedingly high localized dose rates at the package ends depending on the shielding characteristics of the package being examined.

The staff found the reconfiguration analysis acceptable since it is conservative to consider the modification of the source strength and distribution physics parameters to reflect an AF region compressed to 50% void fraction with an unmodified axial burnup profile. The compressed AF region is shifted towards the bottom of the fuel storage compartment, which increases dose rates at the rear (bottom) of the cask. The applicant also modifies the source strength and distribution physics parameters to reflect an AF region compressed to 50% void fraction with a flat axial burnup profile.

Staff finds this HAC analysis acceptable because the calculated dose rates on the surface of the cask are below 1000 mrem/hr. The TN40-HT therefore complies with 10 CFR 71.47(b)(2). The applicant included tolerances for selected dimensions in the drawings in the Appendix 1.6.4 of the application. The applicant reduced dimensions where the effect of the tolerances would have a significant effect of dose rates to a minimum value, with a special focus on those dose rates with smaller margins to the regulatory limits. The staff found this approach acceptable since minimum dimensions will result in conservative calculated dose rates.

5.4 Shielding Analysis

5.4.1 Methods

The applicant used three sequential steps to establish that the TN-40HT in the transport configuration which meets the dose rate regulatory requirements of 10 CFR 71.47 and 10 CFR 71.51 for both NCT and HAC. First, a transport FQT is created. The shielding analysis bounding source is then selected from the FQT. The bounding source and the shielding model are then used to perform the NCT and HAC maximum total dose rate analyses. The applicant determined the locations around the package which produce the highest total dose rate values for a variety of BECT source combinations.

Once the locations were selected, the applicant modified the shielding model into what is referred to as a response function (RF) model. The resulting detector dose rate values are referred to as RFs and are used as proportionality constants to scale the source strength of a given nuclear particle at the energy of interest into a maximum dose rate at that cask location. The RF values are then used via a processing script to evaluate a large array of single axial zone FA BECT source combinations generated by ORIGEN-ARP to account for all plausible FA BECT combinations that are likely to be considered for transport in the TN-40HT.

With the bounding source BECT established and the TN-40HT transport shielding model developed, the shielding analysis work to demonstrate 10 CFR 71.47 and 10 CFR 71.51 dose rate limit compliance is performed. The applicant presents the maximum total dose rates with relative uncertainties that are reported for all 10 CFR 71.47 points of interest in table 5-28 through table 5-32 of the application for the TN-40HT NCT and HAC transport configuration cases. The Monte Carlo uncertainty for all tallies is very small, less than 5%.

The staff noted that the applicant used an explicit approach in their analyses representing the axial burnup profiles, where source terms were determined individually for each axial section, based on the local burnup calculated from the assembly average burnup and the axial profile. The staff found this approach acceptable since it is more accurate for the TN-40HT. The staff concludes that the profiles used for the design basis calculations are acceptable.

The applicant calculated the NCT vehicle surface maximum total dose rate results and they are summarized in table 5-29 of the application. The same for the NCT 2 meters from vehicle surfaces maximum total dose rate. Dose rates results are summarized in table 5-30 of the application. The NCT minimum required distance for normally occupied spaces results are summarized in table 5-31 of the application.

Since MCNP is a statistical code, the calculated values have an uncertainty associated with them. This uncertainty needs to be considered in the dose rate evaluation. The applicant shows how it considered the uncertainty in section 5.4.1 of the application. The staff reviewed this information and found it acceptable.

5.4.2 Confirmatory Analysis

5.4.2.1 Source term Confirmatory Calculations

The staff used the ORIGEN-ARP code from the SCALE 6.1 code package to verify the spent fuel source term for the loading patterns and fuel qualification strategy. The staff verified the spent fuel gamma and neutron source terms using the data provided for burnup, enrichment, and cooling times from table 5-23 of the application.

The staff was able to confirm independently the gamma and neutron source terms in the tables above are appropriate or conservative. The staff found that although the reactor operating parameters used by the applicant in the depletion calculations are unknown, there is reasonable assurance that the reactor operating parameters were reasonably bounding based on these calculations.

The staff found that the neutron and gamma source terms for the TN-40HT are acceptable.

5.4.2.2 Shielding Confirmatory Calculations

The staff performed confirmatory calculations using the Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) code Version 4.0 available through the Radiation Safety Information Computational Center at Oak Ridge National Laboratory (ORNL). The UNF-ST&ARDS code is a comprehensive integrated data and analysis tool developed for the U.S. Department of Energy Office of Nuclear Energy Spent Fuel and Waste Disposition program with support from the NRC.

This code uses ORIGAMI for source term evaluations and Monaco/MAVRIC for the dose calculations. The staff’s model was built by ORNL using design basis data for the TN-40HT drawings in chapter 1, appendix 1.6.4 of the application. Some of the notable differences between the staff’s calculation method and that of the applicant is that the staff’s model represents the fuel rods/pins explicitly versus using a homogenized fuel mixture. The burnup profile is also represented by depleting each axial zone individually rather than using an adjustment factor and the staff’s model includes the full loading pattern with all fuel assemblies modeled simultaneously using the design basis BECT combinations.

The results from the staff’s model agree with those of the applicant’s providing additional assurance that the TN-40HT package in this application can meet the regulatory dose requirements in 10 CFR 71.47 and 71.51.

For all dose and dose rate calculations, the applicant used a BECT combination that maximized the dose rate for each location. The maximum dose rate reported for the top of a cask configuration may use a different BECT than that reported for the side or bottom of a cask. The staff did not perform calculations with the same BECT as the applicant for each location, however for the BECT combinations the staff evaluated the staff’s model calculated the dose rates for all the reported locations. Even with different BECT combinations when compared to the applicant, the staff’s results reasonable approximate those reported by the applicant which provides the staff confidence that the applicant’s model is adequately for generating realistic dose rates for the TN-40HT package.

5.4.3 Input and Output Data

The principal input data used in the shielding model is the dimensions shown in the drawings in chapter 1 of the application, the fuel specifications, and the material compositions listed in section 5.3 of the application. The applicant provides a sample input file for MCNP in appendix 5.A of the application. Staff reviewed the applicant's input and finds that the data was captured in the input.

5.4.4 Flux-to-Dose-Rate Conversion

The applicant used the ANSI/ANS 6.1.1-1977 flux-to-dose rate conversion factors in all the shielding evaluations. The staff finds this acceptable per section 5.5.4.3 of NUREG-2216. The applicant's analysis showed that radiation steaming through the lead slump circular segment does not result in an increase of the maximum design basis dose rate. The staff examined the lead slump analysis and found it acceptable because assuming void in this affected area is conservative.

5.5 Evaluation Findings

The staff reviewed the package shielding design, calculated dose rates, material specification, and models for dose rate calculations. The staff found the applicant used the dimensions and material compositions consistent with the package drawings and bill of materials. The applicant's dose rate calculations, including source term and shielding model assumptions are conservative. The staff has reviewed the external radiation levels of the package and vehicle as it will be prepared for shipment and concludes that they satisfy 10 CFR 71.47(b) for packages transported by exclusive-use vehicle and 10 CFR 71.51 for a package under HAC respectively.

The staff reviewed the shielding analysis provided by the applicant, which provided the sensitivity study of the source terms, MCNP modeling parameters, and dose rates. The staff found them acceptable since the dose rates were calculated from all basket regions to obtain the fuel loading that result in the highest dose rate at that tally point location. The staff also performed confirmatory analysis for source terms using ORIGEN-ARP from SCALE 6.1 depletion code. The confirmatory analysis showed that there is a close (5%) agreement with the applicant's calculations.

The staff followed the guidance provided in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," March 2020, in its review. Based on its review of the information and representations provided in the application, the staff has reasonable assurance that the proposed package design and contents satisfy the shielding requirements and external dose rate limits in 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

This section presents the findings of the criticality safety review for the TN-40HT transportation package. The application includes a criticality safety analysis that credits the reduced reactivity due to fuel burnup for the 14x14 PWR fuel assembly content. Staff evaluated the package against the requirements of 10 CFR Part 71, including the general requirements for fissile material packages in 10 CFR 71.55 and the standards for arrays of fissile material packages in 10 CFR 71.59. Staff also reviewed the applicant's criticality safety analysis of the TN-40HT package that was presented in the SAR. The staff's review considered the criticality safety requirements of 10 CRF Part 71 as well as the review guidance provided in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material".

6.1 Description of the Criticality Design

6.1.1 Packaging Design Features

The TN-40HT package consists of a cylindrical, steel shell containment vessel, with a bolted closure lid. The basket consists of 40 stainless steel fuel compartments with aluminum and neutron poison plates between each adjacent fuel compartment that contains a minimum amount of B-10 that is credited at 90% of content. Criticality safety is maintained by the fixed geometry of the TN-40HT basket as well as the borated aluminum present in the basket structure.

The applicant requested that the TN-40HT be authorized to transport 14x14 PWR fuel, including WE 14X14 Standard, WE 14X14 OFA, Exxon 14X14 Standard (includes high burnup standard), and Exxon 14X14 TOPROD.

The staff confirmed that the text, tables, figures, and sketches describing the criticality design features are consistent with each other and with the information in the General Evaluation section of the application.

6.1.2 Codes and Standards

The regulations that are applicable in this review of the criticality safety of the TN-40HT transportation package include the fissile material requirements of 10 CFR Part 71, specifically the general requirements for fissile material packages in 10 CFR 71.55, and the standards for arrays of fissile material packages in 10 CFR 71.59. In addition, the staff used the review guidance available in NUREG-2216.

The staff verified that the applicant identified the established codes and standards used in all aspects of the criticality design and evaluation and that the applicant used them appropriately.

6.1.3 Criticality Safety Index

The applicant demonstrated that infinite arrays of TN-40HT packages are subcritical under NCT and HAC. Therefore, in accordance with 10 CFR 71.59(c), the criticality safety index (CSI) is zero. The staff confirmed that the CSI is zero through hand calculations.

6.2 Spent Nuclear Fuel Contents

The TN-40HT package is designed to transport and store a maximum of 40 intact WE and Exxon 14x14 PWR fuel assemblies. Table 1-2 of the application lists the fuel parameters for each fuel type. The criticality analyses are performed using the most reactive fuel assembly, which is the WE 14x14 STD.

The staff confirmed that the application clearly and adequately describes the package contents, providing those specifications that are relevant to the criticality safety of the package. The application should show the entire range of contents specifications, or characteristics, that the applicant considered and should specify the limiting values for the content’s specifications.

6.3 General Considerations for Criticality Evaluations

Based on information presented in section 6.3 of the SAR, the applicant evaluated compliance with the requirements specified in 10 CFR Part 71 by modeling the fuel assemblies in conservative conditions for both HAC and normal conditions of transport (NCT) using various assumptions that are considered proprietary. They include appropriate conditions for both moderation and configurations of the fuel assemblies, burnable poisons, flooding, spacing, fuel density, axial burnup zones, and perturbations of the cask under HAC. These are specified in sections 6.3, 6.5 and 6.6 of the SAR.

The staff used as reference NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," issued April 1997, which provides general guidance for preparing criticality evaluations of transportation packages.

6.3.1 Model Configuration

The applicant evaluated three-dimensional models of a single package and arrays of packages under both NCT and HAC using the SCALE 6.1.3 code package and ENDF/B-VII nuclear data. The SCALE package of codes uses multi-group cross-section data and analysis sequences for Monte Carlo neutron transport, as well as the STARBUCS code that evaluates burnup-credit for criticality safety. The STARBUCS code determines the U-235 enrichment values for various burnup and cooling times and enables the modeling of parameters important for determining the burnup credit and serves as a tool to evaluate various assumptions used as part of the criticality safety analysis.

The applicant explicitly modeled the TN-40HT in the most reactive configuration determined in The Prairie Island ISFSI SAR, Revision 19, and included a full-length model of the basket along with the top lid and bottom and used reflection to model an infinite array of casks using the basket dimension listed in table 6-2 of the application. The fixed poison is modeled assuming a plate thickness of 0.125 in. with the minimum B-10 areal density. Fuel assemblies within the basket are modeled as array of fuel pins and guide/instrument tubes. The spacer grids and sub-components are not modeled. The fuel compartment tubes that surround each fuel assembly are 0.4375 in. thick that have the fixed poison plates affixed to aluminum plates that are 0.3125 in. thick in an egg crate configuration. There is a total of 13 poison plates in the basket located at all the faces where at least five fuel assemblies are lined up, providing all the 30 fuel assemblies in the basket are surrounded by poison plates on all four faces except for the outer 10 assemblies that do not have poison plates on the radially outward face. All the geometry parameters used by the applicant are described in table 6-3 of the application.

The structural evaluation determined that the basket does not experience any permanent significant distortion under HAC, and therefore the applicant used the same basket geometry for both the NCT and HAC analyses. The model conservatively excludes the solid neutron shield. Fuel assemblies are shifted toward the corners that are closest to the center of the cask.

For reconfigured fuel assemblies, the TN-40HT is authorized to transport intact WE and Exxon 14x14 PWR fuel types. Intact fuel may undergo damage under HAC during transportation. The type and extent of fuel rod damage under HAC or the potential high burnup fuel reconfiguration during transport conditions are discussed in section 6.3.4 of the application.

The staff found the model configuration acceptable because the applicant's analysis includes a model for demonstrating compliance with 10 CFR 71.55(b) and that the model is consistent with the as-designed package, including tolerances and materials specifications of package components that maximize reactivity. The reviewer also coordinated with the structural evaluation, thermal evaluation, and materials evaluation reviewers to determine the effects of the NCT and HAC on the packaging and its contents.

6.3.1.1 Calculation Models

The applicant performed an analysis of fresh fuel using the maximum allowable enrichment in addition to performing fuel burnup at 5, 10, 15 and 20 years of cooling time. The maximum allowable fresh fuel enrichment was determined for the WE 14x14 fuel in a reconfigured configuration. The USL was determined in section 6.8.1.5 of the application.

For the burnup credit analysis, the applicant used a methodology to take credit for the depletion of fissile materials in the fuel assemblies in the TN-40HT. Loading curves were developed in terms of initial fuel enrichment as a function of average burnup and cooling times for the reconfigured WE 14x14 fuel assembly types and are based on the most reactive assembly in the most reactive TN-40HT configuration. Fresh water flooding is assumed in the analysis.

A total of 28 isotopes are included in the models of burned fuel assemblies, which includes 12 actinides and 16 fission products, and are specified in table 6-4 of the application. The loading curves demonstrate the acceptable combinations of average burnup and initial fuel enrichment for various cooling periods after fuel assembly discharge. These combinations were developed using the STARBUCS module with ORIGEN-ARP libraries that were generated using TRITON simulations. The ORIGEN-ARP libraries for the WE 14x14 STD fuel assemblies are described in section 6.3.3 of the SAR. Loading curves were made at cooling periods of 5, 10, 15, and 20 years. As specified in table 8-1 in chapter 8 of the SAR, the minimum cooling time for a fuel assembly is 12 years before it is allowed to be loaded into the TN-40HT. The applicant conservatively used a cooling time of 10 years in their analysis. The applicant used the initial maximum fuel enrichment for burnups of 5 through 55 Gwd/MTU. For each fuel assembly averaged burnup value and the initial maximum fuel enrichment value met the KSAFE (k_s) value, all applicable biases and bias uncertainties associated with using burnup credit was determined in section 6.8 of the application.

6.3.2 Material Properties

The applicant used the SCALE 6.1.3 code package which contains a standard material data library for common elements, compounds, and mixtures, which contains all the materials used for the TN-40HT burnup credit criticality analysis. A compilation of the materials used in the analysis are listed in table 6-5 of the application. The B/Al poison plates assume a 90% credit for the 10B in the plates.

The staff verified the materials that are used in the criticality models for the packaging and contents. Also, the staff verified that the applicant provided appropriate mass densities and atom densities for materials used in the models of the packaging and contents. Material properties were at the specifications and tolerances maximize reactivity and are consistent with the condition of the package under the tests of 10 CFR 71.71, “Normal Conditions of Transport,” and 10 CFR 71.73.

6.3.3 Computer Codes

As mentioned previously, the applicant used the SCALE 6.3.1 code package and the ENDF/B-VII nuclear data. The burnup credit analysis was performed using the STARBUCS module and the fresh fuel criticality analysis is performed using the CSAS5 control module of the same package of codes. The applicant's criticality analysis uses the 238-group ENDF/B-VII cross-section library. Details of the fuel and basket component design are listed in table 6-5 of the application.

6.3.3.1 ORIGEN-ARP Cross Section Libraries

The bounding ORIGEN-ARP libraries for the WE 14x14 fuel types were generated using bounding depletion parameters and developed by performing TRITON depletion calculations for a range of enrichment and burnup values. A proprietary discussion of this methodology is described in section 6.3.3.3 of the application and covers the allowable fuel assembly classes and TRITON depletion parameters using the recommendations of NUREG/CR-7108, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses-Isotopic Composition Predictions" and are tabulated in table 6-8 of the application.

The applicant also developed a procedure for generating the ORIGEN-ARP libraries for the WE 14x14 fuel types and looked at a burnup range from 0-60 GWd/MTU. This proprietary methodology is described in section 6.3.3.3 of the application.

6.3.4 Determination of Maximum Reactivity

The applicant used several different models for the criticality safety analyses of the TN-40HT and calculated the USL based on the benchmarked critical experiments with both fresh fuel and burnup credit assumptions. The most reactive cask configuration was determined as well as parameters for the maximum initial enrichment for each fuel type, the nuclides of importance, burnup and enrichment limits, horizontal and axial burnup profiles, and loading curves. The analyses support assembly average burnups of up to 60 GWd/MTU and enrichments of up to 5.0 wt.% U-235. Staff found that this approach for determining maximum reactivity of the TN-40HT was acceptable. The staff also verified that the application evaluates each type of allowable contents and clearly demonstrates that some contents were bounded by the contents for which the applicant performed evaluations.

6.4 Single Package and Arrays Under NCT

The applicant demonstrated that the NCT analysis is bounded by the HAC analysis with a proprietary discussion. Staff finds that this is appropriate for this application. Also, the staff ensured that the criticality evaluation demonstrates that a single package is subcritical in the as-designed condition for compliance with 10 CFR 71.55(b) and under both NCT and HAC for compliance with 10 CFR 71.55(d) and (e), respectively.

6.5 Package Arrays Under Hypothetical Accident Conditions

The applicant demonstrated compliance with the array of damaged package requirements of 10 CFR 71.59(a)(2) and are based on the assumption that the fuel is damaged due to the accident, reconfigured, and water in-leakage into the cask cavity. To support this assumption, the applicant evaluated the damage to the WE 14x14 fuel assemblies with a fuel reconfiguration study. To the HAC analysis, the TN-40HT is loaded with the most reactive WE 14x14 reconfigured fuel, the cask cavity if flooded with full density water, and an infinite array of casks are modeled.

Staff verified that the TN-40HT has a CSI, given in 10 CFR 71.59(b) as CSI = 50/“N” of 0 because “N” is infinite. The number “N” is based on all of the following conditions being satisfied: that five times “N” undamaged packages with nothing between the packages is subcritical; that two times “N” damaged packages, if each package is subjected to the tests specified in 10 CFR Part 71.73 (HAC is subcritical with optimum interspersed hydrogenous moderation; and that the value of “N” cannot be less than 0.5.

6.5.1 Most Reactive Configuration

The most reactive configuration under HAC was determined based on a single cask analysis with vacuum boundary conditions but did vary the external moderator density for an infinite array. The bounding WE 14x14 fuel type and fresh fuel assumption enriched to 4.0 wt.% U-235 are used for this HAC analysis.

Various reconfiguration scenarios were considered with water in-leakage to maximize the reactivity of the system, and include cladding failure and missing rods, fuel cladding gap, clad thinning, uniform and non-uniform pitch expansion, single shear, axial off-set, and both internal and external moderator density.

6.5.2 Infinite Arrays of Damaged Casks Analysis

Based on the most reactive configuration determined in section 6.6.1 of the SAR, the applicant applied reflective boundary condition in the radial direction and periodic boundary conditions in the axial direct to reflect infinite arrays of damaged casks in accordance with 10 CFR 71.59. Burnup credit was used to develop loading curves of the minimum burnup required as a function of initial enrichment and cooling time. The initial enrichment was calculated for the WE 14x14 fuel assemblies for fresh fuel and burnups ranging from 5-60 GWd/MTU. The burnup, initial enrichment, and cooling time (BECT) combinations are generated such that the k_{eff} is below the ks values and are shown in table 6-1 of the application for the WE 14x14 fuel types.

6.5.3 Misload Analysis

The applicant used the criterion in NUREG-2216, section 6.4.7.5, to evaluate the potential single and multiple misload events. The single misload is evaluated by misloading one assembly that is severely underburned and highly reactive. The multiple misload is evaluated by misloading all assemblies with moderately burned fuel assemblies. The applicant used the STARBUCS and CSAS5 modules of the SCALE 6.1.3 code to perform their analyses using the set of recommendations in NUREG-2216 for performing a misload analysis.

The applicant’s analysis uses the ORIGEN-ARP library they created in section 6.3.3 of the application and use bounding depletion parameters and control rods insertion up to the first 15 GWd/MTU of the depletion cycle. The subsequent criticality models in section 6.3.4 of the application are created with the bounding fuel type. The most reactive configuration in HAC with water in-leakage is used for the analysis. Appropriate biases and bias uncertainties are accounted for the isotopes of interest, and a 2% administrative margin is used. The misload analysis was performed by generating an equal reactivity curve followed by quantifying the fuel assemblies from the inventory falling above and below the curve. There are two equal reactivity curves that correspond to the sing and multiple misload analyses for the TN-40HT with the B/Al poison Type I.

For the single misload analysis, the equal reactivity curve is created by finding the maximum allowable enrichment for fresh fuel and determining the minimum burnup corresponding to an enrichment of 5.0 wt.% U-235 for the misloaded fuel assembly. For the multiple misload analysis, the equal reactivity curve is created by determining the burnup points corresponding to the maximum allowable enrichment values. These enrichment points are sourced from the base loading curve generated in section 6.3.4 of the SAR at 5 years cooling time. The curve is generated by connecting three data points that correspond to enrichments of 2.85 wt.%, 3.28 wt.%, and 3.64 wt.% U-235, and were chosen such that all the data points are bounded by the equal reactivity curve.

For the single misload analysis, the base case from section 6.3.4 of the application, which has a burnup of 40 GWd/MTU at 4.24 wt.% U-235 and 5 years cooling time, is modified to replace one fuel assembly in the center of the basket, which represents a severely underburned assembly. For the multiple misload analysis, a similar approach was taken to account for various burnup and enrichment combinations in the TN-40HT.

6.6 Fissile Material Packages for Air Transport

The TN-40HT is not authorized for air transport.

6.7 Critical Benchmark Experiments and Applicable Biases

The applicant provided a comprehensive benchmarking justification in section 6.8.1.1 of the application for fresh fuel and 6.8.1.2 of the application for burnup credit. The details of this benchmarking are considered proprietary, however, the critical experiments with both fresh fuel and burnup credit use neutronics similar to that of the TN-40HT system were selected by the applicant and encompass parameters of enrichment, fuel pitch, EALF, AEG, moderator to fuel ratio, hydrogen to fissile ratio, plutonium content, and fuel rod radii. This benchmarking approach was found to be appropriate by staff because the applicant has benchmarked the computer codes for criticality calculations against appropriate critical experiments. Also, the applicant used the same computer code, hardware, and cross-section library to analyze the benchmark experiments as those used to calculate the multiplication factor for the package evaluations.

The values of the parameters of interest are shown in tables 6-25 and 6-26 of the application for the critical experiments for fresh fuel analysis, and tables 6-27 and 6-28 for the critical experiments for burnup credit analysis. These values were used to plot k_{eff} versus experimental parameters and found by staff to be within the area of applicability. The applicant calculated the upper safety limit (USL) using their results and NUREG/CR-7109, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analysis – Criticality (k_{eff}) Predictions”, and NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology”. Staff finds this approach is acceptable determining the USL.

6.8 Evaluation Findings

Staff reviewed the package criticality safety design, assumptions, modeling, burnup credit evaluation, and material specifications for the criticality safety analyses. Staff found that the applicant used appropriate dimensions and analyses consistent with the package drawings and figures. Staff evaluated the package against the requirements of 10 CFR Part 71, including the general requirements for fissile material packages in 10 CFR 71.55 and the standards for arrays of fissile material packages in 10 CFR 71.59. Staff also followed the guidance provided in NUREG-2216 as part of their review. Staff reviewed the applicant's criticality safety analysis of the TN-40HT package that was presented in the application and found that the applicant adequately evaluated the proposed contents. The staff also performed a confirmatory analysis using the SCALE 6.1.3 code package with STARBUCS and ORIGEN-ARP and found that staff's results agreed closely with those of the applicant calculations. Based on staff review of the TN-40HT package information as provided by the applicant, staff has reasonable assurance that the proposed package design and contents meet the criticality safety requirements of 10 CFR Part 71.

Finally, the staff reviewed the package and concludes that the application adequately describes the package contents, and the package design features that affect nuclear criticality safety in compliance with 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b) and provides an appropriate and bounding evaluation of the package's criticality safety performance in compliance with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a).

7.0 OPERATING PROCEDURES

The applicant summarized the TN-40HT loading and unloading procedures to show a general approach to operational activities. The applicant explained that an operation manual will describe the operational steps in greater detail and will then be used to prepare the site-specific procedures that will address the particular operational considerations related to the TN-40HT. Deviations to the provided procedures are acceptable if justified by the applicable Licensee or Certificate Holder in their quality assurance program to maintain equal or better package effectiveness and continued compliance with the applicable 10 CFR Part 71 requirements.

Since the TN-40HT is in use for storage under a 10 CFR Part 72 license, the section "Preparation for loading" includes only steps to verify that the cask packaging complies with the requirements in chapter 1, appendix 1.6.4, and chapter 9 of the application, including (i) reviewing the fabrication, maintenance, and design change control records for each package to verify that the configuration of the cask as used for storage complies with the conditions for approval for use as a transportation package; (ii) reviewing the fabrication records for the cask to verify that all required acceptance testing and inspections have been performed; and (iii) reviewing the maintenance records to verify that all required periodic inspections and tests have been performed.

In the same fashion, the loading of contents procedure assumes a TN-40HT in use for storage under a 10 CFR Part 72 license and thus includes only steps to verify that the cask contents comply with the requirements in chapter 1 of the application, including the review of the loading records to verify each fuel assembly loaded for storage satisfies the requirements listed in CoC No. 9389, the review of the loading records to verify the cask was vacuum dried and backfilled with helium during loading for storage, and the review of the maintenance records of the cask for situations where air may have leaked into the cask while it was in its storage configuration on the storage pad. If air has leaked into the cask while it was in its storage configuration, an evaluation prior to transportation of the fuel cladding for potential rod splitting due to exposure to an oxidizing atmosphere shall be performed, using the methodology given in NUREG-2215.

The applicant will use helium for fabrication, maintenance, periodic, and pre-shipment leak rate tests (if required). As noted by the applicant in SAR section 8.1.3, "Preparation for Transport", a periodic leakage rate test shall be performed within 12 months prior to shipment, and a pre-shipment leak test is not required if there has been no assembly of the lid, vent port cover, or drain port cover, or loosening of any of the closure bolts since the periodic leakage rate test was last performed. Allowable leakage rates are as follows:

- Leakage rate testing with a resulting acceptance criterion of 2.7×10^{-5} ref-cm^3/sec and a sensitivity of 1.3×10^{-5} ref-cm^3/sec or less to meet the intent of the fabrication, maintenance, and periodic leakage tests.

- Leakage rate testing with a resulting acceptance criterion of 2.7×10^{-5} ref-cm^3/sec and a sensitivity of 1.3×10^{-5} ref-cm^3/sec or less or no detected leakage when tested to a sensitivity of 1×10^{-3} ref-cm^3/sec to meet the intent of pre-shipment leak rate tests.

The NRC staff has reviewed the description of the operating procedures and finds that the package will be prepared, loaded, transported, received, and unloaded in a manner consistent with its design. The NRC staff has reviewed the description of the special instructions to inspect, handle, and to safely open a package and concludes that the procedures for providing the special instructions to the consignee are in accordance with the requirements of 10 CFR 71.89.

8.0 ACCEPTANCE TESTS AND MAINTENANCE

Acceptance tests are mainly performed at the fabricator's facility prior to delivery of the cask for use. These tests and inspections, which are only applicable to cask components identified as quality category A, B, or C on the applicable drawings for package approval, are performed in accordance with written procedures with the results being documented and retained. Visual Inspections are performed on all cask components for any evidence of damage or deformation such as, but not limited to, cracks, pinholes, and uncontrolled voids. Surface finish inspections are performed on all containment boundary sealing surfaces for conformance with the applicable drawings.

Weld examinations should verify that the welds were performed using processes and personnel, both qualified in accordance with the applicable sections of the ASME B&PV Code; dimensional inspections shall be performed of the welds for conformance with the applicable drawings for package approval. An NDE as indicated on the applicable drawings for package approval shall be performed with personnel qualified and certified in accordance with SNT-TC-1A.

The acceptance tests for use of the TN-40HT under 10 CFR Part 71 that were performed during fabrication of TN-40HT under a 10 CFR Part 72 license for storage may be verified from fabrication Quality Assurance records.

Acceptance testing and measurements for the neutron absorber include the measurement of the thermal conductivity, the verification of the areal density (through neutron transmission with a minimum of 37.5 mg ^{10}B/cm^2), the dimensional inspection of the plates, and the verification of the specification for the chemical composition. The material shall be heat treated before dimensional and visual inspections.

Gamma shielding in the cask consists of thick carbon steel components that are fabricated to industry standard specifications as indicated on the applicable drawings for package approval. The acceptance of the gamma shielding is fulfilled by the certification of the steel, visual inspections, and dimensional inspections.

Neutron shielding material shall be certified to verify the following criteria are met: (a) neutron shielding material shall have a density of 0.056 lb/in3 to 0.060 lb/in^3, and (b) shielding material shall have a composition as shown in table 7-14 of the application.

Thermal acceptance testing is not required for the TN-40HT package.

CONDITIONS

In addition to the package description, drawings and contents, the following conditions were included in the CoC:

The TN-40HTpackage shall:

> Be prepared for shipment and operated in accordance with the Operating Procedures in chapter 8.0 of the application, and
>
> Meet the Acceptance Tests and Maintenance Program of chapter 9.0 of the application.

The personnel barrier shall be always installed during shipment of a loaded package.

Transport by air is not authorized.

The CoC's expiration date is December 31, 2028.

CONCLUSION

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

Issued with certificate of compliance No. 9389, Revision No. 0.