SAFETY EVALUATION REPORT
Docket No. 71-9358
Model No. TN-LC Package
Certificate of Compliance No. 9358
Revision No. 0

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SUMMARY

By application dated June 7, 2011, as supplemented August 17, 2011, May 4, August 30, September 28, November 14, November 27 and December 18, 2012, Transnuclear, Inc. (TN) requested approval of the Model No. TN-LC package as a Type B(U)F-96 package. The consolidated application, "TN-LC Transportation Package Safety Analysis Report," Revision No. 6, dated November 2012, as supplemented, supersedes all previous revisions of the application.

The packaging body is a right circular cylinder, approximately 197.5 inches long and 30 inches in diameter, with top and bottom end flanges connected by two cylindrical shells. A 3.5-inch thick cast-in-place lead shielding fills the annulus between the inner and outer shells. Two removable stainless steel lifting trunnions are bolted to attachment blocks, which, in turn, are welded to the outer shell of the packaging to allow attachment of the impact limiters to the package, each with eight 1-inch diameter by 16-inch long attachment bolts. Two pocket trunnions built in the bottom flange are available for rotating the package. The containment boundary for the package consists of the inner shell, the bottom flange, the bottom plug and its inner O-ring, the top flange, the lid, the lid inner O-ring, and the vent and drain port plug bolts and seals.

The package transports, by exclusive use only, irradiated test, research, and commercial reactor fuel in a closed transport vehicle or an ISO container. Specific basket designs are used for a wide range of fuel configurations, i.e., highly enriched aluminum-uranium plate fuel, highly enriched aluminum-uranium pin fuel, and commercial light water reactor fuel assemblies and pins. The packaging may be loaded either in a spent fuel pool or a hot cell environment. The spent fuel payload is shipped dry in a helium atmosphere.

NRC staff reviewed the application using the guidance in NUREG-1617 "Standard Review Plan for Transportation Packages for Spent Fuel." The analyses performed by the applicant demonstrate that the package provides adequate structural, thermal, containment, and shielding protection under normal and accident conditions. 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.

References

Transnuclear, Inc., "TN-LC Transportation Package Safety Analysis Report," Revision No. 6, November 2012, as supplemented.

1.0 GENERAL INFORMATION

1.1 Packaging

The Model No. TN-LC package design consists of a cylindrical steel shell containment system, surrounding a payload basket cavity, with a shielded closure lid and top and bottom impact limiters. Four different basket structures can be placed in the payload basket cavity to transport a wide range of fuel configurations. These basket structures control spacing between fuel elements, and limit the number of fuel elements that may be included in a shipment. The steel inner shell used for containment and a steel outer shell for structural support connect the top and bottom end flange forgings. A thick lead shielding is located between the two cylindrical steel shells, in the bottom end assembly, and in the lid.

The inner diameter of the package is 18 inches, and the packaging consists of 1-inch steel, 3.5-inch lead, and 1.5-inch steel in the radial direction. The Model No. TN-LC package is limited to a maximum heat load of 3.0 kW.

The packaging may be loaded either in a spent fuel pool or a hot cell environment. The spent fuel payload is shipped dry in a helium atmosphere. Nominal weights and dimensions are as follows:

- Overall length with impact limiters:	230 inches
- Overall length without impact limiters:	197.50 inches
- Cavity length (minimum):	182.50 inches
- Cavity inner diameter:	18 inches
- Lid thickness:	7.50 inches
- Weight of contents:	7,100 lbs
- Weight of lid:	1,000 lbs
- Weight of impact limiters:	3,000 lbs
- Total loaded weight of the package:	51,000 lbs

1.2 Contents

There are different basket designs for multiple fuel types and configurations. Details for each basket type are provided in Appendices 1.4.2 through 1.4.5 of the application. As there are different basket heights (and combinations of stacked baskets), stainless steel or aluminum spacers are provided to limit axial movement of the payload. The TN-LC-MTR fuel basket holds fifty-four intact Material Test Reactor (MTR) fuel elements, each with ten to twenty-three flat or curved plates. The fissile fuel plates are an aluminum-uranium alloy or an aluminum alloy reinforced with uranium oxide or uranium silicide. The fuel plates are clad with aluminum. The initial uranium enrichment is up to 94.0 weight percent U^{235} with a maximum burn up of 620,000 GWd/MTU. MTR fuel elements with damaged cladding are authorized provided the total surface area of the damage does not exceed 5% of the total surface area of the damaged element.

The TN-LC-TRIGA basket holds up to 180 Training, Research, and Isotope General Atomic Reactor (TRIGA) Fuel Assemblies/Elements. Two general design types of TRIGA fuel assemblies/elements are included: (i) TRIGA fuel elements consist of a uranium/zirconium hydride matrix clad with aluminum or 304 stainless steel, and (ii) TRIGA fuel follower control rods contain both boron carbide and fissile uranium hydride. The maximum burnup of the TRIGA fuel is 450,000 GWd/MTU. Only intact fuel, with no known or suspected cladding defects greater than hairline cracks or pinhole leaks, is permitted.

The TN-LC-NRUX basket accommodates 26 fuel assemblies of either National Research Universal Reactor (NRU) or National Experimental Reactor (NRX) Mk I intact fuel assemblies. The fissile material is a uranium-aluminum alloy, clad in aluminum. The maximum U^{235} enrichment and depletion is 93% and 80%, respectively. Both intact fuel assemblies (fuel assemblies containing fuel rods, with no known or suspected cladding defects greater than hairline cracks or pinhole leaks) and damaged fuel assemblies are permitted. The extent of the damage is limited such that the total surface area of the damaged cladding does not exceed 5% of the total surface area of each rod.

The TN-LC-1FA basket assembly contains either one intact pressurized water reactor (PWR) or boiling water reactor (BWR) fuel assembly, with a maximum burnup of 62 GWd/MTU. Poison Rod Assemblies (PRAs), a cluster of absorber rods containing B_4C pellets, are inserted into the guide tubes of the PWR fuel assembly for criticality control. The basket can also contain one 25 pin can with up to twenty-five individual intact fuel rods of PWR, BWR, mixed-oxide (MOX) or Evolutionary Pressurized Reactor (EPR) fuel. These rods are limited to a burn up of 90 GWd/MTU. The maximum U^{235} enrichment on all of the fuels is 5 wt.%.

1.3 Materials

The inner (containment) and outer shells of the packaging body are made from American Society of Mechanical Engineers (ASME) SA-240, Type XM19 plate, or any ASME SA-240 austenitic stainless steel having the same or greater strength and ductility. Except for the closure bolts, trunnions and impact limiter attachments, the packaging is primarily constructed of welded austenitic stainless steel. The industrial standards governing these components are ASME, American Society for Testing and Materials (ASTM), depending on the Safety Classifications of the components. The containment vessel is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Boiler and Pressure Vessel (BP&V) Code. Subsection NB is the recommended code of construction for containment in NUREG-1617.

The staff reviewed the properties of the materials listed in the application and found that they matched the properties listed in Section IID of the ASME Code. The staff finds the use of these materials acceptable and in compliance with the guidance in NUREG-1617.

1.4 Drawings

The applicant provided drawings of the package in Section 1.4, “Appendices,” of the application. The staff reviewed these drawings and found that they sufficiently describe the locations, dimensions, and tolerances of the containment system, basket, and neutron absorbing material (when used). Therefore, the staff finds that the applicant meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5).

1.5 Evaluation Findings

A general description of the Model No. TN-LC package is presented in Section 1 of the package application, with special attention to design and operating characteristics and principal safety considerations. Drawings for structures, systems, and components important to safety are included in the application.

The application identifies the TN Quality Assurance Program and the applicable codes and standards for the design, fabrication, assembly, testing, operation, and maintenance of the package.

The staff concludes that the information presented in this section of the application provides an adequate basis for the evaluation of the Model No. TN-LC package against 10 CFR Part 71 requirements for each technical discipline.

2.0 STRUCTURAL REVIEW

The objective of the structural review is to verify that the structural and materials performance of the package is adequately evaluated to meet the requirements of 10 CFR Part 71, including the tests and conditions specified under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

2.1 Structural Design

2.1.1 Description of Structural Design

The Model No. TN-LC package consists of three principal structural components: the packaging body, the basket, and the impact limiters.

The packaging body, a right circular cylinder approximately 197.5 inches long and 30 inches in diameter, is composed of top and bottom end flanges connected by two cylindrical shells. The 3.5-inch thick cast-in-place lead shielding fills the annulus between the inner and outer shells. Together with the bottom end plate and the 7.5-inch thick closure lid, both with encased lead shielding, the packaging body assembly provides a 182.5-inch long by 18-inch diameter payload cavity. Section 1.2.1 of the application notes that the packaging is equipped with two removable stainless steel lifting trunnions bolted to the attachment blocks, which, in turn, are welded to the outer shell of the packaging. Two pocket trunnions built in the bottom flange are available for rotating the package. As described in Section 1.2.1.1.1 of the application, the containment boundary for the package consists of the inner shell, the bottom flange, the bottom plug and its inner O-ring, the top flange, the lid, the lid inner O-ring, and the vent and drain port plug bolts and seals. Drawing 65200-71-01 of the application delineates structural design details for the package body assembly. In addition to the two trunnion blocks, anchor blocks are welded to the outer shell to allow attachment of the impact limiters to the package, each with eight 1-inch diameter by 16-inch long attachment bolts.

Appendices 1.4.2 through 1.4.5 of the application provide descriptions for the four basket designs, TN-LC-NRUX, TN-LC-MTR, TN-LC-TRIGA, and TN-LC-1FA, respectively. The baskets are fabricated primarily with stainless steel tubes, plates, and bars, and configured to allow fuel of various physical attributes to be loaded into and maintained in proper locations in the basket cavity. They are also afforded with assortments of stiffeners, guide rails and plates as well as spacers for transmitting inertia loads laterally to the package inner shell. To accommodate different lengths and combinations of stacked baskets, stainless steel or aluminum spacers are used to limit axial movement of the payload.

Drawing 65200-71-20 depicts design details for the two identical impact limiters measuring 66 inches in diameter and 27.75 inches long. The impact limiters are fabricated with gusset plate partitioned balsa wood and redwood blocks encased in the inner and outer Type 304 stainless steel shells. The stainless steel assembly of gussets and shells locates, supports, confines, and protects the wood energy absorption material. Each impact limiter is attached to the package by eight 1-8 UNC bolts made from SA-540 Grade B23 Class 1 material.

2.1.2 Design Criteria

The applicant demonstrates the structural capabilities of the package by analyses. A 1/3-scale model drop testing of a structurally similar packaging is used, however, to benchmark the impact limiter finite element analysis model, which was subsequently adapted, for determining bounding deceleration g-loads for the package structural evaluation by analysis. Section 2.1.2 of the application summarizes the structural design criteria, including load combinations, for the package. These design criteria are reviewed below:

ASME Code Division 1, Section III, Subsections NB and NG, are used for evaluating the structural performance of the containment boundary and of the fuel baskets, respectively. This is in conformance to the NUREG/CR-3854, “Fabrication Criteria for Shipping Containers,” guidance for Category I shipping containers. With the exception of the neutron shield shell, which is designed, fabricated, and inspected in accordance with the ASME Code, Subsection NF, for construction of nuclear facility support components, the Subsection NB stress criteria are also considered for the non-containment packaging components such as the cask body outer shell. To the maximum practical extent, the Subsection NF criteria are used for design, fabrication, and inspection of the basket hold down ring to facilitate loading of the BWR fuel assemblies. Tables 2-1, 2-2, and 2-3 of the application summarize stress intensity limits for the containment vessel, containment closure lid bolt, and cask basket, respectively. As discussed in Section 2.1.4, code alternatives and corresponding justifications and compensatory measures are presented in Appendix 2.13.13 of the application.

Section 2.1.2.1 of the application notes that, in addition to the ASME Code stress allowable, the acceptability of the containment boundary under the applied loads is also evaluated for preclusion of material fatigue failure and brittle fracture.

For the impact limiter, the application notes that the stainless steel shell casing is designed to support and protect the wood blocks under normal environment conditions, such as moisture, pressure, and temperature. However, the casing shell and the wood blocks are expected to undergo large inelastic deformations in dissipating kinetic energy of the impact limiter upon landing on an essentially unyielding target surface. As noted in Appendix 2.13.12, the impact limiter attachment bolts are designed to remain connected to the package body to prevent separation of the limiters from the package during the 30-ft free drop accident.

The removable trunnions used for on-site package lifting and transfer operations are evaluated to meet the 10 CFR 71.45(a) lifting attachment standard, which requires a minimum safety factor of three against material yielding strength.

Section 2.1.2.1 of the application notes that load cases are applied and combined per the Regulatory Guide (RG) 7.8 guidance. Section 2.13.1.4.1 of Appendix 2.13.1 to the application lists 12 individual load cases from which load combinations are defined for 14 NCT and 13 HAC tests and conditions, which are consistent with RG 7.8 provisions.

2.1.3 Weights and Centers of Gravity

The package has a gross weight of 51,000 lbs. Table 2-10 summarizes information about weights of package components and corresponding centers of gravity for the various payload types with individual baskets. Locations of the package centers of gravity, with individual baskets and contents, are calculated to range from 98.12" to 100.58" measured from the base of the package.

2.1.4 General Considerations for Package Structural Analysis

The applicant performs structural analyses of the package using both hand calculations and finite element computer modeling. The former approach by equations and formulas is used primarily for evaluating packaging components with clearly defined structural behavior and load paths, such as those associated with structural welds, basket bolts, cask closure lid bolts, lifting trunnions, and buckling capabilities of structural components with simple geometry. In the following, the staff reviews general considerations for structural analysis by finite element computer modeling for the package body and basket structural components for a variety of loading conditions, including the NCT and HAC free drops, and load combinations thereof. Section 2.13.1.2.2 of Appendix 2.13.1 to the application notes the use of the general-purpose finite element analysis code, ANSYS, for structural analysis of the package body. The half-symmetry package body analysis model, which is constructed exclusively with the SOLID45 brick elements for different structural components, includes the outer shell, inner shell, top and bottom flanges, and inner/outer shell gamma shielding and shielding caps. For contact interfaces between components, the surface-to-surface contact elements CONT173 and TARGE170 are used. The bolt interactions between structural components in axial and transverse directions are simulated with sets of nonlinear spring elements COMBIN39.

As presented in Sections 2.13.8.5 through 2.13.8.8 of Appendix 2.13.8 to the application, the ANSYS code with additional element types than those used for the package body analysis, as appropriate, is also used for analyzing the TN-LC-1FA, TN-LC-MTR, TN-LC-TRIGA, and TN-LC-NRUX fuel baskets, respectively.

For both the package body and basket components quasi-static analysis, the application of force, displacement, and thermal boundary conditions follows well-established analysis practices, including components thermal gradient consideration and simulation of free-drop inertia effects of the non-structural entities with equivalent pressure loads.

As loading and response conditions dictate, the explicit dynamics of the LS-DYNA finite element analysis code are used for evaluating transient dynamic performance of the package components. This includes modeling of the package free drop tests and the performance of closure lid bolts due to delayed impact in Appendices 2.13.2 and 2.13.7 to the application, respectively.

2.1.5 Conclusion

The staff reviewed the package structural design and concludes that it is adequately described. The design criteria and structural evaluation approaches, including the finite element structural analysis approaches, are in accordance with NUREG-1617 guidelines and are, therefore, acceptable.

2.2 Mechanical Properties of Materials

2.2.1 Shell, Containment Vessel, and Lid Forging

The inner (containment) and outer shells of the packaging body are made from ASME SA-240, Type XM19 plate, or any ASME SA-240 austenitic stainless steel having the same or greater strength and ductility. Except for the closure bolts, trunnions and impact limiter attachments, the packaging is primarily constructed of welded austenitic stainless steel. The industrial standards governing these components are ASME, ASTM, depending on the Safety Classifications of the components. Generic 304 or 300 series stainless steel is used for non-safety related or Safety Class C components. The industrial codes governing their manufacturing and procurement quality (ASME Section III, Subsection NB or NF) are appropriate in accordance with the guidance in NUREG-1617. The staff has reviewed the properties for these materials and found them consistent with literature values. The staff finds the materials used in the construction of the packaging acceptable.

The packaging containment vessel consists of the 1 inch thick inner shell, bottom and top end forgings, a top lid cover assembly forging, a bottom plug, lid bolts, vent and drain port closure bolts and seals, and inner O-ring seals. The inner shell is SA-240 Type XM19 nitrogen-strengthened austenitic stainless steel, or any ASME SA-240 austenitic stainless steel having the same or greater strength and ductility, and the top and bottom flange forging material is SA-182 Grade FXM19. The top lid forging is constructed from SA-182 Grade F304 austenitic stainless steel. Shear bearing blocks made from ASME SA-182-F6NM are attached to the containment vessel located at the mid-length, on the bottom of the cask and are designed to help carry longitudinal loads encountered during transportation. If the shear bearing blocks are constructed from multiple pieces, the fabricator is required to perform a heat-treatment of the entire sub-assembly to ensure that the mechanical properties are restored.

The lid forging is attached to the package body with twenty (20), SA-540 Grade B23, Class 1, 1-inch diameter bolts and stainless steel washers. These bolts will conform to the impact testing requirements of Subsection NB to ensure adequate ductility at low-temperatures. Closure of the bottom plug (with or without gamma shielding) is accomplished by eight (8), SA-540 Grade B23, Class 1, half-inch diameter cap screws and stainless steel washers. Each of the vent and drain ports are closed by a single half-inch brass or ASTM A193, Grade B8 bolt with an elastomer seal under the head of the bolt.

The Model No. TN-LC package containment vessel is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Boiler and Pressure Vessel (BP&V) Code. Subsection NB is the recommended code of construction for containment in NUREG-1617. Exceptions to the ASME code of construction are highlighted in Tables 2.13.13-1 and 2.13.13-2. The exceptions include ASME Code stamping, and the use of ASME Code certified materials, which have been previously approved (ADAMS Accession No. ML091540454). The use of more recent versions of SNT-TC-1A, listed in Table 2.13.13-1, rather than a specific version of ASME code, is not inconsistent with the guidance of NUREG-1617. The staff finds all these exceptions listed in Tables 2.13.13-1 and 2.13.13-2 acceptable.

The staff finds the applicant's request to use an efficiency factor of 1.0 for longitudinal seam welds on the fuel compartment tubes using visual examination acceptable because both seam welds are made of thin gauge 3.4 mm (0.135") austenitic stainless steel (which is known for its high ductility and good welding characteristics) and the welds receive 100% VT examinations from both sides of the weld.

Note 40 on sheet 2 of licensing drawing 65200-71-01 permits the applicant to use progressive PT in lieu of volumetric examination for construction of the inner and outer shells provided that the CoC holder performs a flaw size analysis in conformance with Section XI, Division III, of the ASME Code. The weld metal deposition will be limited to the lesser of the critical flaw size or 3/8 of an inch. The staff finds this approach acceptable and follows the principal guidance in Interim Staff Guidance Document 18 (ISG-18), "The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage." Alternative welding configurations, for example cited on note 19 of sheet 2 of licensing drawing 65200-71-01, were also found to be acceptable to the staff, as these configurations must meet the applicable construction code requirements, match or exceed the strength of the weld design required in the design, and must be approved by the CoC holder.

The staff reviewed the properties of the materials listed in the application and found that they matched the properties listed in Section IID of the ASME Code. The staff finds the use of these materials acceptable and in compliance with the guidance in NUREG-1617.

2.2.2 Seals

All seal contact surfaces are stainless steel and are machined to a 32-root mean square or finer surface finish. It is reasonable to assume that the transportation time of the contents will be less than 6 months in any conceivable scenario, thus limiting the containment seal to an exposure of less than 1×10^4 rads (the threshold dose for fluoropolymers). The temperature of retraction (TR-10) of Parker FKM V1289-75 is specified to be no higher than -40^{o}C (-40^{o}F), which ensures that the seal will meet the requirements of 10 CFR 71.61(b)(1)(ii).

The staff finds that the low-temperature properties of the sealing material will be adequate. The maximum temperature of the containment O-ring seal under NCT is 96^{o}C (205^{o}F). The maximum temperature during HAC is 226^{o}C (439^{o}F) and 135^{o}C (275^{o}F) during the steady state cool-down conditions following HAC. The properties of the elastomeric seal chosen by the applicant will be sufficient to operate under these temperatures. The staff has reviewed the properties of the elastomeric material against widely used literature sources and determined its adequacy.

2.2.3 Attachments including impact limiters

The package is lifted using two removable martensitic stainless steel trunnions (SA-182 F6NM) which are bolted to the cask body using eight ASME-540 Gr. B23 CL.1, 1-8 UNC bolts. The impact limiters are constructed from ASTM A240 Type 304 stainless steel and welded according to American Welding Society (AWS) D1.6:2007 "Structural Welding Code – Stainless Steel," as specified by the welding code on note 16 of drawing 65200-71-20. The use of the AWS welding code has been accepted by the staff in previous applications for impact limiters on spent nuclear fuel packagings, e.g., HI-STAR 60 (ADAMS Accession No. ML091540454). The eight attachment bolts to the impact limiter are made of ASME SA-540 Gr. B23 Class 1 steel in accordance with Subsection NF of the ASME Code, which is consistent with the guidance found in Table 1-1 of NUREG-1617, "Fabrication, Examination, and Testing Criteria for SNF Transportation Packages based on the B&PV Code." These bolts have a safety classification of B, which is in accordance with the guidance in Section 5.5.2 of NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety.” Fusible plugs and seals on the impact limiters are made of RILSAN BMN-68 (or equivalent) and N406-60 nitrile, respectively. The fusible plugs and seals are meant to fail in the event of a hypothetical accident to prevent pressure build up within the impact limiters. The staff reviewed the general characteristics properties of these material classes (nylons and nitrile rubbers) and found them acceptable for the design, as these materials will melt or burn during under HACs. As part of the acceptance test for the packaging, Section 8.1.5.3 requires that each impact limiter container will be pressurized to a pressure between 2.0 and 3.0 psig and the weld seams and penetrations will be tested for leakage using a soap bubble test. If bubbles are detected, the weld will be repaired and the soap bubble test re-performed.

2.2.4 TN-LC-MTR Basket

The TN-LC-MTR basket is constructed primarily of ASME SA-240 Type 304 stainless steel according to Subsection NG of the ASME Code. Aluminum, ASTM B209 Type 6061, surrounds the basket to provide fit-up with the inner shell of the TN-LC. The materials and code of construction for the basket follow the guidance of NUREG-1617. The applicant assumed that the mechanical properties of the 6061 aluminum alloy are in the annealed state due to the elevated operating temperatures, which the staff finds acceptable for a heat-treatable alloy. The aluminum is primarily used for fit-up and heat transfer and is procured as a Safety Category “A” component. The staff verified the thermo-mechanical properties of basket materials against Section IID of the ASME Code and well-established literature values and finds their use in the packaging acceptable.

2.2.5 TN-LC-TRIGA Basket

The TN-LC-TRIGA basket is constructed primarily of ASME SA-240 Type 304 stainless steel according to Subsection NG of the ASME Code. Aluminum railing, ASTM B209 Type 6061, fits between the basket and the inner shell of the TN-LC. The materials and code of construction for the basket follow the guidance of NUREG-1617. The applicant assumed that the mechanical properties of 6061 aluminum alloy are in the annealed state due to elevated operating temperatures, which the staff finds a valid assumption given the operating environment of the material. The aluminum is primarily used for fit-up and heat transfer and is procured as a Safety Category “A” component. The staff verified the mechanical and physical properties of the aluminum and finds its use in the packaging acceptable. Piping, ASME SA-213 or SA-312 Type 304, is used as supporting ring material around the aluminum railing. The staff has reviewed the properties of ASME SA-213 and SA-312 Type 304 stainless steel and found them acceptable for the application. The staff verified the thermo-mechanical properties of basket materials against Section IID of the ASME Code and well-established literature values and finds their use in the packaging acceptable.

2.2.6 TN-LC-NRUX Basket

The TN-LC-NRUX basket is constructed primarily of ASME SA-240 Type 304 stainless steel according to Subsections NG and NF of the ASME Code. The materials and code of construction for the basket follow the guidance of NUREG-1617. The staff has compared the properties of ASME SA-479 and SA-240 Type 304 stainless steels listed in the SAR against the ASME Code and found them acceptable. The tube cap and spacer blocks on the basket are made of ASTM B209 Type 6061 and are Safety Category “A” components. The tube cap is designed to transmit longitudinal fuel assembly loads to the cask body; the spacers are used to aid fit-up of the basket and are procured as Safety Class A and B components. The staff verified the thermo-mechanical properties of basket materials against Section IID of the ASME Code and well-established literature values and finds their use in the packaging acceptable.

2.2.7 TN-LC-1FA Basket

The TN-LC-1FA basket is constructed primarily of ASME SA-240 Type 304 stainless steel according to Subsection NG of the ASME Code, which is the recommended code of construction in NUREG-1617. Cylindrical aluminum railing, ASTM B209 Type 6061, surrounds the basket to provide fit-up with the inner shell of the TN-LC. Sixteen ASME SA-540 Gr.B23 Class 1 bolts attach the outer aluminum railing to the inner stainless steel compartment. The applicant assumed that the mechanical properties of the 6061 aluminum alloy are in the annealed state due to elevated operating temperatures, which the staff finds acceptable for a heat-treatable alloy. The aluminum is primarily procured as a Safety Category A component. The materials and code of construction for the basket follow the guidance of NUREG-1617. The staff verified the thermo-mechanical properties of basket materials against Section IID of the ASME Code and well-established literature values and finds their use in the packaging acceptable.

2.2.8 TN-LC-1FA Basket Inserts

A sleeve of ASME SA-240 Type 304 stainless steel can be inserted to accommodate one BWR assembly or a 25-pin fuel can made of ASME SA-240 Type 304. The 25-pin fuel can is welded in accordance with Subsection NG of the ASME Code, from ASME SA-240 Type 304 stainless steel. The exterior railings are made of SA-240 NITRONIC 60 stainless steel. The lid of the can is attached with ASME SA-540 Gr.B23 Class 1 socket head cap screws. Additional shielding on the 25-pin fuel can is composed of ASTM B29 lead. The staff verified the thermo-mechanical properties of these materials against Section IID of the ASME Code and finds their use in the packaging acceptable.

2.2.9 Shielding materials

Primary gamma shielding is provided by ASTM B29 lead, which is a commonly used material in transportation packages. The ASTM standard requires a minimum of 99.9% elemental lead for its composition. The staff finds the use of this material consistent with the guidance in NUREG-1617. The annular lead shield is cast-in-place between the inner and outer shells of the TN-LC. The upper end structure is nominally 3.5 inches thick. The shield at the bottom is made from lead sheet packed firmly into place or poured, with a nominal thickness of 3.5 inches. The bottom lead cavity is closed using a 1.5-inch stainless steel plate.

Neutron shielding is provided by VYAL B, a proprietary vinyl ester resin mixed with alumina hydrate and zinc borate or Resin F, a borated reinforced polymer. The mixing methodologies, casting of VYAL B and Resin F are described in Chapters 5.3.1, 5.3.2, and 5.3.3 and are referenced in Chapter 8.1.6.2, “Neutron Shield,” of the application. The minimum hydrogen and boron content of these materials is listed in Section 8.1.6.2, acceptance tests of the neutron shield are done by density measurements. Lead discs, which are not important to the safety of the package, are attached to the impact limiters but no shielding credit is taken for these lead discs in the application.

2.2.10 Package Contents

The TN-LC-MTR fuel basket holds fifty-four intact MTR fuel elements, each with ten to twenty-three flat or curved plates. The fissile fuel plates are an aluminum-uranium alloy or an aluminum alloy reinforced with uranium oxide or uranium silicide. The fuel plates are clad with aluminum. The initial uranium enrichment is up to 94.0 wt.% U^{235} with a maximum burn up of 620,000 GWd/MTU. MTR fuel elements with damaged cladding are authorized provided the total surface area of the damage does not exceed 5% of the total surface area of the damaged element. The loading of fuel with a 5% damaged surface area is consistent with previously approved applications (ADAMS Accession No. ML100820126). The applicant estimated that the thermal characteristics of the fuel are a composite of the characteristics of the cladding and the fissile fuel plate reinforced with uranium oxide or uranium silicide. The applicant assumed that the mechanical and thermo-physical properties of the aluminum used were identical to annealed 6061 aluminum alloy. In the engineering judgment of the staff, this is not entirely accurate; however, the staff recognizes that the margins for thermal damage occurring to the package contents are sufficiently wide so that there is no safety concern. The staff reviewed the thermal expansion coefficients of the fuel, including irradiation impacts used to determine if the fuel will fit in the basket. The staff found that the values cited were reasonable and based on NUREG/CR-7024, “Material Property Correlations: Comparisons between FRAPCON-3.4, FRAPTRAN 1.4,” and the ASME Code.”

The TN-LC-TRIGA basket holds up to 180 TRIGA Fuel Assemblies/Elements. TRIGA fuel elements consist of a uranium/zirconium hydride matrix clad with aluminum or 304 stainless steel. TRIGA fuel follower control rods contain both boron carbide and fissile uranium hydride. The maximum burnup of the TRIGA fuel is 450,000 GWd/MTU. Only intact fuel with no known or suspected cladding defects greater than hairline cracks or pinhole leaks is permitted. The applicant cited “Experimental Heat Transfer Analysis of the IPR-R1 TRIGA Reactor,” International Atomic Energy Agency Publications, 2007 as a source for the thermal properties of uranium hydride. The staff finds this source acceptable for use, particularly given the large margin of safety for the thermal properties of the TRIGA fuel. It assumed the mechanical and thermo-physical properties of the aluminum used in TRIGA fuel were identical to annealed 6061 aluminum alloy. In the engineering judgment of the staff, this is not entirely accurate; however, the staff recognizes that the margins for thermal damage occurring to the package contents are sufficiently wide so that there is no safety concern. The staff reviewed the thermal expansion coefficients of the fuel, including irradiation impacts used to determine if the fuel will fit in the basket. The staff found that the values cited were reasonable and based on NUREG/CR-7024, “Material Property Correlations: Comparisons between FRAPCON-3.4, FRAPTRAN 1.4,” and the ASME Code.

The TN-LC-NRUX basket is designed to accommodate 26 NRU or NRX intact fuel assemblies. The fissile material is a uranium-aluminum alloy, clad in aluminum. The maximum U^{235} enrichment and depletion is 93% and 80%, respectively. Both intact fuel assemblies (fuel assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks) and damaged fuel assemblies are permitted. The extent of the damage is limited such that the total surface area of the damaged cladding does not exceed 5% of the total surface area of each rod. The applicant cited IAEA-TECDOC-643, “Research Reactor Core Conversion Guidebook, Vol. 4: Fuels (Appendices I-K)” for the thermal conductivities of the uranium-aluminum alloy. The applicant assumed that the mechanical and thermo-physical properties of the aluminum used were identical to annealed 6061 aluminum alloy. In the engineering judgment of the staff, this is not entirely accurate; however, the staff recognizes that the margins for thermal damage occurring to the package contents are sufficiently wide that there is no safety concern. The staff reviewed the thermal expansion coefficients of the fuel, including irradiation impacts used to determine if the fuel will fit in the basket. The staff found that the values cited were reasonable and based on NUREG/CR-7024, "Material Property Correlations: Comparisons between FRAPCON-3.4, FRAPTRAN 1.4," and the ASME Code.

The TN-LC-1FA basket assembly contains either: (a) one intact PWR or BWR fuel assembly with a maximum burnup of 62 GWd/MTU (Poison Rod Assemblies –PRAs-, a cluster of absorber rods containing B_4C pellets, are inserted into the guide tubes of the PWR fuel assembly for criticality control), or (b) one 25 pin can with up to twenty-five individual intact fuel rods of PWR, BWR, MOX or EPR fuel, limited to 90 GWd/MTU with a maximum U^{235} enrichment on all of the fuels of 5 wt%. The applicant cited thermal conductivity values for the irradiated fuel described in NUREG/CR-6534, "FRAPCON-3 Code Updated with MOX Fuel Properties" to confirm the thermal properties of the high burnup spent fuel, which the staff finds acceptable. The thermal conductivities of the zirconium cladding were taken from NUREG/CR-6150, the most recent version of MATPRO: A Library of Materials Properties for Light-Water-Reactor Accident Analysis." The staff reviewed the thermal expansion coefficients of the fuel, including irradiation impacts used to determine if the fuel will fit in the basket. The staff found that the values cited were reasonable and based on NUREG/CR-7024, "Material Property Correlations: Comparisons between FRAPCON-3.4, FRAPTRAN 1.4," and the ASME Code. While there is insufficient data to support the mechanical properties of commercial high burnup fuel, the applicant chose very conservative assumptions for fuel reconfiguration and subcriticality of the fuel will be maintained even if the fuel reconfigures under NCT or HAC.

The maximum cladding temperature under HAC (Table 3-7) is 594°F (368°C). This temperature is below the temperature at which zirconium-based claddings will be damaged. Aluminum-based claddings may experience some deformation at these temperatures; however, subcriticality of the fuel will be maintained even if the fuel is damaged under HAC. The maximum temperature for aluminum-based claddings under NCT (Table 3-2) is 266°F (130°C), which is significantly below the ASME code limit of 400°F for structural use of aluminum; therefore the staff finds the NCT temperatures of the aluminum-based cladding acceptable. The maximum temperature for zirconium-based claddings under NCT is 542°F (283°C), which is significantly below the limit of 400°C recommended in ISG-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel." Therefore, the staff finds the NCT temperatures of the zirconium-based cladding acceptable.

2.2.11 Galvanic Corrosion

The Model No. TN-LC package body, baskets, and contents are made of materials which are significantly resistant to corrosion in their anticipated environments, e.g., water and dilute solutions of boric acid. The expected time frame for components of the packaging to be submerged is on the order of a few days. There is no concern regarding galvanic corrosion taking place between the components of the package.

2.2.12 Criticality Control

Neutron absorbing poison plates consisting of either a boron-aluminum alloy, a boron carbide aluminum metal matrix composite or Boral® are used for criticality control in the TN-LC-TRIGA and TN-LC-1FA basket designs. The minimum B^{10} areal density of the poison plate materials is specified in the Certificate of Compliance. Poison Rod Assemblies (PRAs), containing B_4C

pellets inserted into the guide tubes of PWR fuel assemblies intended for transport, are also proposed to maintain criticality for PWR assemblies under HAC.

2.2.13 Impact Limiting Materials

The stainless steel-encased wood impact limiters utilize balsa wood and red wood. The properties of wood are highly dependent on wood species, density and moisture content. The applicant established a density range of 10 –12 lb/cu. ft. and 18.7 – 27.5 lb/cu. ft. for balsa wood and red wood, respectively. Moisture contents for the woods are between 6 –10%. These specifications are listed on Sheet 1 of Licensing Drawing 65200-71-20. Five samples of wood per lot will be tested to verify that each lot of wood meets these requirements. Compression tests will be run on five samples of balsa wood and red wood to verify that each lot of wood meets the required crush strengths specified in Section 8.1.9, "Impact Limiter Wood Test."

The wood in the impact limiters is joined using a phenol-resorcinol type standard wood-glue, which is more resistant to fire and aging than wood, as stated in the Chapter 10 of USDA's Wood Handbook, "Adhesives with Wood Materials Bond Formation and Performance" (General Technical Report FPL–GTR–190). The structural analysis does not take credit for the glue inside the impact limiters. Therefore, the staff finds the use of phenol-resorcinol glues acceptable for joining the blocks in the impact limiter.

Conservative wood thermal properties were used in the thermal analysis to assess the worst-case scenario. These properties include: minimum and maximum thermal conductivity, thermal diffusivity, specific heat, density, and linear charring rate. The properties for both balsa wood and red wood were considered given the acceptable density and moisture content ranges and expected material property variations. Then, a conservative thermal property value was selected. For example, there is a wide variation between the minimum thermal conductivity of the proposed balsa wood and the maximum thermal conductivity of the proposed red wood. In this case, the safety analysis considered the minimum thermal conductivity during normal conditions of transport and the maximum conductivity during the hypothetical accident conditions. Thermal modeling indicated that the lower linear char rate of 55 mm/hr documented for red wood would be more conservative than the expected higher linear char rate of balsa wood.

2.2.13.1Balsa wood

Compressive tests shall demonstrate that the balsa wood has an average crush strength parallel to the grain between 1498–1900 psi, with a minimum lockup stain of 80%. The staff finds the mimimum lock up strains acceptable, based on Figure 6 of the Jet Propulsion Laboratory (JPL) Techical Report 32-1295, "Effects of Sterizlation on the Energy-Dissipating Properties of Balsa Wood."

The applicant cited mechanical testing on samples of balsa wood by BalTek Inc. with varying densities and produced a linear best-fit to estimate the compressive strength of balsa wood parallel to the grain. In a similar manner the applicant provided data for the perpendicular properties of the balsa wood. The staff also reviewed JPL Technical Report 32-944, "Environmental and Physical Effects on the Response of Balsa Wood as an Energy Dissipator [sic]," which indicates a small effect (± 5%) of moisture content on the compressive strength and specific energy when the moisture is held between 6 and 10%. JPL Technical Report 32-944 was, also the basis for the change of energy absorption by a factor of 1.4 and 0.9 at -40^{o}F (-40^{o}C) and 150^{o}F (66.6^{o}C), respectively for balsa wood. The staff is reviewing the properties of wood for use as an impact limiting material as a generic technical issue. Based on engineering judgment, the staff finds the properties of the wood as listed in the SAR sufficient to maintain containment under hypothetical accident conditions.

2.2.13.2 Red wood

Compressive tests shall demonstrate that the red wood has an average crush strength parallel to the grain between 5000 – 7500 psi, with a minimum lock stain of 60%. The staff finds the mimimum lock up strains acceptable, based Table I of NUREG/CR-0322, "The Effects of Temperature on the Energy-Absorbing Characteristics of Red wood."

The mechanical properties of red wood listed in the Safety Analysis Report were derived from Chapter 5 of the Wood Handbook (FPL-GTR-113). The applicant assumed a linear relationship between the ratio of the compressive strength of red wood and specific gravity. The relationship in the Wood Handbook between specific gravity and compressive strength is 7210 psi ($S.G^{0.94}$). The assumed linearity is acceptable to the staff as a linear, best-fit trend line matches the relationship over the range of specific densities required for the red wood with a chi-squared value greater than 0.99. Citing NUREG/CR-0322, the applicant assumed a 10% decrease in compressive strength and energy absorption and from 70°F (21°C) to 230°F (110°C) and a 30% increase in compressive strength and energy absorption and from 70°F (21°C) to -40°F (-40°C). The strain/strain curves for the impact limiting wood are based on the highest and lowest compressive stresses measured for the wood and then modified correspondingly for temperature affects. The applicant cited mechanical testing on red wood samples from "Compression Tests of Red wood Samples," Forest Product Laboratory, 1985. This data set clearly shows a positive correlation between the red wood density and compressive strength. The staff is reviewing the properties of wood for use as an impact limiting material as a generic technical issue. Based on engineering judgment, the staff finds the properties of the wood as listed in the SAR sufficient to maintain containment under hypothetical accident conditions.

2.3 Fabrication Stresses

Appendix 2.13.4 to the application calculates fabrication stresses in the inner shell of the package body due to lead pouring and subsequent cool down to room temperature. The hydrostatic pressure of a column of molten lead results in a maximum compressive stress of 748.3 psi in the inner shell. Section 2.13.14.3 considers the lead freezing temperature of 620°F and room temperature of 70°F to estimate a constant stress of less than 400 psi in the lead for the lead cooling accomplished in about a week. For an average hoop stress of 400 psi in the 3.5-inch thick lead, an interface pressure of 140 psi develops between the lead and cask inner shell. This corresponds to a hoop stress of 1,400 psi in the inner shell, which is negligibly small and, is therefore, acceptable.

2.4 General Standard for All Packages

2.4.1 Minimum Package Size

The overall package dimensions of approximately 230 inches long and 66 inches in diameter exceed the minimum dimension requirement of 4 inches (10 cm), which meets the requirements of 10 CFR 71.43(a) for minimum size.

2.4.2 Tamper-Indicating Features

The primary access path into the package is through the closure lid. Like the small access paths of the vent port, test port, drain port, and bottom plug, they are all covered by the impact limiter during transportation. A security wire seal is installed in the front impact limiter above an attachment bolt prior to each shipment. The presence of this seal demonstrates that unauthorized entry into the package has not occurred. This satisfies the tamper-proof requirements of 10 CFR 71.43(b).

2.4.3 Positive Closure

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

2.5 Lifting and Tie-Down Standards for All Packages

2.5.1 Lifting Devices

The package is lifted by two upper removable trunnions. Drawings 65200-71-01, sheets 5 and 9, delineate geometries of the trunnions and trunnion attachment blocks, which are welded to the package body. The eight 1-8UNC attachment bolts are designed to act in tension as the shear load is supported by the tight-fitting trunnion flange shoulder and a recess in the trunnion attachment block. Section 2.13.5.2 of Appendix 2.13.5 to the application considers the package design weight of 49,000 lbs (without impact limiters attached), with a dynamic load factor of 1.15, in evaluating the vertical cask lift. At six times the package design weight, a concentrated vertical force of 169,050 lbs is applied to each trunnion for the single-load-path lifting configuration.

Using the “Bijlaard’s method,” Table 2.13.5-5 of Appendix 2.13.5 of the application calculates a maximum local stress intensity of 29.5 ksi induced in the outer shell by the trunnion. Table 2.13.5-6 lists the maximum calculated stress intensities at the trunnion shoulder, flange, and hub. The stress intensities, which are always greater than the corresponding maximum principal stresses, are all shown below the material yield strength of 43.3 ksi. This satisfies the 10 CFR 71.45(a) requirement of a minimum safety factor of three against the material yield strength. Additionally, the calculated stress safety margin of 0.18 for trunnion shoulder is smaller than that of 0.47 for the outer shell local stress. Thus, the staff agrees with the applicant’s conclusion that an excessive load on the trunnion will cause failure of the trunnion and will not impair the ability of the package to perform per 10 CFR 71.45(a) requirements.

2.5.2 Tie-Down Devices

During transport, the package is secured to a transportation skid. Section 2.5.2 of the application notes that the vertical and transverse package transport loads, per 10 CFR 71.45(b)(1), are reacted by saddles and tie-down straps. The horizontal force component at 10 g along the direction the vehicle travels, however, is resisted by a shear key extending from the transportation skid into the pocket bearing block welded to the package body outer shell. Section 2.13.5.2.2 of Appendix 2.13.5 performs stress analyses of the shear key assembly. As summarized in Section 2.5.2, all stresses along the load path and in the shear key slot have large safety margins against material yield strengths. The minimum margin of safety at 0.86 occurs in the base metal junction between the pad plate and the cask outer shell. This suggests that, under excessive load, the weld between the shear key pad plate and the package outer shell would fail in shear. That is, the package body would stay intact without impairing the ability of the package to meet the other requirements of 10 CFR Part 71. Therefore, the package tie-down devices meet the requirements of 10 CFR Part 71.45(b)(1) and (3).

2.6 NCT

2.6.1 Heat

Considering hot environmental conditions, Chapter 3 of the application calculates the package maximum normal operating pressure (MNOP) of 16.9 psig and maximum temperatures at various package locations. These provide the basis for assuming conservatively a baseline cask internal pressure of 30 psig, which is greater than the MNOP, for the package stress analysis. Table 3-10 of the application summarizes these maximum component temperatures considered significant for establishing the "at-temperature stress allowables" for stress margins evaluation.

Section 2.6.1.2 of the application summarizes the differential thermal expansion (DTE) evaluations of package components, in Appendix 2.13.10, for possible interference among package components. Tables 2.13.10-7 through 2.13.10-12 present the maximum allowable irradiated fuel lengths and minimum axial and radial basket component clearances. On the basis of these results, the applicant concludes and the staff agrees that there is adequate clearance between the various components of the package to allowable free thermal expansion and no significant stress will develop in the Model No. TN-LC package due to thermal expansion.

Section 2.6.1.3 of the application presents the stress evaluation, using finite element models, for the package body subject to the combined loadings of bolt preload, package dead load, internal pressure of 30 psig, and thermal hot condition of 100°F. The evaluation, shown in Table 2.13.1-23 of Appendix 2.13.1 demonstrates that all calculated stress values are below the allowables.

The staff reviewed the structural performance of the package under the heat conditions and concludes that the DTE and stress effects have properly been evaluated. Thus, the requirements of 10 CFR 71.71(c)(1) are satisfied.

2.6.2 Cold

Section 2.6.2 of the application evaluates the impacts of a cold environment on the package performance by considering an ambient temperature of -40°F combined with bolt preload, package dead load, and an external pressure of 25 psig. The evaluation, shown in Table 2.13.1-24 of Appendix 2.13.1 demonstrates that all calculated stress values are below the allowables.

On the basis of the above, the staff has reasonable assurance to conclude that the package structural performance is adequate for meeting the cold condition requirements of 10 CFR 71.71(c)(2).

2.6.3 Reduced External Pressure

Section 2.6.3 of the application notes that the effect of reduced external pressure of 3.5 psia is bounded by the conservatively assumed MNOP of 16.9 psig, which amounts to a net package pressure loading of 28.1 psi (16.9 + 14.7 – 3.5 = 28.1). Considering the baseline internal pressure at 30 psig, the combined loadings, which include also bolt preload, horizontal dead load of the package mounted in the transport configuration, and thermal hot, result in calculated stresses all below the allowables. Thus, the requirements of 10 CFR 71.71(c)(3) for reduced external pressure are satisfied.

2.6.4 Increased External Pressure

For an increased external pressure at 20 psia ambient, Section 2.6.4 of the application notes that the case of a baseline external pressure loading of 25 psig is conservatively combined with the package preload, horizontal package dead load, and the thermal cold condition of -20°F for the stress analysis. The resulting stresses as shown in Table 2.13.1-25 of Appendix 2.13.1 are all below the allowable. Thus, the increased external pressure will not adversely affect the package structural performance, and the requirements of 10 CFR 71.71(c)(4) are satisfied.

2.6.5 Vibration

Although transportation packages need not to be evaluated for shock loading, Section 2.6.5 of the application compares shock loading standards of ANSI N14.23 for truck transport to those of NUREG 766510 for rail transport and selected the bounding decelerations associated with the latter for evaluating the Model No. TN-LC package. This results in considering concurrently applied longitudinal and transverse accelerations of 4.7 g on the package for load combination stress analysis, which also include bolt preload, internal pressure, and thermal hot as well as cold conditions. The stress results under those load combinations, as listed in Tables 2.13.1-29 and 2.13.1-30, are calculated to be below the allowables.

For vibration loading, the ANSI N14.23 truck transport standards are determined by the applicant to be governing, which correspond to the package accelerations of 0.6 g and 0.3 g applied in the vertical and transverse directions, respectively. As evaluated in Section 2.6.5 of the application, the maximum stress intensity was estimated to be about 1.5 ksi in the package containment boundary and 1.8 ksi at the outer shell component. These negligibly small stresses would sustain an infinite number of vibration cycles, thereby, demonstrating that the vibration load is not expected to cause structural damages to the packaging containment boundary.

The vibration loading, although with negligibly small effect, is considered together with other package operational loadings for fatigue evaluations of the packaging structural components. Section 2.6.13 and Appendix 2.13.6 of the application evaluate the containment boundary components' cumulative fatigue damages for 1000 shipments for the package. The loading sequence of events considered are the following: operating bolt preload, test pressure, road shock/vibration, pressure and temperature fluctuations, and 1-ft normal condition drop. As noted in the application, the total damage factor for the inner and outer shells is 0.027 and 0.087 for the rest of the containment vessel. They are far less than 1.0 and are acceptable to demonstrate that the fatigue effects on the Model No. TN-LC package containment boundary are acceptable. Section 2.6.12 and Section 2.13.2.6 of Appendix 2.13.2 of the application perform fatigue analysis of the lid and bottom plug bolts. The loading sequence of events considered are the following: operating bolt preload, test pressure, road shock/vibration, pressure and temperature fluctuations, and 1-ft normal condition drop. Using a strength reduction factor, KE, of 4 for high-strength alloy steel, the analysis demonstrated that the bolts will not fail due to fatigue at 75 round-trip shipments.

The above evaluations, in aggregate, demonstrate that the Model No. TN-LC package meets the requirements of 10 CFR 71.71(c)(5) for vibration conditions and tests.

2.6.6 Water Spray

Section 2.6.6 of the application notes that no structural degradation will result from water absorption, and, therefore, the staff agrees that the water spray condition is of no consequence to the package in meeting the requirements of 10 CFR 71.71(c)(6).

2.6.7 Free Drop

Section 2.6.7 of the application notes three drop orientations for package evaluations: a 1-ft end drop of the package on the bottom end, a 1-ft end drop on the lid end, and a 1-ft side drop. Appendix 2.13.12 determines deceleration g-loads of 45 g, 35 g, and 30 g for the lid top, lid bottom, and side drop configurations, respectively, for the package with impact limiter in place. These individual loading cases are then combined with other appropriate loadings, including environmental conditions, and effects of bolt preload, in the cask body stress analysis, as presented in Appendix 2.13.1. Tables 2.13.1-31 to 2.13.1-36 list the load combination stress intensity results. As summarized in Section 2.5.14, the highest stress intensity ratios, each defined as the calculated over the allowable, ranges from 0.32, which occurs in the bottom plug to 0.912, in the outer shell, for the package components. The maximum stress ratio for the inner shell, which is a major part of the containment boundary, is 0.645.

Appendix 2.13.8 to the application presents the structural analysis of the fuel baskets under the NCT 1-ft drop deceleration g-loads and thermal loads for the four fuel baskets, TN-LC-1FA, TN-LC-MTR, TN-LC-TRIGA, and TN-LC-NRUX. Section 2.13.8.3 considers the rigid body decelerations as calculated in Appendix 2.13.12, Table 2.13.12-7, and a dynamic load factor of 1.1 to establish the amplified inertia load of 38 g and 21 g for the cask end and side-drop conditions, respectively. However, a set of baseline g-loads, which envelop the calculated inertia loads are selected for analyzing bounding stress performance of the baskets. This includes the 50 g inertia force for the end and side drops of the TN-LC-1FA basket, 45g inertia force for the TN-LC-MTR and TN-LC-TRIGA basket and the 40 g and 30 g end and side drops, respectively, for the TN-LC-NRUX basket. Section 2.6.15.1 of the application summarizes the structural analysis results for these loading conditions with applicable boundary conditions. The staff reviewed the stress results for all baskets and agrees with the applicant’s conclusion that they meet the ASME Code, subsection NG requirements. Therefore, the baskets are structurally adequate for supporting and positioning the fuel in the cask under the NCT loading conditions.

On the basis of the above, the staff concludes that the package is capable of maintaining its structural integrity to meet the requirements of 10 CFR 71.71(c)(7).

2.6.8 Corner Drop

The corner drop test requirement of 10 CFR 71.71(c)(8) does not apply because the Model No. TN-LC package weighs more than 220 lbs.

2.6.9 Compression

Because the Model No. TN-LC package weighs more than 11,000 lbs, the staff agrees with the applicant's assessment that the TN-LC package need not be evaluated for meeting the 10 CFR 71.71(c)(9) requirements on compression test.

2.6.10 Penetration

Section 2.6.10 of the application notes a lack of sensitive external protuberances for the package and concludes that the 40-inch drop of a 13-lb steel cylinder is of negligible consequence to the package. The staff agrees with this assessment and concludes that the package needs not be evaluated explicitly for satisfying the requirements of the 10 CFR 71.71(c)(10) penetration test.

2.6.11 Lid Bolts and Bottom Plug Bolts

The ability of the closure bolts to maintain a leak-tight seal under the NCT is evaluated below.

Section 2.6.12 and Appendix 2.13.2 of the application notes that, for NCT, the lid bolts analyses follow the NUREG/CR-6007 provisions for the loading conditions including operating preload, gasket seating load, internal pressure, temperature changes, and impact loads. This involves the evaluation of bolt maximum average tensile and shear stresses, combined tensile and shear stresses, and stress ratio interaction equation values. Other performances, such as bolt head bearing stress and bolt thread engagement length are also evaluated. As detailed in Section 2.13.2.2 of Appendix 2.13.2, the closure lid is tightened to the package flange with 20 1-inch diameter SA-540 Grade B23 Class 1 high-strength alloy bolts. A bolt torque of 400 to 450 ft-lb applies a preload of 35,600 lbs to 40,000 lbs to the bolt. Section 2.13.2.9 of Appendix 2.13.2 notes the eight ½-inch diameter SA-540 Grade B23 Class 1 high-strength alloy bolts for the bottom plug, which are tightened by a bolt torque of 40 to 48 ft-lb. They are also evaluated using the same methodology as is for the closure lid bolts. Table 2.13.2-7 summarizes stress evaluation results for the lid bolt and bottom plug bolt, which demonstrate that all stress performance criteria are satisfied. Thus, the staff agrees with the applicant's assessment that a positive, compressive, load is maintained on the clamped joint for all NCT load combinations.

2.7 HAC

2.7.1 30-ft Free Drop

Section 2.7.1 of the application evaluates, by analysis for the HAC 30-ft free drops, the structural performance of the package components, including the impact limiters, the package body and fuel baskets.

Appendix 2.13.12 to the application presents packaging drop testing performance by analysis using LS-DYNA finite element modeling. Section 2.13.12.2 of Appendix 2.13.12 notes the similarity in design configuration and load paths between the TN-LC impact limiters and the 1/3-scale impact limiters used in the drop testing of the NUHOMS MPC197 transportation package (Docket No. 71-9302). Figures 2.13.12-1 through -3 demonstrates that the calculated rigid body time-history responses correlate well with the tested results for the end, side, and slapdown drops, respectively, in terms of response peak, pulse shape, and pulse duration. Table 2.13.12-1 shows that the maximum tested package rigid body decelerations and impact limiter crush depths are all enveloped by the calculated results, which ensure that LS-DYNA predicted package body decelerations are conservative for the package components structural evaluation. Thus, the data correlation as evaluated provides the basis for adapting the NUHOMS MPC197 LS-DYNA finite element modeling approach to calculating rigid body decelerations of the TN-LC package with respect to the worst package drop orientations. Section 2.13.12.4.1 presents modeling details for the TN-LC package components, including the body, impact limiter casing plates, gussets, wood segments, and attachment bolt assemblies**.** To add further credence to the approach, as implemented for the TN-LC package with markedly larger body length-to-diameter aspect ratio from that of the MP-197 package, Section 2.13.12.5 performed sensitivity analyses of the package rigid-body deceleration response with varied modeling parameters. The analyses, which include location of center of gravity, wood material shear modulus, friction coefficient between the unyielding surface and impact limiter shell, and variation of wood crush strength perpendicular to the grain direction, demonstrate acceptable results.

Section 2.13.12.6 of Appendix 2.13.2 presents the evaluation of the impact limiter attachment bolts, considering the peak bolt axial and shear stresses for each bolt for the 10^{o} slapdown drop with firm wood properties. Table 2.13.12-6 lists attributes of the 1-inch diameter bolts modeled with an elastic-plastic material model. The eight 1-inch diameter SA-540 GR. B23 CL. 1 bolts are modeled with circular cross section beams each interacting by gaps and contact spring elements with the bolt sleeve and tunnel with the impact limiter shell. Figure 2.13.12-65 displays a diagram of spatial integration areas associated with 16 integration points along the outer circumference and 4 along the radius for a more accurate determination of the extent to which the bolt section has gone plastic during a free drop event. Tables 2.13.12-17 and -18 summarize the bolt stress and plastic strain results for the 30-ft 10^{o} and 5^{o} slapdown drops, respectively. For the half-symmetry finite element analysis model, in addition to axial and shear stresses, the results include also minimum, average, and maximum strains for each attachment bolts of both the top and bottom impact limiters. By noting that only one out of the eight attachment bolts would have undergone an average plastic strain of greater than 0.1, the ASTM provision for the material, the staff agrees with the applicant's conclusion that the impact limiters will remain connected to the package during and after the postulated 30-ft drop. Additionally, the LS-DYNA analysis results also demonstrate large plastic deformations of the impact limiter components such as woods and shell casing in dissipating the package kinetic energy upon hitting the unyielding surface after the 30-ft drop accident.

Section 2.13.12.4.1 of Appendix 2.13.12 to the application notes that the package body is modeled as rigid and has the same mass, center of gravity, and moment of inertia as the actual package for calculating rigid body responses for the package components stress analysis. The center of gravity of the package is calculated to be at the package geometric center at the distance of 98.75 inch from either end. Table 2.13.12.7 lists the calculated maximum rigid body decelerations and the impact limiter crush depths for the wood material firm and soft conditions. The drop conditions considered are: 30-ft end drop, side drop, slapdown drops at 5^{o} and 10^{o} incident angles with respect to the target surface, and center of gravity-over-corner drop of the TN-LC package. As listed in the following table, the calculated maximum decelerations, for -40^{o}F temperature cold effects, are bounded by the baseline g-loads used for the package drop analyses.

Drop Scenario	Calculated g-Loads (Appendix 2.13.12)	Baseline g-Loads
30-ft End Drop 30-ft Side Drop	82.4 g - Axial 67.3 g - Transverse	95 g 75 g
30-ft Slapdown Drop 5^o 1st Impact	90.1 g -Transverse	130 g
30-ft Slapdown Drop 5^o 2nd Impact	117.8 g - Transverse	130 g
30-ft Slapdown Drop 10^o 1st Impact	86.8 g - Transverse	130 g
30-ft Slapdown Drop 10^o 2nd Impact	130.3 g - Transverse	130 g
30-ft CG-Over-Corner	61.7 g - Axial 14.9 g - Transverse	75 g ---

Table 2-9 of the application summarizes the load combination cases for which applicable internal or external pressures are considered together with baseline decelerations with various 30-ft drop scenarios. The hot at-temperature material properties, including stress allowable, are used for the package stress evaluation. Section 2.13.1.2.5 notes the use of pressure loading to simulate the package payload and impact limiter weight effects. The axial loads are applied as uniform pressure on the load area and the cosine distributed pressure functions are standard practices for the loading applied in the transverse, radial direction. Section 2.13.1.3 of Appendix 2.13.1 presents boundary conditions of the half symmetry finite element model used for solving package drop conditions.

Section 2.7.1.4 recognizes inertia load distribution in the package transverse direction, which varies linearly from zero to a peak baseline deceleration load of 130 g from one end of the package to the other for the 30-ft slapdwon drop. By noting that either the top or the bottom portion of the package body is subject to deceleration higher than the 75g baseline loading associated with the side impact, the applicant considers only calculated stresses in the top and bottom 33.6 inches of the package in evaluating the package slapdown drop stress performance. This stress evaluation approach is also examined by a confirmatory analysis considering a realistic transverse inertia load distribution in Section 2.13.1.0.1 of Appendix 2.13.1. Table 2.13.1-56 lists the stress results corresponding also to the stress contour plot of Figure 2.13.1-31, which demonstrate the applicability of the approach.

Section 2.13.1.8 of Appendix 2.13.1 considers ASME Code, Subsection NB and Appendix F, for stress evaluation of the package pressure retaining boundary and other body components, such as the outer shell, bottom lead cap, and gamma shielding cap for HAC. The stress evaluation considers general primary membrane stress intensity, P_m, local primary membrane stress intensity, P_l, and primary membrane-plus-bending stress, P_l+P_b. Section 2.13.1.8.1 summarizes the at-temperature stress acceptance criteria associated with the elastic and plastic analysis methods of the Appendices F-1341.1 and F-1341.2 provisions, respectively.

Section 2.13.1.8.2 notes the analyzed load combination cases, with applicable internal pressure, external pressure, and temperature, which include also Cases 41 and 42 for the 290 psig immersion test and 620^o F fire accident, respectively. Table 2.13.1-41 through Table 2.13.1-55 summaries stress results for the HAC loading conditions.

Table 2.13.1-46 of Appendix 2.13.1 lists the load combination stress intensity results for the complete package length for a 75 g side drop. Table 2.13.1-53 lists the load combination stress intensity results, considering only the top and bottom portions of the package analyzed with a 130 g side drop. The corresponding stress contour plots of Figures 2.13.1-28 and -29 demonstrate that critical stresses in the inner shell does occur at the center of the package undergoing the 75 g side drop and those in the outer shell at the package top or bottom end for the slapdown drop with the 130 g baseline peak deceleration.

On the basis of the evaluation above, the staff agrees with the summary conclusion of Section 2.13.1.10.2 of Appendix 2.13.1 to the application that the HAC loads will not result in any structural damage to the package and the containment function of the package will be maintained.

Appendix 2.13.3 performs ANSYS finite element analyses to evaluate the lead slump and the containment boundary buckling capability. Section 2.13.3.5 summarizes the analysis results. The ANSYS computations are stable under the applied loads before the non-converged solution occurs, at about 300 g, for both the lid end and bottom drop. Thus, the staff agrees with the applicant's conclusion that the 300 g load is a conservative buckling capability estimate for the package inner shell containment boundary. Section 2.13.3.6 provides that the maximum calculated longitudinal gap, caused by lead slump, is 1.129 inches during the HAC lid-end drop with internal pressure. The maximum calculated gap in the top and bottom end gamma shielding during the side drop is 0.155 inches. The effect of these gaps is evaluated in Chapter 5 of the application.

Appendix 2.13.7 performs LS-DYNA finite element analyses to evaluate the closure lid and bolts due to delayed end-drop impact for an assumed gap of 1 inch between the package and the internal of the basket and fuel. The analysis approach follows that for an approved TN transportation cask, which calculates a maximum separation of 0.047 inches, which occurs for a short duration, at the center of the lid seals and bolt stress intensities meeting NUREG/CR-6007 stress acceptance criteria. Considering the seal maximum allowable decompression of 0.04 inch, the applicant notes that the seal loses contact with the flange for about 5 milliseconds, which may result in an insignificant amount of helium or radioactive material leaking from the package cavity. As displayed in Figure 2.13.7-24, the negligibly small residual seal separation of less than 0.001 inch ensures that the lid remains sealed after the HAC end drop.

Appendix 2.13.8 presents structural analysis of the fuel baskets under HAC for the 30-ft cask drop deceleration g-loads and thermal loads for the four fuel baskets, TN-LC-1FA, TN-LC-MTR, TN-LC-TRIGA, and TN-LC-NRUX. Section 2.13.8.3 considers the rigid body decelerations as calculated in Appendix 2.13.12, Table 2.13.12-7, and a dynamic load factor of 1.1 to establish the amplified inertia load of 91 g and 143 g for the end- and side-drop conditions, respectively. However, a baseline deceleration of 150 g, which envelops the calculated inertia loads, is selected for analyzing the bounding stress performance of the baskets. Section 213.8.4 of Appendix 2.13.8 presents stress acceptance criteria in accordance with the Level D Service limits for support components in Appendix F of Section III of the ASME Code. As loading and structural configurations dictate, both hand calculations and ANSYS finite element computer modeling are used to calculate the structural performance, including the weld adequacy and the margin for critical loads for buckling of the baskets. The staff reviewed the stress analyses and results for the baskets and agrees with the applicant's conclusion in Section 2.7.8.2 of the application that the basket designs meet the ASME Code, Subsection NG and Appendix F requirements. Therefore, the baskets are structurally adequate for supporting and positioning the fuel in the package under HAC loading conditions.

On the basis of the evaluations above, the staff concludes that the package is structurally capable of meeting the requirements of 10 CFR 71.73(c)(1).

2.7.2 Crush

The application notes and the staff agrees that the weight of the Model No. TN-LC package exceeds 1,100 lbs and the crush test specified in 10 CFR 71.73(c)(2) does not apply.

2.7.3 Puncture

Section 2.7.3 of the application argues that the impact limiters will protect the ends of the package body for the 40-inch drop onto the puncture bar and considers that the most severe damage to the package body will occur on the outer cylinder shell midway between the impact limiters. Using the Nelms equation for the lead-backed shell, the applicant calculates a required thickness of 0.656 inches to preclude shell puncture. This provides the basis for the staff to agree with the applicant's conclusion that the 1.5-inch thick outer shell of the TN-LC package is structurally capable of meeting 10 CFR 71.73(c)(3) requirements on the puncture test.

2.7.4 Thermal

See Section 3.0 of this Safety Evaluation Report (SER) on thermal performance of the package.

2.7.5 Immersion - Fissile Material

The criticality evaluation presented in Chapter 6 of the application assumes optimum hydrogenous moderation of the contents. Thus, the staff agrees with the applicant's conclusion that the effects and consequences of water in-leakage are conservatively addressed to meet the requirements of 10 CFR 71.73(c)(5) immersion test.

2.7.6 Immersion - All Packages

As reviewed in the next paragraph, the TN-LC package is structurally adequate for the deep water immersion test pressure of 290 psig. This pressure is much higher than the equivalent hydrostatic pressure of 21.7 psig associated with the 50-ft water immersion test. Therefore, the staff agrees with the applicant's conclusion that the immersion test does not need to be evaluated to satisfy the requirements of 10 CFR 71.73(c)(6).

2.7.7 Deep Water Immersion Test

Section 2.13.1.8 of Appendix 2.13.1 to the application identifies load combination Case 42 for evaluating the deep water immersion test for an external pressure of 290 psig. Table 2.13.1-54 lists the calculated maximum stress intensities, which are negligibly small. On this basis, the staff concludes that the package containment boundary is capable of resisting the 290 psig water pressure for a period of not less than one hour without collapse, buckling, or in-leakage of water to meet 10 CFR 71.61 requirements.

2.8 Accident Conditions for Air Transport of Plutonium

The application states that this section does not apply to the TN-LC transport package because the package will not be transported by air.

2.9 Accident Conditions for Fissile Material Packages for Air Transport

The application states that this section does not apply to the TN-LC transport package because the package will not be transported by air.

2.10 Special Form

The application states that this section does not apply to the TN-LC transport package because the payloads are not considered to be special form.

2.11 Fuel Rods

MTR, NRU, and NRX fuel elements were modeled, under NCT and HAC, with reduced pitch, until the fuel rods contact, and with increased pitch, up to the maximum allowed by the basket structure. The most reactive credible configuration occurs by maximizing the distance between fuel elements because it maximizes the moderation and hence the reactivity. TRIGA fuel elements were modeled by ignoring the plenums and end fittings and therefore assuming total axial collapse of the plenums. This configuration bounds the geometry for any credible scenario under HAC in which the fuel is damaged: since the fuel assemblies fill most of the lateral void space, lateral bending of the fuel assemblies under HAC is negligible and lateral fuel damage is not expected.

Section 1.1.2 of the application notes that reconfiguration of fuel assuming water in-leakage is employed to demonstrate subcriticality under HAC to meet the 10 CFR 71.55(e) requirements.

Although the applicant assumed total fuel reconfiguration for NCT, the mechanical properties of the fuel cladding with regular burnup are known to be capable of retaining sufficient ductility to ensure continuing maintenance of original fuel geometry during NCT. Section 1.1.2 notes the uncertainty associated with the properties of the zirconium-based cladding materials for the high burnup BWR and PWR spent fuel assemblies. As such, the applicant performs criticality, thermal, and shielding analyses with and without considering fuel reconfiguration to demonstrate the adequacy of the design to meet the intent of the 10 CFR 71.55(d)(2) requirements, which stipulates that, under NCT, “the geometric form of the package contents would not be substantially altered.” To provide defense-in-depth insights into the structural performance of the high burnup fuel under NCT, Appendix 2.13.12 to the application considers available zirconium-based cladding materials data and use the finite element method to also analyze fuel rod structural performance under the NCT 1-ft side and end drops. The results of these analyses provide for a minimum safety factor of 1.57 against the clad yield strengths, which occurs in the 9x9 BWR fuel assembly for side drop. The large stress margins, therefore, provide reasonable assurance for the staff to agree with the applicant’s conclusion that the fuel will maintain its structural integrity during NCT for meeting the intent of the 10 CFR 71.55(d)(2) requirement.

2.12 Evaluation Findings

The staff reviewed the structural and materials performance of the Model No. TN-LC package. Based on the statements and representations contained in the application, the responses to the staff requests for additional information and the conditions given in the Certificate of Compliance, the staff finds that the TN-LC package meets the requirements of 10 CFR Part 71.

3.0 THERMAL REVIEW

3.1 Review Scope and Objective

The staff verified that the package performance has been adequately evaluated for the tests specified under NCT and HAC and that the package design satisfies the thermal requirements of 10 CFR Part 71. Confirmatory calculations were performed by the staff for certain aspects of the design.

3.2 Description of the Thermal Design

3.2.1 Packaging Design Features

The package is designed to transport a variety of payloads including 1 intact PWR assembly, 1 intact BWR assembly, up to 25 intact PWR or BWR fuel rods in a 25 pin can basket, up to 26 intact NRU Fuel assemblies, up to 26 intact NRX Fuel assemblies, up to 54 intact MTR Fuel assemblies, and up to 180 intact TRIGA Fuel Elements.

3.2.2 Codes and Standards

Where appropriate, codes and standards were referenced by the applicant. For standard materials, the ASME Code is referenced by the applicant.

3.2.3 Content Heat Load Specification

The total decay heat of any content shipped in the Model No. TN-LC package will not exceed 3.0 kW. The maximum total decay heat load limits are specified for each basket design, as follows:

maximum heat load	basket design
0.39 kW (15 watts per assembly)	TN-LC-NRUX
1.5 kW (25 watts per fuel element)	TN-LC-MTR
1.5 kW (8 watts per assembly/ element, total of 288 watts per basket)	TN-LC-TRIGA
2.0 kW	TN-LC-1FA/1-BWR
3.0 kW	TN-LC-1FA/1-PWR
3.0 kW (120 W/rod for 25 rods) or 1.98 kW (220 W/rod for 9 rods)	TN-LC-1FA/pincan

3.2.4 Summary Tables of Temperatures

The summary tables of the temperatures of the package components, Tables 3-1 through 3-10 of the application, were verified to include the impact limiters, containment vessel, seals, gamma shielding, and neutron absorbers, and were consistent with the temperatures presented throughout the application for both NCT and HAC. The staff confirmed that the summary tables contained the design temperature limits for each of the critical components for both NCT and HAC.

For HAC, the applicant reported the maximum transient temperatures for essential components, as well as the approximate time at which the maximum temperatures were reached (see Section 3.4.3 and Table 3-6 of the application). For the hypothetical fire accident condition, all components remained below their material property temperature limits with the exception of the package impact limiters and the neutron shielding resin, neither of which is essential for maintaining the containment function of the package. The temperatures and design temperature limit criteria for the package components were reviewed and found to be consistent throughout the application.

3.2.5 Summary Tables of Pressures in the Containment System

A discussion of the pressure in the package body under NCT and HAC is presented in Sections 3.1.4 and 3.3.3 of the application, respectively, and is summarized in Table 3-8. These sections were reviewed and found consistent with the pressures presented in the General Information, Structural Evaluation, and Containment Evaluation sections of the application. The maximum package cavity pressure reported was 16.9 psig for NCT and 90.9 psig for HAC. These pressures are below the respective pressures of 30 psig and 120 psig that are considered for the structural evaluation of the package body.

3.3 Material Properties and Component Specifications

3.3.1 Material Properties

The applicant provided material properties in the form of thermal conductivities, densities, and specific heats for the modeled components of the package. The applicant used surface absorptivity values to model solar insolation onto the package and emissivity values to model radiative heat transfer interaction between the environment and the package surface. The staff reviewed the thermal properties used for the analysis of the package and determined that they were appropriate for the materials specified, with the exception of the emissivity value for stainless steel, as discussed in Section 3.2.1 (page 3-17) of the application. The approach used by the applicant for applying solar insolation loads was consistent with NUREG 1617.

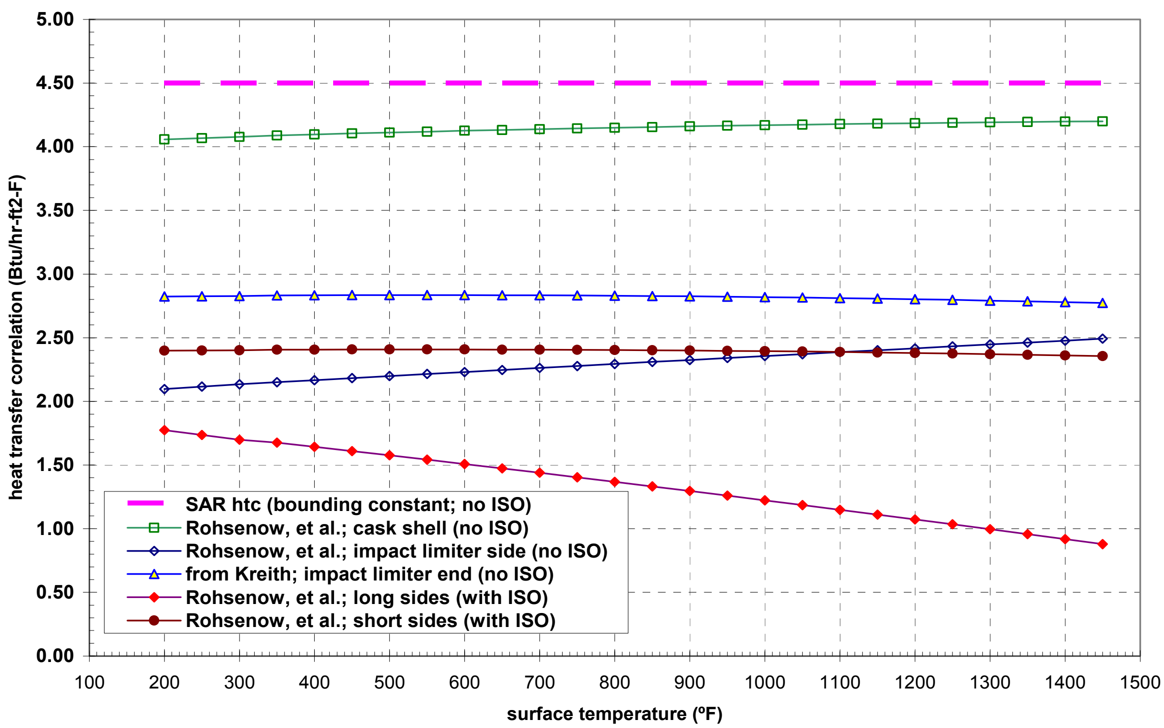
The applicant listed the properties of air in the Section 3.2 of the application and these properties were utilized to analyze the conditions of the package, as required by 10 CFR Part 71 for NCT and HAC.

In Appendix 3.3.1.1 of the application, the thermal properties of air were used to evaluate the heat transfer coefficient for natural convection from the outer surface of the ISO container. Thermal radiation from the ISO container outer surface was added to the convection heat transfer by evaluating the energy equation for radiation heat transfer expressed in terms of a heat transfer coefficient, assuming a view factor of 1.0, from the ISO surface to ambient. In Appendix 3.3.1.3 of the application, the thermal properties of air were used to evaluate the effective conductivity due to conduction and natural convection heat transfer between the inner surface of the ISO container and the outer surfaces of the Model No. TN-LC package, including the impact limiters, at NCT. Thermal radiation between the package surfaces and the ISO container was calculated directly in the ANSYS model, using the surface emissivities described in Appendix 3.3.1.1.

In Appendix 3.4.2 of the application, the thermal convection coefficient on the package outer surface for forced convection heat transfer during the HAC fire was specified at a constant bounding value of 4.5 Btu/hr-ft^2-°F. This value has been verified by NRC staff as bounding for forced convection heat transfer around the Model No. TN-LC package (with or without the ISO container) at the relatively high assumed velocity of 30 ft/sec, as shown in Figure 3-1.

Figure 3-1

Forced convection heat transfer coefficient for HAC fire conditions; assumed 30 ft/sec fire-induced axial velocity along package sides



3.3.2 Technical Specifications of Components

The applicant provided references (in Section 3.5 of the application) for the technical specifications of pre-fabricated package components including O-rings, impact limiters (wood), and neutron absorber materials (polyester resin). All components are rated to perform (at a minimum) in a range of conditions from cold normal conditions with an ambient temperature of -40°F to the hot normal condition of 100°F.

3.3.3 Thermal Design Limits of Package Materials and Components

The staff reviewed and confirmed that the maximum allowable temperatures for each component critical to the proper function of package containment, radiation shielding, and criticality were specified. For HAC fire conditions, the maximum allowable fuel cladding temperature of 1058°F is used by the applicant as a limit for the LWR fuel cladding. This limit is justified and supported by the Pacific Northwest National Laboratory report, PNL-4835, which is a methodology accepted by the NRC staff. The staff is aware that there is not enough data to finalize the limit of 1058°F for high burnup fuel, but still accepts 1058°F as an appropriate criteria given the fact that the "bounding" maximum fuel cladding temperature is calculated to be 694°F, as shown in Table 3-3 (the case without ISO container under HAC) which is far below 1058°F with a significant margin of 364°F."

A limit of 1140°F is used for the aluminum cladding of test reactor fuel (NRUX, MTR, and TRIGA), based on the melting points of aluminum alloys 1100 and 6063. For loading and unloading conditions and NCT, the maximum allowable fuel cladding temperature limit is 752°F for LWR fuel, based on ISG-11, Rev. 3. A limit of 400°F is used for test reactor fuel. This limit is established to ensure integrity of the aluminum cladding that is used for these types of fuels.

3.4 Thermal Evaluation Methods

3.4.1 Evaluation by Analyses

The methods used for the thermal analyses were sufficiently described to permit a complete and independent verification by the staff.

The applicant used the ANSYS® finite element analysis code to perform the thermal evaluation of the package. The applicant assembled several analysis models of the package to determine temperatures that both package components and the fuel would experience during NCT and HAC. The models, described in Section 3.3.1 of the application, are summarized below.

3.4.2 Package Body Models

Several 3 dimensional, half-symmetric, models were developed to determine temperatures of the package for transport with and without an ISO container. The applicant included an air gap of 0.01 inches at the thermal equilibrium condition, between the resin boxes (neutron shielding) and the adjacent shells, to conservatively represent the surface-to-surface contact between these components. Radiation heat transfer across the gaps is neglected in the model. All gaps used in the model were specified in Section 3.3.1.1 of the application. Redwood and balsa within the impact limiters are modeled as an isotropic material with material properties as described in Section 3.2 of the application. Figure 3.2 shows the package body finite element model, with and without the ISO container.

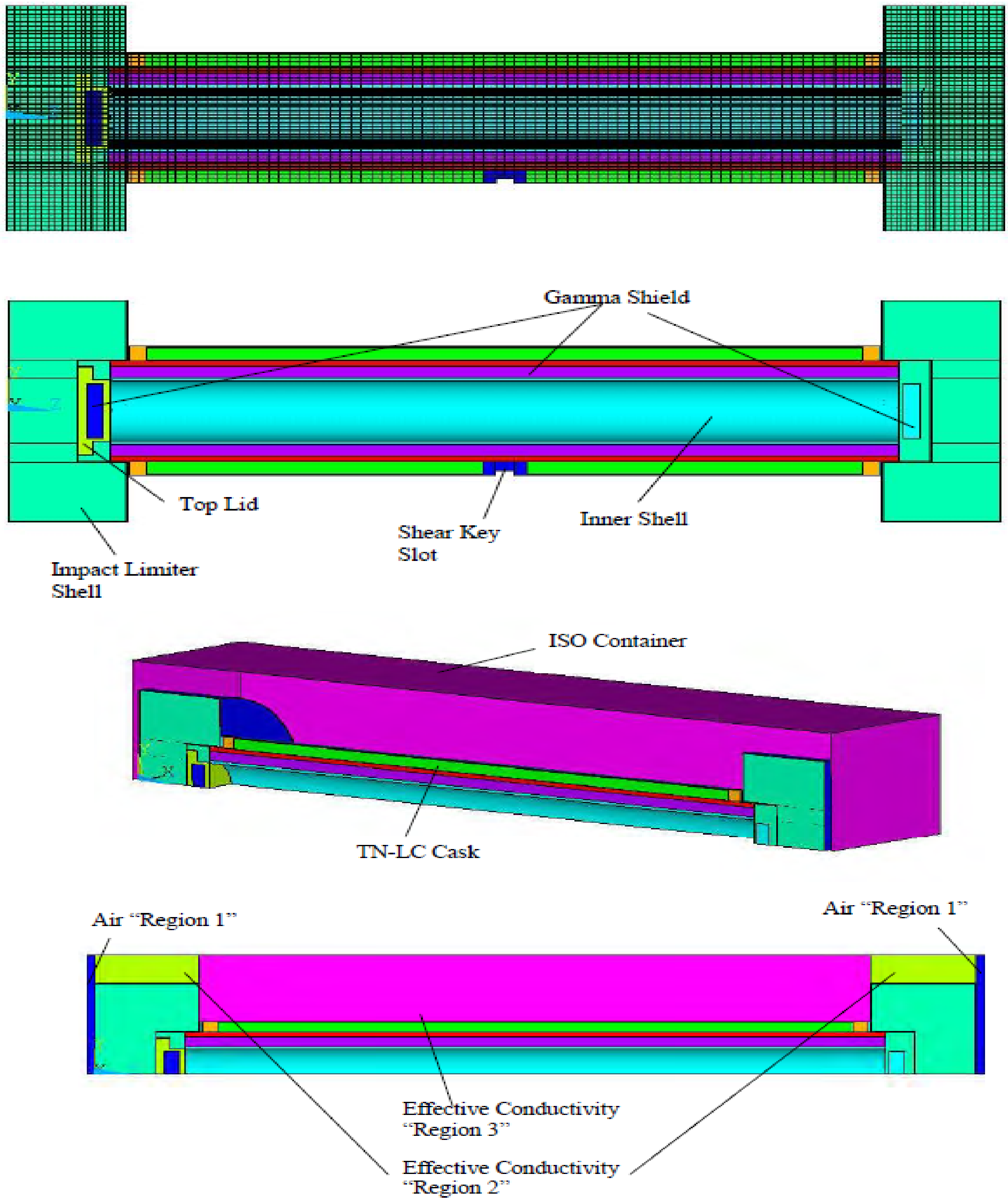


Figure 3.2.

ANSYS model diagrams of TN-LC package (from Figures 3-1 and 3-2 of the application)

The model that includes the ISO container simply adds the ISO box and the air between the package outer surface and the ISO container interior surfaces. Radiation is also modeled between these surfaces using the ANSYS MATRIX50 super-element generated using the AUX12 processor. In both of these models, the decay heat load from the contents was treated as a uniform flux boundary condition on the package inner shell; therefore, the contents were not explicitly modeled. The decay heat load for a given basket configuration was modeled at or above the maximum allowed decay heat for that basket design.

3.4.3 Fuel Basket Models

To analyze the thermal performance of the fuel baskets for the required hot and cold NCT cases, the applicant developed half-symmetric 2-D finite element models of the fuel baskets, representing the fuel region and interior basket components up to the package inner shell wall. The package inner shell wall was treated as a constant temperature boundary, specified at the peak temperature calculated with the detailed 3-D package body model with the ISO container. The 2D “slice” is through the axial location of the maximum peaking factor in the fuel in order to capture the hottest point in the axial distribution of decay heat in the basket.

The decay heat loads applied to the various basket models are listed in Table 3-14 of the application and reproduced in Table 3-1 below for reference. The values specified under the heading “Heat Load Considered in the Package Model” are the values of the heat load applied to the 3-D package body model, which are in some cases higher than the maximum decay heat load permitted in the particular basket design. The values specified under the heading “Total Heat Load Considered in Basket Model” were used in calculations with the 2-D basket models, and correspond to the maximum decay heat load for the basket design. However, the peak inner shell temperatures, calculated using the 3-D package body model at the higher assumed decay heat load, were used in these cases to determine the bounding peak temperatures in the basket for transport.

Table 3-1:TN-LC Basket Heat Load Summary

Basket Type	**No. of Elements/ Assemblies in Basket**	**Heat Load per Element/ Assembly (W)**	**Peaking Factor**	**Total Heat Load Considered in Basket Model (2-D) (kW)**	**Heat Load Considered in Package Model (3-D) (kW)**
TN-LC-NRUX	26 NRU/NRX Assemblies	15/Assembly	1.0	0.390	0.5
TN-LC-MTR	54 Elements	30/Element	1.0	1.62	1.85
TN-LC-TRIGA	180 Elements	8.33/Element	1.0	1.5	1.5
TN-LC-1FA (BWR)	1 BWR Assembly	2000/Assembly	1.2	2.0	2.0
TN-LC-1FA (PWR)	1 PWR Assembly	3000/Assembly	1.1	3.0	3.0
TN-LC-1FA (Pin-Can)	9 Fuel Pins(1)	220/Pin	1.1	2.86	3.0
	25 Fuel Pins	120/Pin	1.1	3.0	3.0

Notes:
1. The thermal model considers 13 fuel pins with heat load of 220 W per pin, which is bounding for the 9 fuel pin configuration.

The heat load for each fuel element or bundle is applied as a uniform volumetric heat generation within a homogenized fuel region. Since the ANSYS models are 2-D, the applicant considered only radial heat transfer from the fuel basket. The bounding effective thermal conductivities calculated for the homogenized fuel regions inside the fuel baskets are presented in Section 3.2.1 of the application. The effective conductivities for the fuel regions include radiation and conduction, but neglect convection heat transfer effects. Effective conductivities for basket structural components and gaps are calculated using the approach described in Section 3.3.1.5 of the application. Calculated effective conductivities for basket components account for conduction only. Heat transfer due to convection is not considered. Neglecting axial conduction and the exclusion of convection are considered modest conservatisms in these models.

The basket components, including the gaseous gaps, were modeled using PLANE55 elements. Radiation between the adjacent surfaces is modeled using the radiation super-element processor (AUX12). LINK32 elements are used in modeling radiating surfaces to create the radiation super-element. The LINK32 elements were unselected prior to the solution of the model. Figure 3.3 shows fuel basket finite element models.

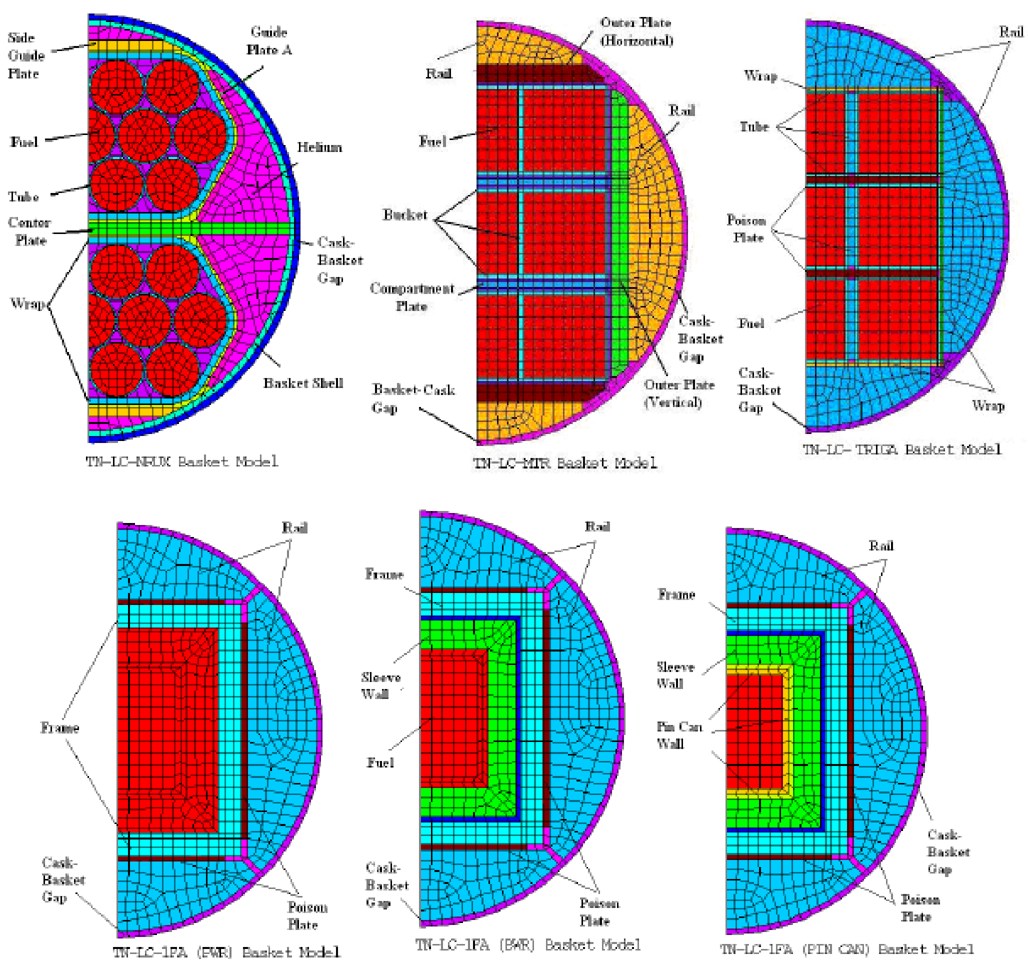


Figure 3.3
Diagrams of ANSYS models of TN-LC baskets, (from SAR Figure 3-11)

3.4.4 Transport Package Model

In order to model the performance of the Model No. TN-LC package with its payload, the heat load for the various baskets is simulated as a heat flux distributed uniformly over the active fuel length of each fuel assembly on the inner surface of the inner shell of the package. The active lengths used for the PWR, MTR, and NRU/NRX fuel assemblies are 144 in., 176, in., and 121 in., respectively. For the PWR assembly and the NRU/NRX fuel assemblies, the active length is significantly shorter than the basket length of 181 in., and the applicant claims this represents an added conservatism in the model. The staff believes that, for the small radial diameter of this package, this assumption is a reasonable engineering approximation, rather than conservatism. Convective heat transfer within the package cavity is not accounted for, and this is considered a nominal conservatism, as convection provides limited additional heat transfer when a package in the horizontal orientation.

3.4.4.1 Transport Package without ISO Container

Without the ISO container, the Model No. TN-LC package is transported in the horizontal position. Therefore, the applicant assumes that the lower half of the package is not exposed to solar insolation and no solar heat flux is considered over these surfaces. The entire surface area of exposed vertical surfaces has the solar heat flux applied.

3.4.4.2 Transport Package in ISO Container

The transport of the Model No. TN-LC package in an ISO container presents a more complex computational regime, for which the applicant includes:

- Exposure of all the surfaces of the ISO container to solar heat flux. This is a conservative assumption.

- Heat dissipation from the package surface by natural convection (inside the ISO container cavity) and by radiation between surfaces. The radiation heat exchange between the surfaces is modeled using the MATRIX50 super element generated by the AUX12 processor.

In order to model natural convection in the ISO container cavity, the entire length of the cavity is divided into three regions.

- “Region 1” includes the space between the ISO container and the ends of the TN-LC impact limiter outer surfaces. The effects of natural convection are ignored for this region and only gaseous conduction due to the presence of air is considered.

- “Region 2” includes the area between the radial outer surface of the impact limiter and the ISO container. Natural convection is implemented using an effective conductivity calculated using empirical correlations for heat transfer across the gap between two horizontal cylinders. The calculation is shown in Section 3.3.1.3 of the SAR.

- “Region 3” includes the area around the neutron shield between the impact limiter inner surfaces and the ISO container. Natural convection is implemented via the same correlation as in Region 2.

The regions described above are shown in Figure 3-2 of the application and reproduced below for reference. The material properties used in the transport package model are listed in Section 3.2 of the application.

3.4.5 Thermal Analysis Results

For the normal operating conditions, the applicant performed a steady state evaluation of the entire package, using both the package body model and multiple fuel basket models. These analyses produced a maximum fuel cladding temperature of 266°F for TRIGA fuel, which is below the limit of 400°F chosen by the applicant for research reactor fuel, and a maximum fuel clad temperature of 542°F for LWR fuel, which is below the limit of 752°F specified in ISG-11, Rev. 3. The maximum seal temperature (for all seals) under normal conditions is 205°F, which is below the seal material operating limit of 400°F.

The applicant utilized the package body model for the hypothetical accident condition analysis, but added a homogenized region representing the fuel basket, fuel region, and basket-to-inner shell gap (discussed in Section 3.6 of this SER) to properly assess the effects of transient conditions on the package. The maximum temperature predicted for the package inner shell in the transient analysis was used as a boundary condition for steady-state analyses of the 2-D model of the pincan basket with 25 PWR fuel rods. These analyses produced a maximum fuel cladding temperature of 694°F in the pincan basket, which is below the limit of 1058°F. Under these conditions, the maximum seal temperature was shown to be 449°F (top lid seal). This seal temperature for the 30-minute fire accident is below the excursion limit of 482°F cited in the SAR for the seal material. The compound chosen for the top lid inner seal is FKM type V1289-75 (as specified in note of Ref. 1 listed in the 2nd Request for Additional Information (RAI) response) which can last 70 hrs at 482°F (250°C), thus exceeding the temperature excursion observed in the test for the top lid seal in both temperature and in duration, and did not result in a significant change in the seal hardness (increase of 1%). Therefore, the top lid seal is not affected by a one-hour exposure to temperature above 400°F, with a peak temperature of about 449°F (232°C).

For the bounding conditions representing the maximum fuel cladding temperatures when the TN-LC package is loaded inside an ISO container, the maximum seal temperatures during HAC are 285°F (141°C) in the confirmatory analysis discussed in SAR Section 3.6.8 and 307°F (153°C) in the alternate confirmatory analysis discussed in SAR Section 3.6.10. These maximum seal temperatures remain well below the normal temperature limit of 400°F (204°C).

3.4.6 Confirmatory Analysis

The staff, along with Pacific Northwest National Laboratory (PNNL), reviewed the finite element models that were submitted by the applicant. The review of the applicant's models revealed two issues which were subsequently communicated to the applicant as a request for additional information. The findings by the staff and the resolution of these issues are discussed below.

3.4.7 Limiting Cases for Package Analysis

Based on confirmatory analyses done by the staff, the limiting configurations for the Model No. TN-LC package under NCT and HAC were incorrectly identified in the original application. Section 3.3.1.4 of the application describes the limiting payload for the package as the TN-LC-1FA fuel pin basket. This basket is also assumed to be limiting for HAC, and therefore, the subsequent analyses completed for the limiting configuration under HAC were incorrect. In addition, Section 3.4 of the application indicated that the limiting configuration for the Model No. TN-LC package under HAC was without the ISO container. Peak fuel cladding temperatures for packages under HAC have generally been shown to occur with the package in the ISO container, due to the thermal inertia of the package in the fire transient, as demonstrated in NUREG/CR-6894, Rev. 1. While the temperatures presented in the applicant's analysis are generally conservative, the staff believes that the use of a "de-coupled" analysis approach for this package may have led to the misidentification of the limiting configurations. In response to an RAI from the staff, the applicant corrected this problem in a revision to their application by developing a coupled analysis of the Model No. TN-LC package, which demonstrated that the limiting contents for the package (for the thermal analysis) was the PWR fuel assembly, and the limiting configuration for the HAC analysis was the package in the ISO container. Further details on this analysis are provided below.

3.4.8 Confirmatory Analysis with the Coupled Model

The applicant developed a coupled model of the Model No. TN-LC package and the ISO container in order to demonstrate that the original de-coupled approach was conservative. The applicant described the coupled modeling approach for a confirmatory analysis in Section 3.6.8 and the coupled modeling approach for an alternate analysis in Section 3.6.10 of the revised application.

The coupled models were created by extruding the 2-D thermal model of the TN-LC-1FA basket (described in Section 3.3.1.4 of the applicant's application) and introducing the elements and nodes from the extruded basket model into the LC package model with the ISO container. Mesh continuity is assured by matching the radial and axial node locations between the package inner cavity wall and the basket. For the evaluations without the ISO container, the elements representing the ISO container and the air within the ISO container cavity are deleted.

The coupled model of the Model No. TN-LC package with the 1FA basket explicitly models the package and basket components including poison plates, aluminum transition rails, and helium back fill gas, and considers a homogenized fuel assembly in the fuel compartment. For the fuel pin contents, the BWR sleeve along with the pin can side wall is explicitly modeled. The 25 fuel pins and the stainless steel tubes that hold them are represented as a homogenized region with effective properties. The effective properties for the PWR assembly and the 25 pin fuel can are provided in Appendices 3.6.6 and 3.6.8.2 of the application, respectively.

The decay heat load for the contents are applied as heat generation boundary conditions on the elements representing the fuel assembly or 25 fuel pin can. The decay heat profile is modeled by using peaking factors applied along the axial length of the fuel as described in Appendix 3.6.8.1 of the applicant's application.

The coupled models are shown in Figures 3.4 and 3.5 below.

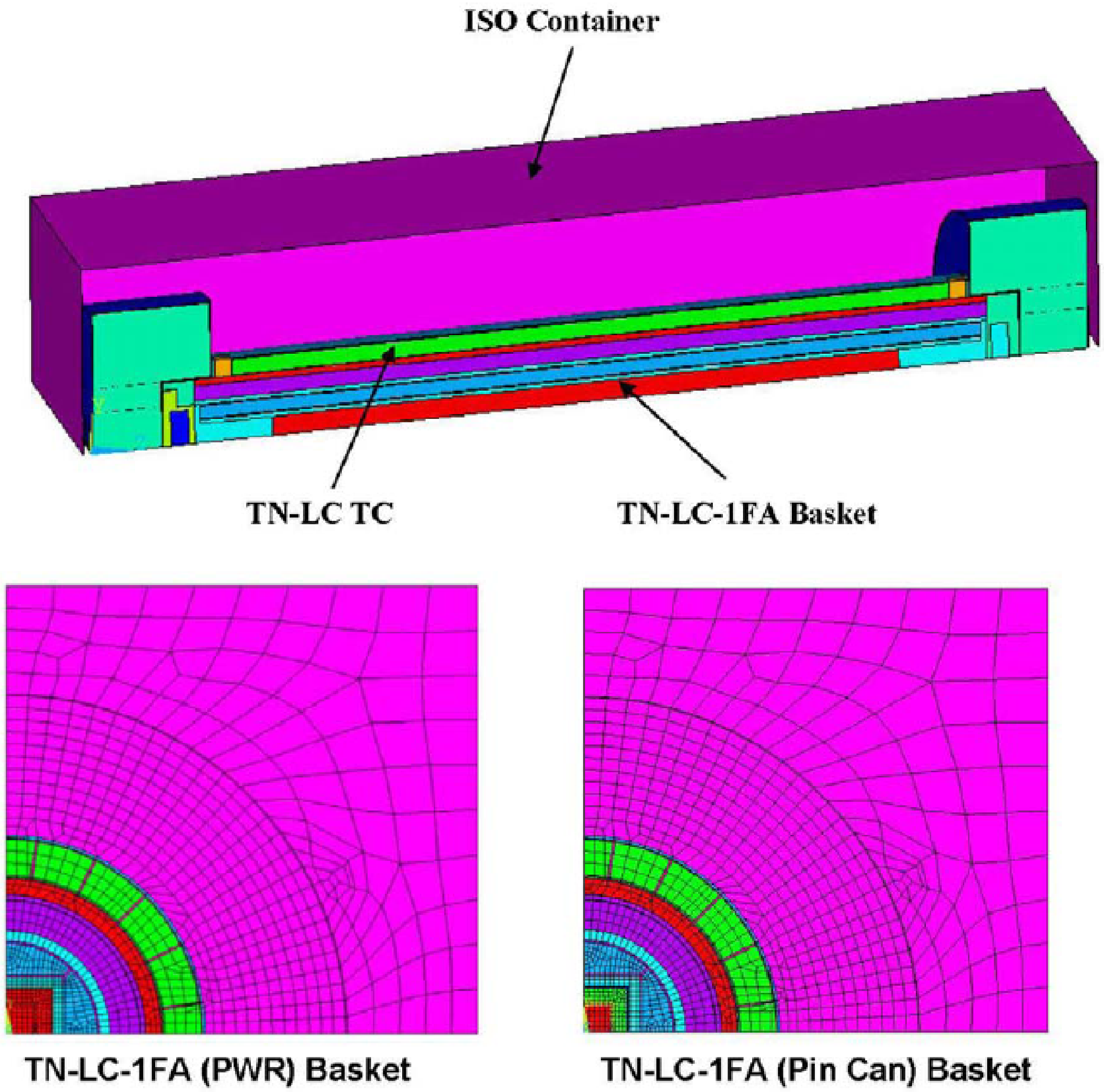
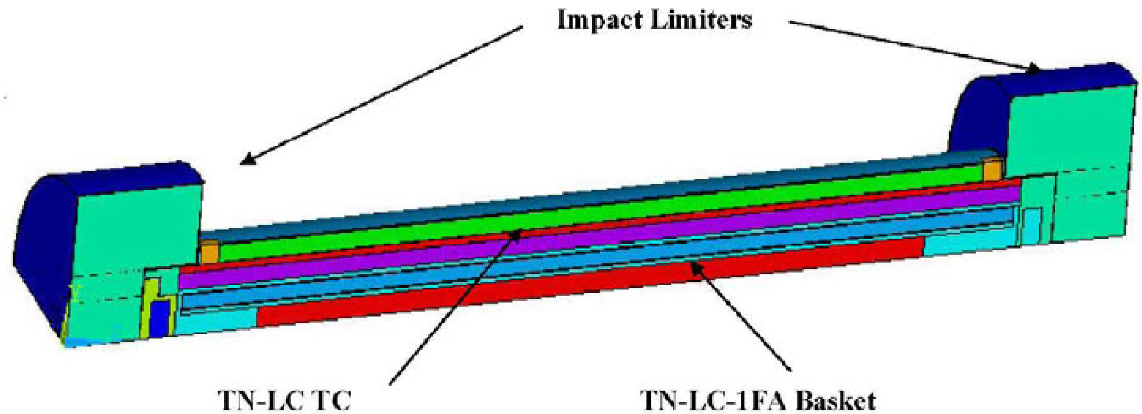


Figure 3.4. TN-LC TC Coupled Model with ISO Container, (from SAR Figure 3-28)



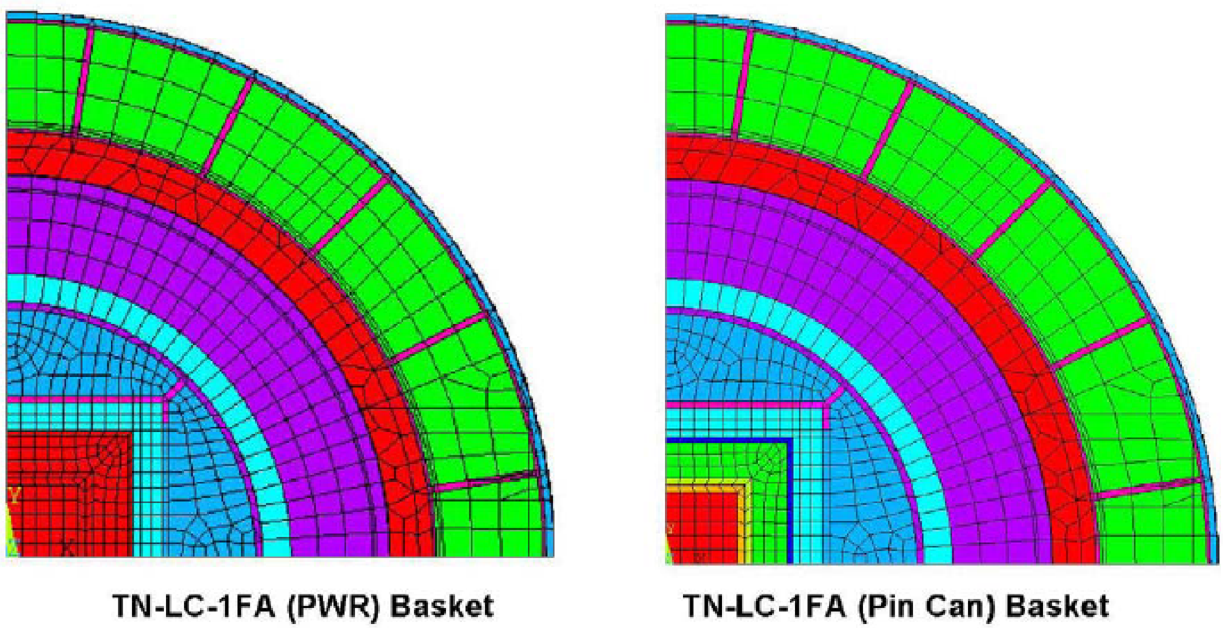


Figure 3.5. TN-LC TC Coupled Model without ISO Container, (from SAR Figure 3-39)

3.4.8.1 NCT Results

The applicant determined that the limiting case for NCT is the package with a PWR fuel assembly inside the ISO container. This is in contrast to the applicant's original finding that temperatures for the 25 pin can was limiting for the package under NCT conditions.

The maximum temperature calculated for the fuel cladding using the coupled model, 522°F, is less than the maximum calculated for the 25 pin can using the de-coupled approach (543°F). The seal temperatures, however, are higher with the coupled model, but still well within the limits of the seal material. The results of the coupled modeling approach demonstrate that this approach is more accurate in determining the actual behavior of the system under NCT, and it is recommended that this approach be used by the applicant as the analysis of record, if possible, for future submittals.

3.4.8.2 HAC Results

In order to apply the HAC fire to the package in the ISO container, the applicant used an average temperature, between the prescribed fire temperature (1475°F) and the maximum package surface temperature (225°F) calculated for NCT conditions, of 850°F for the environment temperature within the ISO container for the duration of the HAC fire exposure (30 minutes). Although the applicant claims this is conservative, it is not a true "coupled" approach and was not demonstrated to be conservative vis-à-vis the traditional approach of applying an environmental temperature to the surface of the ISO container and determining the temperatures of the inboard components of the package based on that exposure. Based on the applicant's approach, it was determined that the limiting configuration for HAC is the package with PWR fuel assemblies inside the ISO container.

3.4.8.3 Coupled Model Review Conclusions

The results of the coupled model simulations are summarized below.

Table 3.2 Analysis Results for Coupled Modeling Approach

Comp.	NCT				HAC			
	With ISO Container		Without ISO Container		With ISO Container		Without ISO Container	
	PWR Assembly	Pin Can	PWR Assembly	Pin Can	PWR Assembly	Pin Can	PWR Assembly	Pin Can
Fuel Cladding	522°F	512°F	508°F	495°F	575°F	570°F	583°F	(1)
Seals	209°F	210°F	188°F	188°F	285°F	284°F	451°F	(1)(2)

1. Bounded by the maximum fuel cladding and seal temperatures for PWR fuel assemblies without ISO container.
2. The calculated seal temperature of 451°F is below the excursion limit of the compound (FKM type V1289-75) chosen for the top lid seal which can last 70 hrs at 482°F.

The staff reviewed the applicant's revised analysis and found that the analysis was acceptable.

3.4.9 Alternate Confirmatory Analysis with the Coupled Model

There are two uncertainties in the applicant's confirmatory analysis using the coupled model with ISO container:

(1) A fixed temperature of 850°F was used to provide convective heat input to the cask during the 30-min fire. This fixed temperature introduces uncertainty in heat input into the cask during the 30-minute HAC fire.

(2) The exterior horizontal surface of the ISO container represents both top and bottom outer surfaces of the ISO container. However, the heat transfer coefficients from a horizontal heated plane facing upwards are applied to account for heat transfer to the ambient, which over-predicts the heat transfer from the bottom surface of the ISO container.

To avoid uncertainty in heat input into the package and heat transfer coefficients from top and bottom plates of the ISO, staff issued an RAI, and the applicant performed an alternate confirmatory thermal analysis for NCT and HAC fire for the package within the ISO container. The applicant used the following approach: (i) the effective thermal conductivity (k_{eff}) of the air enclosed within the ISO container is calculated based on the methodology described in Section 3.3.1.3 of the application, (ii) the average convection heat transfer coefficients, based on flat surfaces facing upward and downward, are used to apply the convection boundary condition at the horizontal surface of the ISO container.

In addition to the above changes, the thermal mass of the neutron shield resin is retained during post-fire conditions and only the thermal conductivity is changed to air to maintain the total heat capacity of the system throughout the HAC evaluation. All other assumptions and design inputs remain identical to those in the coupled analysis presented in Section 3.6.8 of the application. The maximum temperatures for the fuel cladding and the seals for NCT and HAC resulting from the alternate confirmatory thermal analysis are summarized below.

3.4.9.1 NCT Results

For NCT, the maximum seal temperature of 214°F for the alternate confirmatory thermal analysis exceeds the maximum seal temperature of 205°F determined using the decoupled model listed in Table 3-3. However, considering the large margin to the long-term seal temperature limit of 400°F, the performance of the seal and containment of the package during NCT is not affected.

3.4.9.2 HAC Results

For HAC, the highest short-term seal peak temperature for the package in the ISO container is 307°F, which is bounded by the short-term peak seal temperature of 449°F for the decoupled analysis model. Based on results from both of the confirmatory thermal analyses and the alternate confirmatory analysis, the thermal evaluations based on the decoupled model reported in Sections 3.3 and 3.4 of the application remain bounding. All design criteria for the package, specified in Section 3.1 of the application, are herein satisfied.

The alternate confirmatory analysis results in slightly higher maximum temperatures for the fuel cladding and the seals for NCT compared to the coupled model presented in Section 3.6.8. The maximum temperatures of fuel cladding and seals of the package are 307°F and 595°F for HAC. The maximum fuel cladding temperatures for NCT and HAC remain bounded by the maximum values reported in Tables 3-6 and 3-7 of the application, respectively. The highest short-term seal peak temperature is 307°F under HAC, which is below the seal temperature limit of 400°F. The results of the revised analysis are discussed in Section 3.4.10 of the application.

3.4.9.3 Coupled Model Review Conclusions

The seal temperatures of 449°F at 1.1 hrs, 285°F at 5.4 hrs, and 307°F at 7.8 hrs were calculated using the decoupled model, the coupled model of confirmatory analysis, and the coupled model of alternate confirmatory analysis, respectively, and are bounded by the compound (FKM type V1289-75) chosen for the top lid seal which can last 70 hrs at 482°F.

The staff also agrees that the seal temperatures are bounded under HAC fire for fragmented or reconfigured HBU fuels, in view of the limited effect of the local heat concentration by the HBU fuel rubbles (fragmented or reconfigured) when the top lid seal can last 70 hrs at 482°F and the rubbles are most likely accumulated at the cavity bottom in which the heat concentration is away from the top lid seal.

Table 3.3 Maximum temperatures for fuel cladding and all seals for NCT and HAC.

Loading Condition	*Decoupled Model (Tables 3-1, 3-2, 3-6 and 3-7)*		*Confirmatory Analysis Coupled Model (Section 3.6.8)*		*Alternate Confirmatory Analysis Coupled Model*	
			NCT	*HAC*	*NCT*	*HAC*
	NCT	*HAC*	*N1*	*H1*	*N1A*	*H1A1*
	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)
Fuel Cladding	*543*	*694*	*522*	*575 @ 8.4 hr*	*525*	*595 @ 9.8 hr*
Seals	*205*	*449 @ 1.1 hr*	*209*	*285 @ 5.4 hr*	*214*	*307 @ 7.8 hr*

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

3.4.10 Evaluation of Accessible Surface Temperature

The accessible surfaces of the package include the package outer shell and the vertical and radial surfaces of the impact limiters. When transported with an ISO container, the only accessible surface is the outer surface of the ISO container. The applicant analyzed the ISO container surfaces under normal conditions in shade and determined that the surfaces would not exceed 150°F, while the package outer shell would not exceed 170°F. Because all accessible surfaces of the package, as prepared for transport, will remain below the regulatory limit of 185°F, the staff finds this acceptable for exclusive use shipments without the use of a personnel barrier.

3.5 Thermal Evaluation under NCT

The discussion below constitutes a review of the applicant's original submittal which included the de-coupled modeling approach. While the applicant performed a coupled model in response to staff RAI's, the de-coupled approach remains the applicant's analysis of record, and, as a result was reviewed by the staff.

3.5.1 Heat

The applicant performed steady-state calculations for an ambient temperature of 100°F with solar insolation and a maximum decay heat of 3kW utilizing the models described in Section 3.4 of this SER. The applicant calculated the maximum component temperatures and determined that the limiting component temperatures were derived for a contents heat load of 3kW. The temperatures for the package are reported in Tables 3-9 and 3-10 of the application.

The applicant analyzed baskets for both commercial and research reactor fuel using the heat loads listed in Table 3-1. The applicant applied 0.01 inch gaps between all adjacent components in the basket models. The applicant claims that based on the TN MP197 SAR (Accession No. ML120A216), Section A.3.6.7.4, the assumed gap size of 0.01 inch is approximately two times larger than the contact resistances between the adjacent components and is therefore conservative. The staff agrees with the applicant's assessment.

The heat loads applied for the thermal analysis of the TN-LC fuel baskets are listed in Table 3-1 of this SER. In some cases, higher decay heat values are applied to the (3D) package model, as described in Section 3.4.3 of this SER (also listed in Section A.3.1.2 of the application. The applicant did not employ axial heat transfer in any of the baskets to provide additional conservatism in calculation of the maximum fuel cladding and maximum component temperatures.

The package O-ring seals are not explicitly considered in the models. The maximum seal temperatures are retrieved from the models by selecting the nodes at the locations of the corresponding seal O-rings. Since the seals are between adjacent metal components of relatively high thermal conductivity, this yields an acceptable engineering approximation of the seal material temperatures. The maximum seal temperature of 205°F (96°C) for NCT is below the long-term (continuous use) limit of 400°F (204°C) specified for continued seal function.

The staff reviewed the applicant's models, the procedures used to analyze normal conditions of transport, and the procedures used to apply the decay heat to the fuel regions and found them to be acceptable.

3.5.2 Cold

The applicant evaluated the minimum ambient temperature of -40°F (-40°C), and determined, as expected, that the resulting packaging component temperatures will approach -40°F if no credit is taken for the decay heat load. Since the package materials, including containment structures and the seals, continue to function at this temperature, the minimum temperature condition has no adverse effect on the performance of the package. The temperature results from the applicant's analyses are presented in Tables 3-2 thru 3-5 of the application.

The staff reviewed the information provided by the applicant and agrees with the above conclusion concerning the response of package materials to the cold condition.

3.5.3 Maximum Normal Operating Pressure (MNOP)

The applicant calculates the MNOP within the package body for NCT in Section 3.3.3 of the application, accounting for the package cavity free volume, the quantities of backfill gas, fuel rod fill gas, fission products, and the average package cavity gas temperature. The ideal gas law is then employed to determine the internal package cavity pressure.

The calculation assumes the maximum allowable heat load of 3 kW and a maximum burnup of 70,000 MWD/MTU. For these calculations, a B&W 15x15 assembly is considered the limiting fuel assembly, since the physical dimensions of this fuel design result in the smallest free volume within the package.

The maximum pressure reported for normal conditions was 16.9 psig within the package body. The design basis pressure for the package body is 4.4 atm (50 psig), and the normal design pressure is 1.68 atm (10 psig). The MNOP calculated for the package is within the limits set by the applicant.

3.5.4 Evaluation of Loading/Unloading Operations

The applicant described their analyses of the loading and unloading operations in Section 3.3.4. of the application. Loading of PWR/BWR assemblies take place in a pool with the TN-LC in a vertical orientation. Vacuum drying is considered a standard operation for wet loading operations. Loading for PWR/BWR fuel pins and research reactor fuel elements occur in a dry environment with the TN-LC in a vertical or horizontal position.

3.5.4.1 Wet Loading/Unloading

After loading a fuel assembly, the package is removed from the pool and drained, dried, sealed and backfilled with helium. Draining and vacuum drying is done under the cover of helium gas. The helium environment maintained in the package provides for heat removal from the fuel, and eliminates thermal cycling of the fuel during this operation, meeting the limits for short term operations set in ISG-11. While vacuum drying is considered the bounding operation for wet loading, the maximum basket and fuel cladding temperatures are bounded by those calculated by the applicant for NCT described in Section 3.3 of the application.

The bounding operation for wet unloading is considered the re-flood of the package fuel compartment. The fuel compartment is filled with pool water through its drain port while the vent port is opened, with effluents routed to the loading facility off-gas monitoring system. This also limits the build-up of pressure in the fuel compartment. Fuel cladding temperatures are calculated to be significantly less during the re-flood than they are for vacuum drying during package loading, therefore, fuel cladding temperatures for re-flood are bounded by the vacuum drying operation.

3.5.4.2 Dry Loading/Unloading

The applicant used the TN-LC-1FA fuel pin basket and the TN-LC-MTR basket models (described in Section 3.3.1.4 of the application) to determine the maximum bounding temperatures for baskets and fuel cladding during dry loading/unloading operations.

Because the applicant uses a de-coupled modeling approach, the applicant applied the TN-LC fuel compartment inner surface temperatures from the TN-LC model for NCT without the ISO container (stating that this model is bounding for dry loading/unloading conditions) to the surfaces of the baskets mentioned above. The applicant conservatively chose inner shell temperatures of 210°F for the 3.0 kW pin can basket and 170°F for the 1.85 kW MTR basket. The applicant calculated new effective fuel conductivities given that dry loading/unloading would take place in an air environment. The calculations of the new conductivities used the same methodology described in detail in Appendix 3.6.6 of the application. The results of the fuel conductivity calculations are presented in Section 3.3.4.2 and the resulting temperatures are summarized in Table 3-19 and compared to NCT temperatures in Table 3-20 of the application. Fuel cladding temperatures increase approximately 58°F for the MTR fuel cladding and 177°F for the pin-can basket over the values for NCT. These values are still within the limits prescribed for the fuel cladding being considered (400°F for MTR and 752°F for BWR/PWR fuel cladding).

3.5.5 Maximum Thermal Stresses

The applicant reports maximum thermal stresses for NCT in Section A.2.6.1.3 of the application. All thermal stresses are below the allowable stresses for critical package components.

3.6 Thermal Evaluation under Hypothetical Accident Conditions

3.6.1 Initial Conditions

The applicant performed a transient thermal analysis to evaluate the package under HAC. The pre-fire condition was a 100°F ambient with full solar insolation, as well as radiation and convection from the surface of the package, based on normal conditions, with the payload of a PWR fuel assembly and a heat load of 3 kW.

The applicant modified the ANSYS® model developed for NCT for the accident condition analysis, which represented the package outside of the ISO container. The HAC model is described below.

3.6.2 HAC Model Assumptions

The NCT three dimensional, half symmetry model of the package cross section was modified by the applicant to include the following conditions:

- Gaps in the package are essentially closed to allow for maximum heat transfer into the package. The gaps are restored following the fire. The gaps used are listed in Table 3-21 of the application.
- The resin used for the neutron shield remains intact during the fire and is substituted with air material properties (conduction only) at the end of the fire.
- Impact limiter damage is modeled based on the crush depths given in Chapter 2 of the application (Section 2.13.12.4.4, Table 2.13.12-7) and are:
  - Minimum axial thickness after HAC drop of 7.0 inches
  - Minimum radial thickness after HAC drop of 5.5 inches.
- Welds of the impact limiter shell are intact; therefore, the wood within the impact limiter would not be exposed to air and would char, but not burn, during the HAC fire. However, the applicant examined the behavior of wood exposed to the HAC fire, as described below.

3.6.3 Impact Limiter Wood Behavior in HAC Fire

While an unlikely scenario, the applicant analyzed the potential for exposure of a portion of the wood in the impact limiter to the HAC fire. As described in Section 3.4.1 of the application, the applicant calculated the rate of charring of the wood that would be expected, if the hypothetical puncture condition (based on 10 CFR 71.73) resulted in the outer steel skin covering the flat end of the top impact limiter being torn off. To represent the thermal effect of charring of wood, the applicant conservatively applied a temperature of 900°F to the inner surface of the impact limiter inner cover for 30 minutes immediately following the fire. This is shown in Figure 3-19 of the application and reproduced in Figure 3.6 below for clarity.

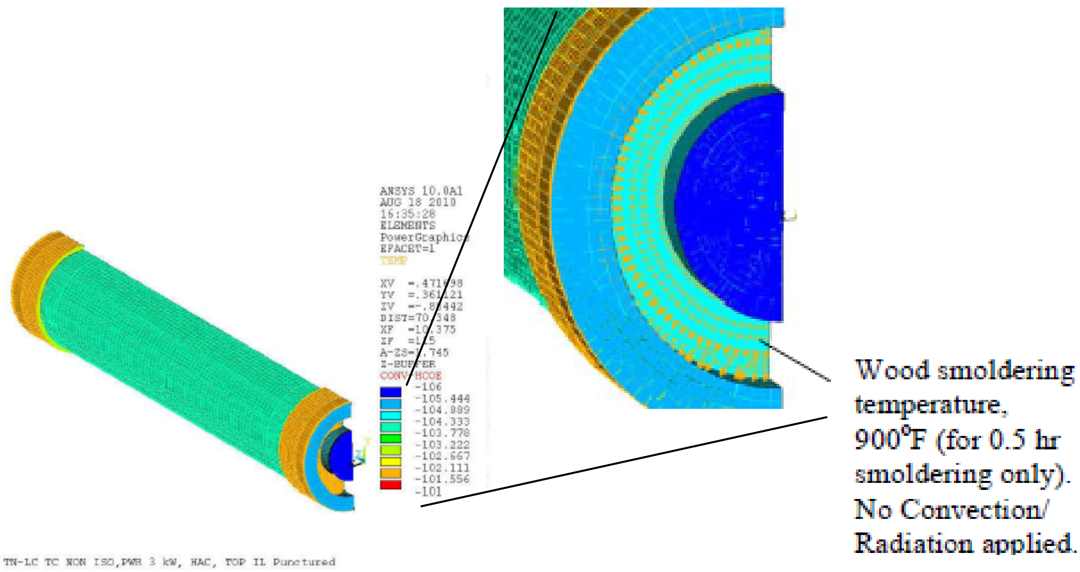


Figure 3.6 Wood Smoldering Conditions During Cool Down Conditions

The density of wood has a large effect on the char rate. However, the char rate of wood (balsa and/or red wood) is not directly used in the thermal model. The applicant applied a constant temperature of 900°F, representing the charring temperature, at the inner surface of the impact limiter where the hypothetical charring of wood occurs.

The applicant calculated the duration needed for charring and the calculation shows the impact limiter inner surface is not exposed to charring wood temperature for approximate 2 hours (t_1) after the end of the 30-minute fire and after that, the impact limiter inner surface is exposed to a charring wood temperature of 900°F for 14.2 minutes (t_2) when a char rate of 55 mm/hr was used for calculation. Using a char rate of 90 mm/hr, the charring times t_1 and t_2 are changed to 1.3 hours and 9 minutes, respectively. After t_1 and t_2 at the end of the 30-minute fire, the wood is completely charred and the impact limiter is exposed to the ambient. Based on the USDA Wood Handbook and Reference 1 listed below, Balsa wood with the charring rate of ranging from 55 mm/hr to 90 mm/hr is appropriate for the impact limiter.

The charring wood temperature of 900°F is applied immediately after the end of the fire and this application envelops any uncertainties during the period of time t_1 computed for the impact limiter inner surface to sense the heat input from the charring wood. The charring wood temperature is applied for a longer duration of 30 minutes instead of 14.2 minutes for the char rate of 55 mm/hr or 9.0 minutes for the char rate of 90 mm/hr, computed for time t_2. Per analysis, an increase in the char rate would decrease this time even further and increases the conservatism in the thermal evaluation.

Reference 1 - A. Gilka-Bötzow, A. Heiduschke, P. Haller, “The Velocity of Combustion in Relation to the Density of Wood,” European Journal of Wood and Wood Products, February 2011, Volume 69, Issue 1, pp 159-162.

3.6.4 Fire Test

For the fire accident, the applicant subjected analysis models for the Model No. TN-LC package (depicted in Figures 3-17 thru 3-19 of the SAR) to an ambient temperature of 1475°F for 30 minutes, as defined by 10 CFR 71.73. A convective coefficient of 4.5 BTU/hr-ft^2-°F is used to model the turbulent nature of the fire environment. The staff finds this value acceptable. The package surfaces were given an emissivity of 0.8 for fire exposure and an emissivity of 0.9 during the cool-down period.

The applicant removed the gaps between the materials for the fire transient, to maximize heat input into each of the models. Following the fire transient, the model gaps were then returned to normal condition for the duration of the post fire transient.

3.6.5 Maximum Temperatures and Pressure

The maximum temperatures calculated by the applicant are given in Table 3-4 below and are highlighted in Tables 3-6 and 3-7 of the application. As before, the 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.

The applicant calculated the maximum internal operating pressure, considering only the helium fill gas. The average gas temperature in the package cavity was calculated to be 581°F (305°C) and the maximum internal operating pressure was determined to be 90.9 psig (6.18 atm) for the package body. Therefore, the values calculated by the applicant are below the pressure limits specified by the applicant.

The staff reviewed the calculations submitted by the applicant and found them to be adequate. Therefore, the staff agrees that the package and canister meet the design requirements for maximum pressures.

<table>
<tr><th colspan="8">Table 3-4 Maximum Calculated Temperatures (°F) for PWR Fuel 3 kW Heat Loading</th></tr>
<tr><th colspan="2" rowspan="2">Location or Package Component</th><th colspan="2">Normal Conditions</th><th colspan="2">Accident Conditions</th><th colspan="2">Maximum Allowable</th></tr>
<tr><th>Non-ISO</th><th>ISO</th><th>Non-ISO</th><th>ISO[1)]</th><th>NCT</th><th>HAC</th></tr>
<tr><td colspan="2">Inner shell</td><td>204</td><td>238</td><td>445</td><td></td><td>--</td><td>--</td></tr>
<tr><td colspan="2">Lead Gamma Shield</td><td>203</td><td>236</td><td>558</td><td></td><td>621</td><td>621</td></tr>
<tr><td colspan="2">Outer Shell</td><td>192</td><td>225</td><td>809</td><td></td><td>--</td><td>--</td></tr>
<tr><td colspan="2">Neutron Shield Boxes</td><td>186</td><td>220</td><td>1200</td><td></td><td>--</td><td>--</td></tr>
<tr><td colspan="2">Neutron Shield Resin</td><td>179</td><td>215</td><td>--</td><td></td><td>320</td><td>--</td></tr>
<tr><td colspan="2">Neutron Shield Shell</td><td>177</td><td>210</td><td>1215</td><td></td><td>--</td><td>--</td></tr>
<tr><td rowspan="3">Package</td><td>Lid</td><td>171</td><td>202</td><td>596</td><td></td><td>--</td><td>--</td></tr>
<tr><td>Bottom Flange</td><td>156</td><td>185</td><td>353</td><td></td><td>--</td><td>--</td></tr>
<tr><td>Top Flange</td><td>173</td><td>205</td><td>624</td><td></td><td>--</td><td>--</td></tr>
<tr><td colspan="2">Wood Impact Limiter</td><td>169</td><td>200</td><td>--</td><td></td><td>320</td><td>--</td></tr>
<tr><td colspan="2">ISO Container</td><td>--</td><td>146</td><td>--</td><td></td><td>--</td><td>--</td></tr>
<tr><td rowspan="6">Seals[2)]</td><td>Bottom Drain</td><td>156</td><td>184</td><td>293</td><td rowspan="6">--</td><td>400</td><td>--</td></tr>
<tr><td>Bottom Plug</td><td>155</td><td>183</td><td>283</td><td>400</td><td>--</td></tr>
<tr><td>Bottom Test</td><td>155</td><td>183</td><td>285</td><td>400</td><td>--</td></tr>
<tr><td>Top Cavity Port</td><td>169</td><td>200</td><td>372</td><td>400</td><td>--</td></tr>
<tr><td>Top Lid</td><td>172</td><td>204</td><td>449[(3)]</td><td>400</td><td>--</td></tr>
<tr><td>Top Test Port</td><td>173</td><td>204</td><td>369</td><td>400</td><td>--</td></tr>
<tr><td colspan="2">PWR Fuel Cladding[2)]</td><td>520</td><td>--</td><td>694</td><td>--</td><td>752</td><td>1058</td></tr>
</table>

1) In the original SAR, the applicant did not evaluate the HAC conditions for the package in the ISO container, believing that the limiting case was the package outside of the ISO container.
2) Temperatures for these components for the applicant's coupled model are provided in Table 3-2.

3) The 449°F temperature is bounded by the compound (FKM type V1289-75) chosen for the top lid seal which can last 70 hrs at 482°F.

3.6.6 Maximum Thermal Stresses

The applicant reports maximum thermal stresses for the hypothetical accident condition fire in Section 2.7.3.2 of the application. All thermal stresses are below the allowable stresses for critical package components.

3.6.7 Thermal Evaluation of Fuel Reconfiguration

The physical configuration of the fuel assembly in the TN-LC basket can potentially be altered during transportation, thereby altering the heat transfer distribution within the package cavity. In order to bound the uncertainties in the physical configuration of the fuel assemblies due to the paucity of the structural properties of commercial high burnup fuel assemblies for NCT and HAC, the applicant performed the thermal calculations of the bounding case with PWR fuel assembly at 3.0 kW heat load in an ISO container to provide defense in depth that the containment is maintained during transportation for NCT and HAC. The applicant modeled the reconfigured fuel as a heat generating region within the fuel compartments with the conductivity of helium.

The following approach is considered to envelope any possible scenario due to the physical reconfiguration of high burnup fuel assemblies during transport:

1) The bounding scenarios with a reconfigured fuel assembly are that the fuel assembly shifts entirely towards the top or the bottom of the package cavity. These scenarios bound the maximum seal temperatures both at the top and the bottom of the package of the package due to changed heat distribution.

2) Because the spacer is located at the bottom of the package to minimize axial movement of the fuel assembly and to maintain the gap between the fuel assembly and the package cavity, and also because the seals located at the top of the package are closer to the cavity than the seals located at the bottom of the package, the thermal evaluation of the reconfigured fuel accumulation towards the bottom of the package is bounded by that towards the top of the package.

3) It is considered that the heat generating elements of the reconfigured fuel assemblies coalesce within 75% of the original active fuel length. To account for a shorter length, the density of the fuel is adjusted to maintain the same heat capacity for the reconfigured fuel as that of the intact fuel.

The calculated maximum temperatures of the package components under fuel assembly reconfiguration are shown in Table 3-24 of the application. The steady-state component temperatures (T_∞) were obtained from the transient model at 25 hours after the end of the 30-minute fire accident.

The maximum seal temperature for NCT evaluation with reconfigured fuel is 240°F at the top of the test port seal for a 3 kW heat load with the TN-LC package in an ISO container. This temperature is significantly below the long-term limit of 400°F specified for a continued seal functional performance. The maximum seal temperature for HAC evaluation with reconfigured fuel is 314°F for the top test port seal. This temperature is significantly below the long-term limit of 400°F specified for continued seal function performance.

As seen from Table 3-24, the maximum seal, neutron/gamma shield and wood temperatures remain below the allowable limits for NCT and the maximum seal, gamma shield temperatures remain below the limits for HAC. Therefore the shielding and containment functions of the TN-LC transport cask are assured for NCT and HAC and the design criteria specified in Section 3.1 are satisfied.

3.7 Appendix

The applicant provided seven (7) Appendices to Chapter 3 of the SAR, which are listed below.

Appendix 3.6.1 Macros for Heat Transfer Coefficient
Appendix 3.6.2 TN-LC Package Mesh Sensitivity
Appendix 3.6.3 Sensitivity Analysis for Material Properties
Appendix 3.6.4 Mesh Sensitivity of Fuel Basket Models
Appendix 3.6.5 Effective Properties for the Homogenized TN-LC-1FA Basket with PWR Fuel Assembly
Appendix 3.6.6 Effective Thermal Properties of the PWR and BWR Fuel Assemblies
Appendix 3.6.7 Bounding Transverse Fuel Effective Thermal Conductivity for UO2, MOX, and EPR Irradiated Fuels

These appendices provided the information necessary for the staff to make its safety findings regarding the adequacy of the design when compared to the requirements in 10 CFR Part 71.

3.7.1 Justification for Assumptions or Analytical Procedures

The applicant justified the approaches taken in modeling the thermal performance of the Model No. TN-LC. The staff was in agreement with the justifications presented by the applicant. While the homogenized fuel models used in this analysis were generally acceptable, the applicant should consider validating the thermal methods used against available spent fuel package temperature data (e.g., INEEL/EPRI dry package storage data, such as *The TN-24P PWR Spent Fuel Storage Package: Testing and Analyses*, EPRI-NP-5128, 1983; Performance Testing and Analyses of the VSC-17 Ventilated Concrete Package, EPRI-TR-100305, 1992; *CASTOR-1C Spent Fuel Storage Package Decay Heat, Heat Transfer, and Shielding Analyses,* PNL-5974, 1984).

3.7.2 Computer Program Description

The applicant provided a brief description of the ANSYS® finite element analysis code in Section 3.1.3.1 of the application. Because ANSYS® is a widely used and accepted general purpose code, and one the staff is familiar with, the description provided by the applicant was adequate.

3.7.3 Computer Input and Output Files

The applicant provided ANSYS® analysis files containing the models used in the thermal analysis with the application.

3.8 Evaluation Findings

The staff has reviewed the package description, the material properties, and component specifications used in the thermal evaluation and has reasonable assurance that the information provided satisfies the thermal requirements of 10 CFR Part 71. The staff has reviewed the methods used in the thermal evaluation and has reasonable assurance that the models are described in sufficient detail to permit an independent review of the package thermal design. The application of the analysis methods, presented in the application, to this package design has been found to be adequate.

The staff has reviewed the accessible surface temperatures of the package, as it will be prepared for shipment, and has reasonable assurance that the requirements of 10 CFR 71.43(g) for packages transported by exclusive-use vehicle have been satisfied. The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that the package material and component temperatures will not extend beyond the specified allowable limits during NCT and HAC consistent with the tests specified in 10 CFR 71.71 and 10 CFR 71.73, respectively.

4.0 CONTAINMENT REVIEW

4.1 Description of the Containment System

4.1.1 Containment Boundary

The containment boundary components of the package consist of the inner shell, the bottom flange, the bottom plug, the bottom plug O-ring, the top flange, the lid, the lid inner O-ring seal and vent and drain port plug bolts and seals. The containment vessel is designed to prevent leakage of radioactive material from the package cavity and maintain an inert atmosphere (helium) in the package cavity; helium assists in heat removal and protects fuel assemblies against fuel cladding degradation which may lead to gross rupture. The staff confirms that all components of the containment system are well defined, and that the package design is evaluated to demonstrate that it satisfies the containment requirements of 10 CFR 71.31(a)(1), 71.31(a)(2), and 71.33, and 71.43.

The applicant described that the penetrations through the containment boundary are the drain port, the vent port, the bottom plug plate, and the lid. Each penetration is designed to be leaktight with O-ring seal closure. The lid and bottom plugs have double O-rings which provide redundant closure. All containment penetrations are displayed in the application and on the licensing drawings.

All containment boundary welds are full penetration bevel or groove welds to ensure structural and containment integrity. The full penetration welds are designed per ASME, Section III, Subsection NB and are fully examined by radiography (RT) or ultrasonic (UT) methods in accordance with Subsection NB. A liquid penetration (PT) examination is also performed on the containment welds. A Viton fluorocarbon elastomeric seal was chosen for this package because it has acceptable characteristics over a wide range of parameters. The staff verified that:

- the maximum Viton fluorocarbon seal temperatures of 205°F (96°C) for NCT and 449°F (232°C) for HAC are below the corresponding temperature limits of 400°F (204°C) for NCT and 482°F (250°C) for HAC, specified for a continuous seal function,
- the Viton fluorocarbon elastomeric seal (VM835-75) is functional down to -40°F (-40°C) and up to +400°F (+204°C), both temperature limits with excellent compression set resistance, and

- the Viton fluorocarbon elastomeric seal (VM835-75) is leaktight at 470°F (243°C) for 10 hours and at 500°F (260°C) for 3 hours under accident conditions. The VM835-75 compound remains leaktight at 482°F (250°C) under HAC.

The staff reviewed the licensing drawings, Chapters 4, 7 and 8 of the application and verified that all containment boundary welds are to be examined and inspected appropriately in accordance with ASME B&PV Code, Section III, Subsection NB.

4.1.2 Codes and Standards

The containment vessel is designed to meet the requirements of ASME BP&V Code, Section III, Subsection NB, Article 3200. The containment vessel is fabricated and examined in accordance with NB-2500, NB-4000, and NB-5000. The weld materials confirm to ASME B&PV Code, Section III, Division 1, NB-2400 and material specifications of Section II, Part C, of ASME B&PV Code. The containment vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200.

4.2 Containment under NCT

The Model No. TN-LC package is designed and tested for a leak rate of 1.0×10^{-7} ref-cm^3/sec.

4.2.1 Pressurization of Containment Vessel

The package cavity is drained, dried, and evacuated prior to backfilling with helium at the end of fuel loading operations. If the package contains design basis fuel and has been in storage, the average helium temperature in the package cavity is 410°F (210°C) under the ambient air of 100°F (38°C) and the maximum solar heat load.

The applicant determined the maximum normal operating pressure (MNOP) of 16.9 psig for NCT based on the maximum head load of 3 kW and the maximum burnup of 70 GWd/MTU. The staff reviewed Sections 2.6.1 and 3.1.4 of the application, and verified that the MNOP of 16.9 psig is below the test pressure of 30 psig considered for the structural evaluation under NCT.

4.2.2 Containment Criteria

The package must meet the leak-tight containment criteria of ANSI N14.5, under NCT.

4.3 Containment under HAC

The package must meet the leak-tight containment criteria of ANSI N14.5, under HAC.

The applicant determined the maximum pressure of 90.9 psig for HAC based on the maximum head load of 3 kW and the maximum burnup of 70 GWD/MTU. The staff reviewed Sections 2.7.4 and 3.1.4 of the application and verified that the maximum pressure of 90.9 psig is bounded by the test pressure of 120 psig considered for the structural evaluation under HAC.

4.4 Evaluation findings

The staff reviewed the containment design features presented in Chapters 1, 4, 7, and 8 of the application, and confirms that (1) the maximum temperatures of fuel cladding and components are below the NCT and HAC limits, (2) the O-ring seals are leaktight under NCT and HAC, (3) containment criteria remain satisfied for fragmented high burnup fuel under NCT and HAC, and (4) the package maintains an inert atmosphere (helium) in the package cavity. Helium assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation which might lead to gross rupture, in compliance with 10 CFR 71.43(d).

Based on the containment evaluation, the staff concludes that the containment design of the Model No. TN-LC package has been adequately described and evaluated and that the package design satisfies the containment requirements of 10 CFR Part 71 under NCT and HAC.

The staff has reviewed the description of the leakage rates under NCT and HAC and has reasonable assurance that there shall be no release of radioactive contents under NCT and HAC. The staff has reviewed the package preparations for shipment and has reasonable assurance that the containment of the package will not extend beyond the specified allowable limits during NCT and HAC consistent with the tests specified in 10 CFR 71.71 and 71.73, respectively.

5.0 SHIELDING REVIEW

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 Model No. TN-LC package, and to verify that the package design meets the external radiation requirements of 10 CFR Part 71 under NCT and HAC.

5.1 Shielding Design Features

The lead and steel shells of the package provide shielding between the fuel and the exterior surface of the package for the attenuation of gamma radiation. Neutron shielding is provided by a borated resin compound, cast into long, slender aluminum containers surrounding the outer shell. The thickness of the resin is 3.75 inches, and the array of resin-filled containers is enclosed within a 0.25 inch thick outer stainless steel shell.

5.2 Radiation Source Specification

The applicant calculated the radiation source terms using the TRITON module of the SCALE6 code package for MTR, NRU/NRX, and TRIGA fuels. The gamma and neutron source terms for BWR and PWR fuel assemblies and rods were calculated using the SAS2H module of the SCALE4.4 code package.

For each fuel type, a bounding source term was developed for the NCT analysis. For MTR, NRU/NRX, and TRIGA fuels, the same source term was used for NCT and HAC analyses. The applicant stated that, because the contribution of neutron radiation sources to the total dose rate is negligible, the total dose rate at some or all of the locations of interest are lower than the regulatory limits. The staff looked at Tables 5.1, 5.2, and 5.3 of the application, and found that, for both gamma and neutron, the dose rates are lower than the regulatory limits. Also, the source terms were verified using the ORIGEN-ARP depletion code. For a BWR or PWR fuel, the contribution of the neutron radiation source of the total dose rate is large and separate HAC source terms are developed because the loss of the neutron shield increases the neutron contribution to the dose rate.

The source term calculation methodology documented for the TN-LC-1FA basket contents was revised in its entirety to consider the use of the TRITON/T-DEPL module for depletion. The validation of the TRITON methodology for this purpose was documented in detail in Section 5.6.4.2.5 of Appendix 5.6.4 of the application. Because SAS2H was used to compute the gamma and neutron source terms, the staff recommended the applicant to perform a 2-D depletion analysis sequence in order to validate the SAS2H calculations.

5.2.1 Selection of Design Basis Assembly

The applicant selected the bounding fuel assembly based on the highest uranium loading. The bounding fuel assembly type selected for PWRs was the B&W 15x15 Mark B10, which has 0.490 metric tons of uranium (MTU) load per fuel assembly. The bounding fuel assembly type selected for BWRs was the GE 7x7, Version GE1, 2, or 3, with 0.198 MTU per assembly. The applicant calculated the source terms for each of design basis fuel assemblies. For the bounding fuel assembly types, detailed material masses in each exposure zone (Top Nozzle, Plenum, In-core Region, and Bottom Nozzle) are provided in Table 5.6.4-5 and Table 5.6.4-6 of the application for the B&W 15x15 and GE 7x7, respectively.

For MOX fuel, the heavy metal loading of a BWR MOX rod is approximately 4.0 kg, while the heavy metal loading of an EPR MOX rod is approximately 2.0 kg. Therefore, applying a source term computed for a BWR MOX rod to an EPR MOX rod is very conservative. The applicant explained that this conservatism might be quantified by developing a representative SAS2H model for EPR MOX rods. The source terms for 9 EPR MOX rods are shown in Table 5.6.4-51 of the application. The gamma and neutron source strength of an EPR MOX rod is approximately 50% of the design basis BWR MOX rod. Therefore, the dose rates from EPR MOX rods will be significantly less than reported in Table 5.6.4-32 of the application.

5.2.2 Gamma Source

The applicant performed gamma source calculations using the TRITON module of the SCALE6 code package. TRITON was developed to allow for a two-dimensional representation of the fuel elements. Because the input is two dimensional, all input is for a basis of 1 MTU or MTHM. All TRITON outputs were also per MTU or MTHM, so the results were scaled by the MTU or MTHM per the content, i.e., per fuel assembly or fuel element. The gamma source terms for BWR and PWR fuel assemblies and rods are calculated using SAS2H module of the SCALE4.4 code package.

5.2.2.1 MTR Fuel

The key characteristics defining the source at a given cooling time are the U-235 mass, enrichment, and burnup. A generic MTR fuel element was developed for different cases, or enrichment types, which bound the MTR fuel element types listed in Chapter 1 of the application. A full summary of the data used to develop the TRITON model is presented in Table 5.6.1-3 of the application.

5.2.2.2 NRU and NRX Fuels

The applicant stated that the NRU fuel matrix is a mixture of aluminum and high-enriched uranium and the use of this type of fuel was stopped two decade ago. Therefore, the decay time for this fuel is at least 18 years. A decay time of 10 years is conservatively selected to bound any potential uncertainties in the source term.

5.2.2.3 TRIGA Fuel

The applicant summarized the data used to develop the 2-D TRITON models in Table 5.6.3-3 of the application. Given the H/Zr atom ratio, the maximum uranium mass in the U-ZrH fuel matrix, and the mass of U-235 and U238, the mass of hydrogen and zirconium might be computed for each fuel type. These masses and the known fuel volumes are then used to compute number densities for input to TRITON.

The decay times to meet dose rate limits, the decay heat, as well as the NCT dose rate value used to select the bounding source, are provided in Table 5.6.3-4 of the application. Based on these results, the bounding source term for NCT dose rate calculations is for ACPR fuel at the maximum burnup and a decay time of 1,870 days. The design basis gamma source term is presented in Table 5.6.3-5 of the application.

5.2.2.4 BWR and PWR Fuel

The applicant established the design basis taking into consideration the fuel qualification table (FQT), which is the combination of burnup, enrichment, and cooling time. Four SAS2H models were developed for each design basis source, representing the top nozzle, plenum, in-core, and bottom nozzle regions. These SAS2H models differ only in the treatment of light elements. An example of SAS2H input file is provided in Section 5.6.4.5.2 of the application for the in-core region of a PWR fuel assembly.

5.2.3 Neutron Source

The applicant performed neutron source calculations with the TRITON module of the SCALE6 code package. The neutron source is comprised of both spontaneous fission and (α,n) reactions with the aluminum in the fuel matrix. Like the gamma calculation, a neutron response function is also generated for NCT. The applicant explained that the response function methodology employed in the analysis of the TN-LC transportation system is derived from underlying assumptions regarding the mathematical analysis of the transport of neutral particles through materials. Rather than numerically solve the transport equation for every possible source, either by Monte Carlo or discretely, it is possible to develop a series of response functions that collapse the physical properties of the system into a set of energy dependent linear factors for specific reaction types of interest.

5.2.3.1 MTR Fuel

In order to determine the dose rate from an actual source, the applicant multiplied the neutron in each energy group for a single fuel element by the response function for that energy group, and summed the results. The neutron sources were presented in Table 5.6.1-7 of the application.

5.2.3.2 NRU and NRX fuels

For the NRU and NRX fuels, the applicant provided the neutron source in Table 5.6.2-6 of the application (NRU (586 g uranium) and NRX with either heavy or light water). The NRX with heavy water source was the bounding source used in the dose rate calculations. The NRU reactor, built as the successor to the NRX reactor, uses heavy water as both moderator and coolant. To bound the actual NRX reactor, TN created the NRX depletion models using both heavy and light water. The results of these depletion models concluded that the NRX neutron source with heavy moderator and heavy coolant is conservative.

5.2.3.3 TRIGA Fuel

For TRIGA fuel, the applicant generated the neutron source using the same TRITON models from which the gamma source is generated. The neutron source is primarily from spontaneous fission because there are few (α,n) target nuclei in the fuel matrix. The neutron NCT response function was provided in Table 5.6.3-8 of the application and the neutron source is for ACPR fuel at maximum burnup.

5.2.3.4 PWR and BWR Fuel

For PWR and BWR fuels, the applicant developed the neutron response functions using the same methodology as the gamma response functions except that the default Cm-244 fission spectrum were utilized in MCNP because the neutron source is primarily from spontaneous fission of Cm-244. Therefore, no explicit neutron energy groups were developed in the MCNP input, and the total number of source particles in the neutron response function models was simply 1 n/s. Staff evaluated the output file provided by the applicant in order to verify that the primary neutron source came from spontaneous fission.

Since the neutron source strength for a UO_2 assembly increases as enrichment decreases for a constant burnup and decay time, the licensee performed source calculations using enrichments of 3.15 wt%, 3.95 wt%, and 4.5 wt% to develop the source terms for the UO_2 fuel. For both fuel types, the spontaneous fission of Cm-244 isotopes accounts for approximately 95% of the total number of neutrons produced. (Alpha,n) reactions in Cm-244 account for 1% of the neutrons produced. Any neutrons generated from subcritical multiplication, (n, 2n) or similar reactions are properly accounted for in the MCNP shielding calculations.

For the PWR and BWR fuel assembly neutron response functions, the response function included the factors 1.152 and 1.326 shown in Table 5.6.4-18 and Table 5.6.4-19, respectively. These factors represent the increase in the neutron source magnitude when an axial source profile was applied because the neutron source computed by SAS2H is for an average assembly burnup.

5.2.3 Uncertainties in the Depletion Calculation

The various uncertainties associated with SAS2H and ORIGEN-S calculations were evaluated by the applicant and incorporated into the shielding calculations. These uncertainties included those inherent to the code as well as the input data. Variations in the gamma and neutron source terms from variations in the input parameters and the variations in heat loads, gamma and neutron source terms resulting from the uncertainty in the isotope calculations were determined. In all cases, the uncertainties in the amount of relevant isotopes were taken from published references.

5.2.4 Specific Power

The applicant used the fuel assembly specific power (expressed in MW/fuel assembly (MW/FA)) and the total time between cycles for the source term calculations. According to NUREG/CR-6701, the most important parameter for the calculation of source terms is the specific power. While specific power for typical US-BWR fuel assemblies does not exceed 8.0 MW/FA, the source terms for this evaluation were calculated using a specific power up to 16 MW/FA (at higher burnups) to result in a conservative estimation of the source terms. The time between cycles utilized was 73 days and represented a typical downtime for US BWRs (60 to 90 days). Specific power for typical US-PWR fuel assemblies was ≤ 18 MW/FA. The time between cycles, utilized in SAS2H/ORIGEN-S depletion models of PWR assemblies, was 30 days and is adequately bounding.

5.3 Shielding Model

The shielding analysis of the Model No. TN-LC transport package is performed with MCNP4A using the continuous energy ENDF/B-VI neutron and photon cross section libraries. The MCNP computer code was used to calculate the dose rate calculations for selected bounding source using a “response function” developed by the code. To develop a response function, a fairly detailed MCNP model of the MTR fuel, basket, and TN-LC package was developed.

For NCT conditions, the model includes both the neutron shield and impact limiters while, for HAC, the neutron shield is replaced with air and the impact limiters are completely removed. Also, 1.2 inches of lead slump were modeled at both the top and bottom ends of the radial cask lead as a result of an end drop. This bounds the maximum lead slump value listed in Chapter 2 of the application.

5.3.1 MTR Basket

The package may transport a maximum of 54 MTR fuel elements. Only the active fuel region of the MTR element was modeled, and the source was evenly distributed throughout the fuel meat material, which was modeled with an active length of 24 inches. Important dimensions of the TN-LC-MTR basket model are summarized in Table 5.6.1-9 of the application.

5.3.2 NRUX Basket

The fuel, basket, and packaging are modeled explicitly in the MCNP computer program. The NRX fuel element type was used for the source geometry, as the self-shielding differences between the NRX and NRUX fuel types are negligible due to the similarity between the fuels. The source was evenly distributed throughout the fuel pellet material, which was modeled with an active length of 8 ft. Only the active fuel region of the NRX element was modeled, and it was assumed that any flow tubes are removed. The end regions and flow tubes were aluminum and provided negligible source. Because the end regions and flow tubes would provide some shielding, it was conservative to neglect them. Under HAC conditions, the impact limiter wood and neutron shield resin were replaced with air. No credit was taken for the distance between the impact limiter outer surface and the cask lid. This bounds any postulated fire or crush damage.

5.3.3 TRIGA Basket

The fuel, basket, and packaging were modeled explicitly in the MCNP computer program. Under HAC, the impact limiter wood and neutron shield resin are replaced with air. This bounds any postulated fire or crush damage. In addition, 1.2 inches of lead slump are modeled at both the top and bottom ends of the cask as a result of an end drop. This bounds the maximum lead slump value of 1.129 inches from Appendix 2.13.3 of the application. The radial lead slump at the cask ends due to a side drop is negligible (<0.2 in.) and has been neglected. HAC dose rates are conservatively computed 1 m from the cask body surface.

5.3.4 1FA Basket

The fuel, basket, and packaging were modeled explicitly in the MCNP computer program. Separate models were developed for the PWR fuel assembly, BWR fuel assembly, 25 or 9 PWR rods in a pin can, 25 or 9 BWR rods in a pin can, and 25 or 9 MOX rods in a pin can. Separate models were also developed for 25 or 9 EPR rods (standard or MOX) in a pin can. EPR rods were treated separately because these rods were longer than a standard PWR rod and, hence, use a different pin can with less axial shielding.

The applicant stated that the gamma and neutron axial profiles were used in the active fuel regions in the fuel assembly models. The axial profiles account for the change in burnup along the axial length, as the fuel assembly source terms were computed for an average assembly burnup. These axial profiles conservatively increase the radial dose rates next to the axial peak.

Under HAC, the impact limiter wood and neutron shield resin are replaced with air. This configuration bounds any postulated fire or crush damage. In addition, 1.2 inches of axial lead slump are modeled at both the top and bottom ends of the radial cask lead as a result of an end drop.

5. 4 Shielding Evaluation

5.4.1 Methods

The applicant utilized a large number of conservative assumptions throughout their shielding calculations to provide assurance that the actual dose rates will always be below the calculated dose rates, as well as below regulatory limits. Minimum dimensions are used where applicable.

The applicant used SAS2H to compute the source terms rather than a more detailed 2-D program, such as TRITON, because the as-modeled fuel assembly designs are simple and may be modeled conservatively in SAS2H. The staff noticed that using SAS2H as a depletion code could not be the best choice for the source term calculations. This code is no longer supported primarily by the developer because it cannot be used with any of the modern cross section libraries (it will only work with ENDF/B-IV and –V). There have been some very significant improvements in data evaluations, particularly for fission products in ENDF/B-VII. SAS2H should be adequate for many applications such as determining assembly averaged characteristics. It is expected that the use of the code should be accompanied by some validation studies that quantify the accuracy for the nuclides of importance, and that any bias in the calculations would be incorporated in a margin for safety. In order to confirm the adequacy of the SAS2H generated source terms, the design basis NCT source terms for the PWR and BWR fuel assemblies were regenerated using the 2-D TRITON module of the SCALE 6 code package. In both cases, the TRITON generated gamma and neutron source terms are less than the SAS2H generated source terms, which allow SAS2H to be more conservative. The results of the shielding evaluation showed that the maximum calculated NCT dose rate for the TN-LC-1FA basket at 2 m is approximately 8.9 mrem/hr. This dose rate includes uncertainties on the response function methodology (approximately 10%) and on MCNP dose rate calculations (approximately 1%). The method by which the gamma and neutron source term uncertainties were evaluated was documented in Section 5.6.4.2.5 of Appendix 5.6.4 of the application. The final results obtained justify the use of 10.0% and 5.0% uncertainty for gamma and neutron dose rates, respectively in the TN-LC-1FA shielding analysis. The staff utilized ORIGEN-ARP depletion code, which is part of the SCALE6.1 depletion code package to do confirmatory analyzes and verify that the results provided by the applicant were in agreement with the staff's calculations. Every combination of burnup, enrichment, and cooling time for the different PWR fuel assemblies was analyzed. The results obtained by the staff were very close to the ones provided by the applicant.

The MCNP5 v.1.40 code package was used for the entire applicant's shielding analyses. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code and is one of the standard codes used in the nuclear industry for calculating dose rates. The staff verified the MCNP input provided by the applicant and found it acceptable. Separate models were developed for neutron and gamma source terms. Geometry splitting and simple Russian roulette were used as a variance reduction technique for most tallies. The importance of the particles increases as the particles traverse the shielding materials.

5.4.2 Input and Output Data

The applicant has submitted a number of input/output cases that were used to generate all the results. According to the applicant, some models were restarted to increase the run time to allow better convergence. Convergence is good (σ<10 percent) for all total dose rates of interest.

5.4.3 Flux-to-Dose-Rate Conversion

The flux-to-dose rate conversion factors were provided in Section 5.4.3 of the application. The factors were consistent with the guidance provided in NUREG-1617.

5.4.4 External Radiation Level

The applicant selected tally locations in order to be consistent with exclusive use transportation in a closed transport vehicle. Circumferential tallies were placed around the packaging: 29 axial locations were utilized. At the ends of the packaging, dose rates were tallied on the impact limiter surfaces and at 2 m from the impact limiter surfaces.

The calculated energy distribution of the source term is used explicitly in the MCNP model. Separate calculations are performed for each of the three source terms (i.e., decay gamma, (n.gamma), neutron, and Co-60). The axial burnup distributions are representative of the fuel type to be loaded. The cobalt source present in the steel hardware is assumed to be uniformly distributed over the appropriate regions. As mentioned before, the source terms calculated by the staff were confirmed to be in agreement with the applicant. Also, the effects of streaming at 2 m from the ends of the transport vehicle were shown to be small.

The basket designs were not circumferentially symmetric. Therefore, the dose rate varied around the perimeter of the package. This effect was most pronounced close to the package surface, and diminishes with distance. The applicant also captured the neutron streaming path at different locations such as the impact limiter and the shear key using explicitly angular mesh tallies.

For the HAC model, the neutron shield resin and impact limiter wood was replaced with air, and the tally surfaces were located 1 m from the outer surfaces of the cask.

5.4.4.1 NCT

All the TN-LC-1FA cases were developed only for the PWR fuel assembly and 25 BWR MOX rods. The 25 BWR MOX rods had the largest neutron source, although the PWR fuel assembly had less shielding. The applicant stated that the lead slump has little effect on the dose rate because the dose rate is dominated by neutrons and peaks near the axial center. The maximum side dose rate for the PWR fuel assembly is 341 mrem/hr. The maximum radial dose rate using the 25 BWR MOX rods resulted to be 436 mrem/hr, which is larger than when the PWR fuel assembly source was utilized.

The applicant used MCNP to calculate dose rates at the various desired locations. Since MCNP calculates neutron or photon fluxes, these values were converted to dose rates and use the response functions from ANSI/ANS 6.1.1-1977. External dose rates were calculated on the surface of the package and 2 m from the edge of the transport vehicle during NCT**.** For NCT the maximum dose rates at 1 m from the surface of the cask were calculated. The applicant looked at reasonable fuel reconfigurations and determined that the expected dose rates would still be below regulatory limits.

5.4.4.2 HAC

For the TN-LC MTR basket, at the package surface, the maximum dose rate occurred on the top of the impact limiter over the port with a dose rate of 57.3 mrem/hr, i.e., below the limit of 1000 mrem/hr. At the vehicle surface, the maximum vehicle surface dose rate of 5.2 mrem/hr occurs on the impact limiter surface over the port, which bounds the vehicle surface dose rates on the underside, side, and bottom ends, and is below the limit of 200 mrem/hr. At 2 m from the vehicle, the maximum dose rate is 6.75 mrem/hr, which is less than the limit of 10 mrem/hr.

In the HAC models, the neutron shield resin and impact limiter wood was replaced with air, and 1.2 inches of lead slump was modeled at both the top and bottom ends. The maximum HAC dose rate of 148 mrem/hr occurred at 1 m from the side of the package near the lead slump region. This dose rate is significantly less than the limit of 1000 mrem/hr.

For the TN-NRUX basket, at the package surface, the maximum dose rate occurred on the top of the impact limiter over the port with a dose rate of 84.5 mrem/hr, below the regulatory limit of 1000 mrem/hr. At the vehicle surface, the maximum vehicle surface dose rate of 84.5 mrem/hr occurred on the impact limiter surface over the port, and at 2 m from the vehicle, the maximum dose rate is 2.79 mrem/hr, which is below the limit of 10 mrem/hr.

In the HAC models, the neutron shield resin and impact limiter wood was replaced with air, and 1.2 inches of lead slump was modeled at both the top and bottom ends. The maximum HAC dose rate of 38.7 mrem/hr occurred 1 m from the side of the package near the lead slump region. This dose rate is significantly less than the limit of 1000 mrem/hr.

For the TN-TRIGA basket, at the package surface, the maximum dose rate occurred on the top of the impact limiter over the port with a dose rate of 95.8 mrem/hr. At the vehicle surface, the maximum vehicle surface dose rate is 49.8 mrem/hr, and at 2 m from the vehicle, the maximum dose rate resulted to be 8.27 mrem/hr, which is below the limit of 10 mrem/hr.

In the HAC models, the neutron shield resin and impact limiter wood was replaced with air, and 1.2 inches of lead slump was modeled at both the top and bottom ends. The maximum HAC dose rate of 79.8 mrem/hr occurred 1 m from the side of the package near the lead slump region. This dose rate is significantly less than the limit of 1000 mrem/hr.

The maximum calculated dose rate for the various contents authorized for the TN-LC-1FA basket is for the PWR fuel assembly and is equal to 8.92 mrem/hour at 2 m. The revised dose rates for all the other contents are less than that for the PWR fuel assembly and are shown in Table 5.6.4-2 of the application, Appendix 5.6.4, for NCT.

In Section 5.6.4.4.4.4, the applicant analyzed the impact of the fuel reconfiguration on the external dose rate under NCT with different fuel assembly models. Based on the approaches provided in NUREG/CR-6835, "Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks," the staff agreed that the assumption of a 100% source term concentration is bounding and acceptable.

5.5 Evaluation Findings

The staff reviewed the description of the package design features related to shielding and the source terms for the design basis fuel and found them acceptable. The methods used are consistent with accepted industry practices and standards. The staff also reviewed the maximum dose rates for NCT and HAC and determined that the reported values were below the regulatory limit in 10 CFR 71.47 and 71.51. Based on its review of the statements and representations provided in the application, the staff has reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and NRC guidance, and that the package design satisfies the shielding and dose rate limits requirements of 10 CFR Part 71.

6.0 CRITICALITY REVIEW

The objective of the criticality review is to verify that the Model No. TN-LC package design satisfies the criticality safety requirements of 10 CFR Part 71. The staff's review considered the guidance presented in NUREG-1617 and NUREG-1609.

6.1 Description of the Criticality Design

The package uses four different baskets designed to transport highly enriched aluminum-uranium plate fuel, highly enriched aluminum-uranium pin fuel, or commercial light water reactor fuel assemblies and pins. The staff verified that the information provided for each of the four basket designs is consistent and that all descriptions, drawings, figures, and tables are sufficiently detailed and complete to support an in-depth criticality evaluation.

6.1.1 Packaging Design Features

The staff reviewed the licensing drawings and found that they sufficiently describe the locations, dimensions, and tolerances of the containment system, basket, and neutron absorbing material (when used). Therefore, the staff finds that the applicant meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5) with respect to the criticality evaluation.

The packaging design feature primarily relied upon to prevent criticality is the basket geometry in each of the four basket designs. The TN-LC-TRIGA and TN-LC-1FA baskets also require fixed neutron poisons for criticality control. For NCT array calculations, the separation between baskets provided by the packaging ensures criticality safety. Staff notes that impact limiters are omitted in all criticality models, although credit is taken for the distance provided by the impact limiters in the NCT array analysis for the MTR basket. Staff agrees that such credit can be taken under NCT because the impact limiters are demonstrated to be in place after NCT tests, and can therefore be relied on for spacing.

The TN-LC-TRIGA basket design includes poison plates between adjacent compartments containing TRIGA fuel. The poison plates are modeled with a B-10 areal density of 5 mg/cm^2. The neutron absorbing poison plates consist of either boron enriched aluminum alloy or metal matrix composite (MMC), which has a minimum B-10 content of 5.56 mg/cm^2 (if 90% B-10 credit is used), or Boral®, which has a minimum B-10 content of 6.67 mg/cm^2 (if 75% B-10 credit is used).

The TN-LC-1FA basket design includes poison plates that surround the basket containing the fuel assembly or rod pins. The poison plates are modeled with a B-10 areal density of 15 mg/cm^2. The neutron absorbing poison plates consist of either boron enriched aluminum alloy or MMC which has a minimum B-10 content of 16.67 mg/cm^2 (if 90% B-10 credit is used) or Boral® which has a minimum B-10 content of 20 mg/cm^2 (if 75% B-10 credit is used). Additionally, Poison Rod Assemblies (PRAs) are required while transporting PWR fuel assemblies (PRAs are not required for BWR or MOX fuel assemblies) in order to ensure that the maximum reactivity is subcritical and below the Upper Subcritical Limit (USL). The minimum required B_4C content of the absorber rods in the PRAs is 40% theoretical density (TD) (75% credit is taken in the criticality analysis, or 30% TD).

Criticality safety for the TN-LC-TRIGA and TN-LC-1FA baskets is maintained by the combination of the neutron poison, the basket geometry, and the separation provided by the packaging.

6.1.2 Summary Table of Criticality Evaluations

The applicant used the methods recommended in NUREG/CR-6361 to establish Upper Subcritical Limits (USLs) for the calculated k_{eff} values of the package. The USL value includes the calculational bias and the administrative margin. The final values reported for k_{eff} include a factor of two times the standard deviation of the statistical uncertainty of the calculated value. The reported values of k_{eff} were then compared against the minimum USL values for each of the basket payloads and these results illustrate that the package design meets the criticality safety requirements of 10 CFR Part 71.

No credit is taken for burnup in any of the analyses, and no credit is taken for the leak-tight performance of the package. Since the Criticality Safety Index (CSI) of this package was assumed by the applicant to be 100 (see Section 6.1.3 of this SER), the HAC analysis of an array of packages was not necessary. Calculations with a triangular array of three packages, the minimum required for a CSI of 100, were performed as part of the NCT analysis. Partial moderation was considered between the packages to maximize any array interaction effects.

Water was modeled in all cavities with the most reactive credible density for both NCT and HAC conditions. Sensitivity studies were performed to determine the most reactive water density. Preferential flooding was not considered due to the drainage holes present in each basket to prevent this occurrence. In all single package models, 12 inches or more of water reflection was assumed.

For modeling fuel damage in the TN-LC-MTR and TN-LC-NURX baskets under both NCT and HAC, fuel elements were modeled with reduced pitch, until the fuel rods contact, and with increased pitch, up to the maximum allowed by the basket structure. In the fuel element models, the most reactive credible configuration occurred from maximizing the distance between fuel elements. Maximizing this distance simulates fuel damaged under HAC and maximizes the moderation and hence the reactivity because the system is under-moderated.

For modeling fuel damage in the TN-LC-TRIGA basket under both NCT and HAC, fuel elements were modeled by ignoring the plenums and end fittings and therefore assuming total axial collapse of the plenums. This configuration bounds the geometry for any credible scenario under HAC in which the fuel is damaged, since the fuel assemblies fill most of the lateral void space, lateral bending of the fuel assemblies under HAC would be negligible and lateral fuel damage is not expected.

For damaged fuel in the TN-LC-1FA basket, the applicant performed evaluations with several different fuel configurations intended to bound the range of possible fuel initial conditions or reconfiguration under HAC. These configurations included:

(i) Single-ended rod shear – in which a row of fuel rods is sheared from the parent fuel assembly, and may relocate to a new location. The fuel pellets are assumed to remain in the fuel rod.

(ii) Double-ended rod shear – in which a row of fuel rods shears radially from the parent fuel assembly and then breaks axially into two pieces, with half of the rods moving up or down to make an extra row in that region of the fuel assembly. This configuration is conservatively modeled as an extra full-length row of fuel rods in the assembly, artificially increasing the amount of fuel present.

(iii) Rod pitch variation – where the initial rod pitch is expanded to allow more moderation in the assembly, limited only by the internal dimension of the guide sleeve.

The applicant subsequently evaluated these damaged fuel configurations for the reactivity effect of rod removal and loss of cladding. A maximum of 20 rods were removed from each fuel assembly model. The most reactive missing rods models were then evaluated with 6" of the fuel rods decladded. The applicant assumed that the fuel pellets stay stacked on top of each other and also retain the lattice configuration of the fuel assembly. Therefore, the bare fuel rods for the assembly were modeled according to the lattice configuration of the assembly, with no missing rod locations in the 6" section of the array.

In the TN-LC-MTR, TN-LC-NRUX, and TN-LC-TRIGA basket analyses, fuel damage was conservatively modeled in both the NCT and HAC models. Also, internal and external flooding was considered for both the NCT and HAC models. Therefore, for these baskets, the most reactive case is the NCT array. In the TN-LC-1FA basket analysis, fuel was modeled as undamaged in the NCT cases and damaged in the HAC cases. Therefore, the HAC single package case is limiting in the TN-LC-1FA basket analysis.

The applicant performed analyses for a single package under conditions of 10 CFR 71.55(b), (d), and (e), and for undamaged and damaged arrays of packages under their respective conditions specified in 10 CFR 71.59(a)(1) and (2). The results of these analyses were presented in tables that showed the calculated keff's and their standard errors. Table 6-1 of the application contains a summary of the final analysis results of the criticality safety analyses. The package or package array is considered to be subcritical if k_{safe} (k_s) for each of the analysis cases is less than the USL. The computed k_{safe} is equated as $k_s = k_{eff} + 2\sigma <$ USL. Staff reviewed these tables and found that the most reactive cases are clearly indicated, and were demonstrated to be less than the USL.

The maximum k_s for each analysis case, as calculated by the applicant, is summarized in Table 6-1 below for each of the available baskets. The results are less than the associated minimum USL value for each basket type and these results illustrate that the package design meets the criticality safety requirements of 10 CFR Part 71 and that the package would remain subcritical under NCT and HAC.

Table 6-1
Summary of Criticality Evaluations

	MTR Payload	NRUX Payload	TRIGA Payload	1FA Payload
Normal Conditions of Transport (NCT)				
Analysis Case	k_s	k_s	k_s	k_s
Single Package Maximum	≤0.918	0.872	0.887	0.8895
Array Maximum (3 packages)	0.918	0.874	0.896	0.9047
Hypothetical Accident Conditions (HAC)				
Analysis Case	k_s	k_s	k_s	k_s
Single Package Maximum	≤0.918	0.872	0.887	0.9418
Array Maximum (1 package)	≤0.918	0.872	0.887	0.9418
USL	0.9213	0.9227	0.9301	0.9420

6.1.3 Criticality Safety Index

The applicant assumes a Criticality Safety Index (CSI) with a value of 100 because the applicant chose to not perform HAC array calculations and therefore being required to assume the maximum number of damaged packages per 10 CFR 71.59(a)(2) is 1, ensuring that the value of "N" = 0.5 (two times "N" = 1). The staff finds that the CSI was appropriately determined per 10 CFR 71.59(b). The staff finds that the applicant meets 10 CFR 71.59(a)(3) because the value of N is not less than 0.5.

6.2 Fissile Material Contents

The applicant does not request credit for burnup of the fuel. Each basket and its allowed contents are described below:

- TN-LC-MTR – designed to transport up to 54 intact MTR fuel assemblies. The basket structure consists of six interlocking layers containing up to nine MTR fuel elements in a 3x3 arrangement for each layer. Each element is composed of between 10 to 23 fuel plates held in place by two parallel aluminum side plates. The fuel plates are compacts composed of aluminum cladding surrounding uranium-aluminum fuel meat. The fuel meat can have one of several chemical compositions: U3O8-Al, U-Al, or U3Si2-Al.

- TN-LC-NRUX – designed to transport up to 26 intact NRU or NRX fuel assemblies. The basket structure consists of two bundles of 13 tubes. The NRU and NRX fuel matrices are a mixture of aluminum and high-enriched uranium. All cladding and structural materials are fabricated from aluminum.

- TN-LC-TRIGA –designed to transport up to 180 intact TRIGA fuel elements in five baskets containing 36 fuel elements. The basket structure consists of a welded assembly of stainless steel square tubes in a 3x3 arrangement that contain four TRIGA fuel elements, separated by borated aluminum, aluminum/B_4C metal matrix composite, or Boral® neutron poison plates for criticality control. The fuel assemblies consist of a cylindrical active fuel region located between two axial graphite reflectors. The fuel consists of a matrix that is a mixture of uranium and zirconium hydride.

- TN-LC-1FA – designed to transport up to 1 intact PWR Assembly, 1 intact BWR Assembly, or 25 intact PWR (including MOX and EPR) or BWR fuel rods in a 25 pin can basket. The basket structure is surrounded by four borated aluminum, aluminum/B_4C metal matrix composite, or Boral® neutron poison plates for criticality control. The PWR compartment surrounds the BWR fuel assembly compartment and the 25 pin can is placed in the BWR compartment. Additional reactivity control is not necessary for the BWR fuel assembly and 25 pin can transportation.

Intact fuel assemblies are defined as fuel assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks. Damaged fuel assemblies with cladding damage in excess of pin hole leaks or hairline cracks are authorized. The extent of the damage is limited such that the total surface area of the damaged cladding does not exceed 5% of the total surface area of each rod. Only intact fuel is allowed in the TN-LC-1FA basket.

The staff finds that the applicant has described the contents in sufficient detail to provide an adequate basis for this evaluation. The staff finds that the applicant has defined adequately the type, maximum quantity, and chemical and physical form of the fissile material in compliance with the requirements of 10 CFR 71.31(a)(1), 10 CFR 71.33(b)(1), 10 CFR 71.33(b)(2), and 10 CFR 71.33(b)(3).

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The applicant evaluated three-dimensional models of a single package under both NCT and HAC. The applicant explicitly models the fuel and basket, and neglects items that have little effect on reactivity, such as the bottom of the basket. The applicant modeled the package body conservatively neglecting the neutron shielding material, allowing more neutron communication between packages in an array, as well as better neutron reflection from the package wall in the single package. Preferential flooding is not considered, due to the drainage holes present in each basket.

The applicant's conclusion is that NCT and HAC conditions have no adverse effect on the geometric form of the package contents important to criticality safety. Despite this conclusion, the applicant modeled fuel damage for each of the basket analyses under HAC. The applicant also conservatively modeled fuel damage under NCT for the TN-LC-MTR, TN-LC-NRUX, and TN-LC-TRIGA basket analyses.

The staff examined the models used for the criticality calculations and verified that the dimensions and materials are consistent with those in the drawings of the actual package. The applicant discusses differences between the drawings in Sections 6.10.1.3.1, 6.10.2.3.1, 6.10.3.3.1, and 6.10.4.3.1 of the application.

The staff verified that the applicant did consider deviations from nominal design configurations by considering tolerances in order to maximize the reactivity, thus analyzing more conservative conditions. Also, any items in the package neglected for simplicity (e.g., lifting lugs, tube cap, etc.) have little effect on the reactivity. The packaging is conservatively modeled without the neutron shield and impact limiters in both the NCT and HAC models. The neutron shield is conservatively ignored to eliminate neutron absorption from the analysis. The omission of the impact limiters results in a minimized separation of casks when modeling an array, possibly increasing the system reactivity. The staff finds this acceptable and finds that the differences will not impact the criticality safety evaluation.

6.3.2 Material Properties

The staff verified that the appropriate atom densities are provided for all materials used in the models of the packaging and contents. No credit was taken for burnable poisons in the fuel. There are no materials in the package that need to be adjusted to be consistent with accident conditions, i.e., there are no materials used in the model that change form, such as a potential melting of the neutron shield or neutron absorbers, that are assumed in the calculations and needed to maintain subcriticality. The staff agrees that the material property descriptions presented in Sections 6.3.1 and 6.3.2 of the application are consistent with the condition of the package under the tests of 10 CFR 71.71 and 71.73.

The applicant's calculations take credit for only 75% of the minimum acceptable B-10 areal density in the BORAL basket absorber material and 90% of the minimum acceptable B-10 areal density in the borated aluminum alloy and boron-carbide/aluminum metal matrix composite basket absorber materials. The percentage credit was commensurate with the degree of boron content verification provided in the acceptance testing program as described in Section 8.1.7 of the application. NUREG/CR-5661 states that: "a percentage of neutron absorber material greater than 75% may be considered in the analysis only if comprehensive acceptance tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented." Section 8.1.7 of the application discusses the tests used to verify the presence and uniformity of the neutron absorber. The specified acceptance testing states that at any location in the material, the minimum specified areal density of B-10 will be found with 95% probability and 95% confidence. Therefore, the staff finds that assuming 90% of the minimum neutron absorber content is acceptable.

The basket materials do not degrade to a point where there could be any impact on criticality safety. A structural analysis was performed which demonstrates that the basket poison plates will remain in place during all accident conditions. The neutron flux in the package is very low such that depletion of the B-10 is negligible.

The compositions and densities for the materials used in the computer models were reviewed by the staff and determined to be acceptable. The staff notes that these materials are not unique and are commonly used in other spent fuel storage and transportation applications.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant performed the criticality evaluations for the TN-LC-MTR, TN-LC-NRUX, and TN-LC-TRIGA baskets using the three-dimensional Monte Carlo computer program MCNP5 (version v1.40) with continuous energy cross-sections primarily from the ENDF/B-VI database. ENDF/B-VII cross-sections were used for uranium isotopes, because they are more accurate than ENDF/B-VI data. Some ENDF/B-V cross-sections were used for other materials when ENDF/B-VI cross-sections were not available. In addition, the applicant used the appropriate

MCNP option to properly account for the hydrogen bound to water in all basket models and the hydrogen and zirconium in the zirconium hydride in the TRIGA fuel basket model.

The applicant performed the criticality evaluations for the TN-LC-1FA basket using the three-dimensional Monte Carlo computer program KENO-V.a of the SCALE 6 software package. In these models, 44 group ENDF/B-V cross sections were used with the NITAWL module for treatment of resonances.

The staff finds that these codes and the aforementioned cross-sections are appropriate for the TN-LC basket analyses. The MCNP and SCALE codes are industry standards for performing criticality analyses and are widely used in industry application for criticality calculations. As a result, the MCNP and SCALE codes and their associated cross-section sets have been extensively benchmarked against critical experiments. Thus, the staff agrees that the codes and cross-section sets used by the applicant are appropriate for this particular package design and contents.

The applicant included a sufficient number of particle histories in its calculations to achieve a statistical standard deviation of less than 0.001 in the calculated values of k_{eff}. To achieve this, most cases were run with 5000 neutrons per generation for 250 generations. The staff considers this to be sufficient for this application.

The application contains sample input and output files for each of the basket types, which staff reviewed to confirm that the model inputs and outputs were consistent with the descriptions in the application. The staff verified that the multiplication factors from the output files agree with those reported in the evaluation and that the calculation of the multiplication factors properly converged. The staff also verified that the information regarding the model configuration, material properties, and cross sections was properly represented in the input files.

6.3.4 Demonstration of Maximum Reactivity

In all models, fuel is modeled as fresh, although the Model No. TN-LC package is a spent fuel package and fuel will typically be much less reactive due to U-235 depletion and fission product buildup. Full flooding is modeled in all NCT and HAC cases, although the package is leak tight and water intrusion during NCT and HAC is not credible. Fuel is modeled in the most reactive configuration within the baskets, including postulated fuel damage as a result of an accident. Additional information about the conservative assumptions used in the criticality analyses is included in the individual appendices for each basket in Chapter 6 of the application.

The applicant also addressed the effects of partial and preferential flooding. The applicant shows in Table 6.10.2-9 for HEU NRU fuel that the fully flooded condition bounds all of the partial flooding cases. It is assumed that the other HEU fuels allowed in the TN-LC package will behave in a similar manner. Due to the design of the baskets (i.e., holes in the basket), preferential flooding is not possible in the Model No. TN-LC package.

6.3.4.1 TN-LC-MTR Basket

Full-water moderation and postulated HAC damage to the fuel is modeled for both the NCT and HAC cases. The postulated fuel damage is fuel plate pitch expansion, which increases the moderation. The NCT single package is identical to the HAC single package; thus, the reactivities are equal. The NCT array produces the limiting reactivity.

The allowable contents for the TN-LC-MTR basket consist of MTR fuel encompassing many different uranium plate fuels of varying enrichment. As discussed in Section 6.10.1.9.3 of the application, the applicant performed a parametric evaluation to determine the limiting (most reactive) fuel type of the potential known MTR fuels for a single fuel element at nominal dimensions in water and enclosed by a layer of stainless steel to simulate the basket. Parameters varied include the number of fuel plates, the U-235 loading per plate, active fuel width, active fuel length, and U-235 enrichment. Reflective boundary conditions were used on all six surfaces of the model. The results of this evaluation showed the HFBR fuel (one type of allowed MTR fuel) was limiting. The HFBR fuel element exhibited the peak reactivity in the damaged fuel configuration (maximum plate pitch) and features the maximum fissile loading, and was therefore used in the initial NCT and HAC calculations. The staff finds this acceptable.

The applicant performed the following sensitivity studies to ensure they are modeling the most reactive configuration:

(i) Single package bounding fuel movement – the applicant determined that pushing the fuel elements as close together as possible in the horizontal plane increases reactivity (Table 6.10.1-12 of the application).
(ii) Single package manufacturing tolerance – the applicant determined that minimizing the bucket thickness (and thus the neutron absorption in the steel) increases reactivity. Decreasing the thickness of the inner compartment plate, which is located between the buckets, increases reactivity. Conversely, increasing the thickness of the outer plates proves to be more reactive (Table 6.10.1-13 of the application).
(iii) Single package moderator density – the applicant determined that the reactivity decreases significantly as the water density decreases, and is at a maximum with full flooding (Table 6.10.1-14 of the application).
(iv) Single package bounding element characteristics – the applicant determined that reactivity is maximized from (1) minimizing structural material outside of the fueled region (cladding and side plate volume), (2) maximizing active fuel width, (3) minimizing active fuel height, and (4) minimizing fuel thickness (Table 6.10.1-15 of the application).

Staff found the methods used to identify the parameter values that maximize k_{eff} were appropriate and found the set of parameters used in the analysis to be acceptable.

The staff finds that the analysis demonstrated that the applicant found the maximum reactivity for contents in the TN-LC-MTR basket per the requirements of 10 CFR 71.55(b).

6.3.4.2 TN-LC-NRUX Basket

The allowable contents for the TN-LC-NRUX basket consist of either NRU or NRX fuel in a matrix of aluminum and high-enriched uranium at 93 wt.%. NRU and NRX fuel are modeled with U-235 fuel loadings and enrichment (94 wt.%) that bound the known values for these fuel types. An enrichment of 94 wt.% is used in the models which bounds the actual enrichment of 93 wt.%. The applicant performed an evaluation to compare the reactivity of NRU and NRX fuel, allowing pitch contraction and expansion until constrained by the basket tubes. For both NRU and NRX fuel, the reactivity increased as the pitch increased, which simulated a damaged fuel configuration. The NRU fuel exhibited the peak reactivity in the damaged fuel configuration and was therefore used in additional tolerance studies. The staff finds this acceptable.

The applicant then performed analyses to determine the most reactive mechanical configuration of the basket and contents. Fabrication tolerances found to increase reactivity were the following: (1) reducing the tube wall thickness of the tubes that hold the fuel, (2) reducing the thickness of the steel plate between the basket tube assemblies and then minimizing the tube wrap thickness between the bundles, (3) modeling the remaining tube wrap at the maximum thickness, and (4) locating active fuel such that it is touching the bottom end of the cask.

The applicant also determined that the reactivity decreases significantly as the water density decreases, and is at a maximum with full flooding. Other changes had a negligible effect on reactivity.

The staff finds that the analysis demonstrated that the applicant found the maximum reactivity for contents in the TN-LC-NURX basket per the requirements of 10 CFR 71.55(b).

6.3.4.3 TN-LC-TRIGA Basket

The allowable contents for the TN-LC-TRIGA basket consist of seven possible TRIGA fuel elements in a matrix of zirconium hydride and high-enriched uranium (six at 20 wt.% U-235 and one at 70 wt.% U-235). The applicant performed a parametric evaluation to determine the limiting (most reactive) fuel type of the seven types of TRIGA fuels for a single fuel compartment from the TN-LC-TRIGA basket with four TRIGA fuel assemblies at nominal dimensions and enclosed by a layer of stainless steel to simulate the basket. Reflective boundary conditions were used on all six surfaces of the model. Calculations were performed with no moderation and fully flooded. The results of this evaluation showed the 70 wt.% TRIGA fuel was limiting. This type of TRIGA fuel was used as the bounding fuel assembly and was therefore used in additional tolerance studies. The staff finds this acceptable.

The applicant performed studies of four radial configurations of the TRIGA fuel assemblies to ensure they are modeling the most reactive configuration, as follows: (1) Fuel assemblies are evenly spaced so that the distance between adjacent assemblies and the distance between an assembly and a wall of the compartment are equal, (2) All fuel assemblies are pushed to the center of each compartment and are touching, (3) All fuel assemblies are pushed to the four corners of each compartment, and (4) All fuel assemblies are touching and are pushed as close to the center of the basket as possible.

All four configurations assume that the basket is fully flooded and have poison plates consisting of a loading of 5 mg/cm^2 of B-10. The applicant determined that fuel assemblies located at the four corners of each compartment increases reactivity (Table 6.10.3-15 of the application). The applicant also determined that the reactivity decreases significantly as the water density decreases, and is at a maximum with full flooding. Other changes had a negligible effect on reactivity.

In addition, the applicant discussed the reactivity effects due to the tolerances of the neutron absorber panels. They state that the effects are negligible and within the statistical uncertainty of the calculations. The staff finds this acceptable.

Structural analysis of the cask and basket in Chapter 2 has demonstrated that they maintain their structural integrity after an accident. However, to conservatively bound damage in the fuel, the applicant modeled the TRIGA fuel assemblies assuming total axial collapse of the plenums. As a result, the fuel assemblies were modeled without the end fittings and cladding beyond the active fuel region. The fuel was then moved axially so that the fuel assemblies in adjacent baskets are touching. The fuel could not achieve this configuration without damage to the end regions. Therefore, this configuration can be considered to be a bounding geometry for any credible scenario under HAC in which the fuel is damaged.

Staff found the methods used to identify the parameter values that maximize k_{eff} to be appropriate and found the set of parameters used in the analysis to be acceptable.

The staff finds that the analysis demonstrated that the applicant found the maximum reactivity for contents in the TN-LC-TRIGA basket per the requirements of 10 CFR 71.55(b).

6.3.4.4 TN-LC-1FA Basket

The allowable contents for the TN-LC-1FA basket consist of one intact PWR or BWR fuel assembly, or up to 25 intact individual PWR, BWR, EPR or MOX fuel pins. The applicant performed an evaluation to determine the limiting (most reactive) fuel type of the PWR and BWR fuel assemblies shown in Tables 6.10.4-2 and 6.10.4-3, respectively, of the application. The TN-LC-1FA basket and other components of the package were modeled with nominal dimensions. Calculations were performed with the package fully flooded, and full water reflection was assumed outside the package. The PWR fuel assemblies were evaluated at their nominal pitch and the maximum pitch possible before the fuel rods contact the compartment inner width (simulating fuel damage from pitch expansion). This was done to determine whether the most reactive fuel in NCT is different from the most reactive fuel in HAC. For BWR fuel assemblies, the most reactive fuel for each fuel category was obtained at nominal pitch. The most reactive PWR fuel under NCT is the BW 15x15 B11 (Case ID: P_A010), and for HAC, it is the WE 14x14 Std/LOPAR/ZCA/ZCB (Case ID: P_A045). These two PWR fuels were used as the bounding fuel assemblies for additional tolerance and sensitivity studies. The staff finds this acceptable.

For BWR fuel assemblies, the most reactive fuel was obtained at nominal pitch. Based on similar conditions used for the PWR evaluation, the most reactive BWR fuel was the Allis Chalmers – LaCrosse fuel, as shown in Table 6.10.4-16. The results for BWR fuels were significantly less than those for PWR fuels and are not discussed further in this SER.

Only the rod pitch analysis is performed for BWR fuels since it is demonstrated that the most reactive fuel is the PWR, and any BWR damaged fuel will be bounded by that of the PWR fuel.

The applicant performed the following sensitivity studies to ensure that the most reactive configuration is being modeled:

> Single package bounding fuel movement – the applicant determined the effect of varying the fuel assembly position within the compartment. The three positions compared were the center, bottom-center, and bottom left-corner positions. The center case was most reactive and was chosen as the standard most reactive position for NCT and HAC.
>
> Single package manufacturing tolerance – the applicant determined the effects of varying the fuel compartment thickness and reducing the poison plate to its minimum thickness tolerance. The results, presented in Table 6.10.4-11 show that the nominal compartment thickness and a poison plate thickness of 0.20 inch results in the most reactive configuration (Case ID: P_D001). The B-10 loading is held constant during this analysis, i.e., the 15 mg B-10/cm^2 is modeled in each case. The result in Case P_D001 also represents the most reactive PWR fuel under NCT.

Single package moderator density – the applicant determined that the reactivity decreases significantly as the water density decreases, and is at a maximum with full flooding under both NCT and HAC scenarios.

Single package damaged fuel scenarios – the applicant performed evaluations with several different fuel configurations intended to bound the range of possible fuel initial conditions or reconfiguration under HAC (single-ended rod shear, double-ended rod shear, and rod pitch variation). For single-ended and double-ended rod shear analyses, it was shown that the BW 15x15 B11 fuel assembly results in the most reactive configuration. However, the WE 14x14 Std/LOPAR/ ZCA/ZCB fuel assembly remains the most reactive for damaged fuel cases when the fuel is at its maximum pitch. The applicant also evaluated these damaged fuel configurations for the reactivity effect of rod removal and loss of cladding. The results demonstrate that rod removal and loss of cladding have an insignificant effect on the reactivity from damage models consisting of single-ended rod shear, double-ended rod shear, and rod pitch variation.

In addition, the applicant discussed the reactivity effects due to the tolerances of the neutron absorber panels. They state that the effects are negligible and within the statistical uncertainty of the calculations. The staff finds this acceptable.

The PWR fuel assembly results from the above analyses exceeded the USL, both for NCT and HAC. Therefore, the PWR fuel assemblies must be poisoned with PRAs that are inserted into the fuel assembly. The BWR fuel assemblies do not require PRAs. Requirements for the location and number of PRAs for PWR fuel assemblies are discussed in Section 6.4 of this SER.

For the 25 Pin Can, the applicant modeled fuel pins from the most reactive PWR and BWR fuels, as well as fuel parameters for MOX, EPR, and generic UO2 fuel obtained from Tables 6.10.4-4 and 6.10.4-5 of the application. The results show that the highest reactivity is obtained while transporting 25 MOX fuel pins, and this reactivity is bounded by the reactivity obtained with PWR and BWR fuel assemblies.

Staff found the methods used to identify the parameter values that maximize k_{eff} were appropriate and found the set of parameters used in the analysis to be acceptable.

The staff finds that the analysis demonstrated that the applicant found the maximum reactivity for contents in the TN-LC-1FA basket per the requirements of 10 CFR 71.55(b).

6.3.5 Confirmatory Analyses

Staff performed confirmatory analyses on the most reactive configurations described by the applicant. The SCALE 6 computer software package was used as an alternate independent code to the MCNP code used by the applicant for the analyses of the TN-LC-MTR, TN-LC-NURX, and TN-LC-TRIGA baskets. Staff calculations were performed with the CSAS26 and CSAS25 criticality sequences of the SCALE 6 suite of codes. Staff used the 238-group and 44-group cross section libraries derived from ENDF/B-V data, as appropriate. Significant parameters were varied to ensure maximum reactivity peaks were adequately captured and in all instances staff calculations were bounded by or in close agreement with the applicant's results.

The staff independently modeled the package using the engineering drawings and information presented in Chapter 6 of the application. Staff reviews of the criticality analysis confirm that the most reactive conditions were properly identified and that k_{eff} for these conditions meet the subcriticality requirements of 10 CFR Part 71. Overall, the staff's confirmatory analyses showed acceptable agreement with the applicant's results and support the conclusion that there is reasonable assurance the package will remain subcritical under NCT and HAC.

6.4 Single Package Evaluation

In the single package evaluation, fuel is modeled in the most reactive damaged condition with water present at the density at which the reactivity is maximized.

6.4.1 Configuration

The staff verified that the applicant's evaluation demonstrates that a single package is subcritical under both NCT and HAC. The applicant floods the inside of the package with water when performing calculations for the damaged condition, and evaluates a single flooded package for the TN-LC-NRUX, TN-LC-TRIGA, and TN-LC-1FA baskets. The applicant did not perform an explicit single package evaluation for the TN-LC-MTR basket, as the NCT array cases bound the single package cases, because the same fuel geometry and moderation assumptions are utilized in both analyses. Additionally, the applicant evaluated moderator variation external to the package in an array up to and including full water reflection, which is expected to approximate the effect of having a single, fully reflected package.

The PWR fuel assembly models with the conservatisms from Section 6.10.4.3.1 of the application exceed the USL, both for NCT and HAC. Therefore, the PWR fuel assemblies must be poisoned with PRAs that are inserted into the guide tubes of the fuel assembly. The BWR fuel assemblies will remain under the USL without any PRAs. The PRA requirements under all conditions of transport are summarized in Table 6.10.4-26 of the application. The applicant modeled the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents. This is discussed in Sections 6.1.1, 6.2, and 6.3.4 of this SER.

6.4.2 Results

The PRA configuration for the PWR assembly classes that require 5 PRAs is shown in Figure 6.10.4-12 of the application. For the WE 14x14 and WE 15x15 assembly classes, configuration 1 or configuration 2, shown in Figure 6.10.4-13, is used. For the B&W 15x15, B&W 17x17, and WE 17x17 assembly classes, configuration 3 or configuration 4 of Figure 6.10.4-15 is used.

The PRA requirement under all conditions of transport is summarized in Table 6.10.4-26 of the application. This table contains the number of PRAs necessary for each assembly class, maximum enrichment allowed, the linear density of each PRA (before the 75% credit is applied for analysis or the actual minimum 40% TD required), and the minimum diameter of each PRA.

The staff confirmed that the results of the applicant's criticality calculations for each of the baskets are consistent with the information presented in the summary table discussed in Section 6.1.3 of this SER. Because values of k_{eff} are less than the USL under the conditions specified in 10 CFR 71.71 and 71.73, the staff verified that this meets the requirements of 10 CFR 71.55(d)(1) and 71.55(e) respectively which requires that the contents be subcritical.

The applicant modeled the fissile material rearranging into the most reactive credible configuration in compliance with 10 CFR 71.55(e)(1).

The staff did not verify that there would be no leakage of water into the containment system per 10 CFR 71.55(d)(3) because the applicant assumes full in-leakage of water at its most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents for NCT and HAC. The staff finds that this meets 10 CFR 71.55(d)(3) and 10 CFR 71.55(e)(2). The applicant also performed calculations with the package fully reflected by water on all sides. The staff finds that this meets 10 CFR 71.55(e)(3).

The staff verified that (1) the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents, (2) water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents, and (3) there is full reflection by water on all sides consistent with the damaged condition of the package. The staff finds that the requirements of 10 CFR 71.55(e)(1) through (3) are met.

For the NCT tests specified in 10 CFR 71.71, the staff verified that there will be no substantial reduction in the effectiveness of the packaging for criticality prevention, including (1) the total effective volume of the packaging on which nuclear safety is assessed will not be reduced by more than 5%, (2) the effective spacing between the fissile contents and the outer surface of the package is not reduced by more than 5%, and (3) there is no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm cube. The staff finds that the applicant meets the requirements in 10 CFR 71.55(d)(4).

6.5 Evaluation of Package Arrays Under NCT

6.5.1 Configuration

The applicant specified a CSI of 100; therefore, calculations are performed using an array of three packages for NCT (N=0.5 and 5N=2.5 in NCT array cases per 10 CFR 71.59(a)(1), thus 3 packages are modeled). The packages were modeled in a triangular array and reflected with at least 12 inches of water. The three packages of the TN-LC-MTR array analyses are separated by 36 inches, which is the minimum length permitted by the impact limiters. The impact limiters were neglected in the analyses of the TN-LC-MTR, TN-LC-TRIGA, and TN-LC-1FA baskets, which is conservative, since the impact limiters are not damaged during NCT. Excluding the impact limiters in the models conservatively minimizes the separation between the packages, which acts to increase reactivity by increasing interaction between the packages.

The most reactive single package case is used as the basis for the NCT array model. Therefore, the package is flooded, and the basket tolerances are modeled at the most reactive values. Although there is no fuel damage under NCT, the fuel damage assumption from the single package models (e.g., maximum pitch expansion) is also conservatively used.

The applicant modeled the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents. This is discussed in Sections 6.1.1, 6.2, and 6.3.4 of this SER.

6.5.2 Results

The applicant evaluated package arrays under NCT using the combined worst case contents configuration and optimal internal moderation from the single package evaluation. For the TN-LC-MTR, TN-LC-NURX, and TN-LC-TRIGA baskets, damaged fuel was conservatively assumed in the NCT calculations. The water density was varied from 0 to 1.0 g/cm3 between the packages to determine the most reactive configuration. The criticality analysis showed that a triangular array of three packages will remain subcritical with close full-water reflection and optimum interspersed hydrogenous moderation. The maximum reactivity occurs with no water between the packages.

The applicant also used the NCT array analysis of the TN-LC-MTR basket to determine the range of allowable configurations that bound the fuel elements described in Table 6.10.1-2 and Table 6.10.1-3 of the application. For a given number of fuel plates per element, the maximum fissile loading (expressed as grams of U-235 per plate) for an element is shown in Table 6.10.1-4 of the application.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the package meets the regulatory requirements of 10 CFR 71.59(a)(1) by demonstrating that an array of at least 5N packages (triangular array of three packages in this case) with nothing between the packages is subcritical.

6.6 Evaluation of Package Arrays Under HAC

Because of the CSI value of 100, no HAC array cases were performed, i.e., 2N=1 per 10 CFR 71.59(a)(2).

6.7 Fissile Material Packages for Air Transport

Air transport was not sought by the applicant and is not authorized.

6.8 Benchmark Evaluations

A separate benchmark evaluation was performed for each basket analysis. The USL for each of the four analyses is as follows:

TN-LC-MTR USL = 0.9213
TN-LC-NRUX USL = 0.9227
TN-LC-TRIGA USL = 0.9301
TN-LC-1FA USL = 0.9420

The USL and bias were determined for each basket using USLSTATS for several trending parameters. The USLSTATS tool in SCALE 6 provides trending analysis for bias assessment. However, because the number of experiments for MTR and TRIGA analyses is less than the 25 required for USLSTATS to accurately be used, the normality test for these analyses was performed using the Anderson-Darling normality test, and the USL was recalculated using a method developed for small sample statistics. In all cases, USLSTATS provided a conservative USL over the one calculated from Anderson-Darling.

As stated earlier, the package is considered to be subcritical if k_{safe} (k_s) is less than the USL. The computed k_{safe} is equated as $k_s = k_{eff} + 2\sigma < USL$.

Staff reviewed the benchmark evaluations and bias determination performed by the applicant and found them to be applicable for each of the baskets. In all instances an administrative margin of 0.05 was used. Additionally, the benchmark evaluations are nearly identical to that previously accepted for other spent fuel transportation package designs.

6.8.1 TN-LC-MTR Basket

6.8.1.1 Experiments and Applicability

The applicant used the MCNP5 v1.40 code with the cross section data from the ENDF/B-V library. This code is widely used in industry applications for criticality calculations and has therefore been extensively benchmarked against critical experiments.

The applicant performed benchmark comparisons on 35 selected critical experiments that were chosen to bound the variables in the TN-LC package design for MTR fuel. The important selection parameters were high-enriched uranium plate-type fuel with a thermal spectrum. The benchmark parameters bounded the parameters in the analysis with respect to (1) energy of the average neutron lethargy causing fission (EALF), (2) U-235 number density, (3) channel width, (4) hydrogen to U-235 atom ratio (H/U-235), and (5) plate pitch.

Of these 35 benchmarks, 17 benchmarks were directly applicable, while 18 benchmarks were applicable to a lesser degree. To compensate for the benchmarks that were not directly applicable, trending was performed both on all 35 benchmark experiments and on the subset of 17 directly applicable benchmark experiments. The USL selected was the minimum of both experimental sets.

The staff reviewed the benchmark comparisons and agrees that the Monte Carlo computer program MCNP5 v1.40 used for the analysis was adequately benchmarked to representative critical experiments relevant to the package design and contents specified.

6.8.1.2 Bias Determination

The applicant used the method recommended in NUREG/CR-6361 to calculate a USL. Separate USL ranges were calculated for the five parameters of (1) EALF, (2) U-235 number density, (3) channel width, (4) H/U-235, and (5) plate pitch, which matched the values of the fuel assemblies to be transported in the TN-LC. All parameters were evaluated for trends and to determine the minimum USL for each parameter using the full benchmark set of 35 experiments and the subset of directly applicable benchmark results. The applicant then selected the minimum value among these five parameters, which resulted in an overall USL of 0.9213. The criticality evaluation used the same cross section set, fuel materials, and similar material/geometry options that were used in the benchmark calculations.

6.8.2 TN-LC-NRUX Basket

6.8.2.1 Experiments and Applicability

The applicant used the MCNP5 v1.40 code with the cross section data from the ENDF/B-V library. The applicant performed benchmark comparisons on 64 selected critical experiments that were chosen to bound the variables in the TN-LC package design for NRU and NRX fuel. The important selection parameters were high-enriched uranium fuel with a uranium-aluminide fuel matrix, aluminum cladding, and a thermal spectrum. The benchmark parameters bounded the parameters in the analysis with respect to (1) energy of the average neutron lethargy causing fission (EALF), (2) hydrogen to U-235 atom ratio (H/U-235), and (3) moderator to fuel ratio (VM/VF).

Of these 64 benchmarks, 29 benchmarks had cylindrical fuel elements, while 35 benchmarks had flat fuel elements. Trending was performed on all 64 benchmark experiments and on the subsets of 29 benchmark experiments with cylindrical fuel elements and 35 benchmark experiments with flat fuel elements. The USL selected was the minimum of these experimental sets.

The staff reviewed the benchmark comparisons and agrees that the Monte Carlo computer program MCNP5 v1.40 used for the analysis was adequately benchmarked to representative critical experiments relevant to the package design and contents specified.

6.8.2.2 Bias Determination

The applicant used the method recommended in NUREG/CR-6361 to calculate a USL. Separate USL ranges were calculated for the three parameters of (1) EALF, (2) H/U-235, and (3) VM/VF, which matched the values of the fuel assemblies to be transported in the TN-LC. All parameters were evaluated for trends and to determine the minimum USL for each parameter using the full benchmark set of 64 benchmark experiments and on the subsets of 29 benchmark experiments with cylindrical fuel elements and 35 benchmark experiments with flat fuel elements. The applicant then selected the minimum value among these three parameters, which resulted in an overall USL of 0.9227. The criticality evaluation used the same cross section set, fuel materials, and similar material/geometry options that were used in the benchmark calculations.

6.8.3 TN-LC-TRIGA Basket

6.8.3.1 Experiments and Applicability

The applicant is using the MCNP5 v1.40 code with the cross section data from the ENDF/B-V library. The applicant performed benchmark comparisons on 21 selected critical experiments that were chosen to bound the variables in the TN-LC package design for TRIGA fuel. The important selection parameters were high-enriched zirconium hydride fuel with a thermal spectrum. The benchmark parameters bounded the parameters in the analysis with respect to (1) energy of the average neutron lethargy causing fission (EALF), (2) U-235 number density, and (3) hydrogen to U-235 atom ratio (H/U-235).

Of these 21 benchmarks, only two were directly applicable. Because a sample set of two benchmarks is not of sufficient size to obtain a statistical distribution, additional benchmarks were selected to supplement the two available TRIGA benchmarks: a set of 10 highly enriched (93%) uranium solution benchmarks and a set of 9 low-enriched (10%) uranium solution benchmarks. These benchmarks were chosen to simulate fuel intimately mixed with moderator, since the zirconium hydride fuel in the TRIGA assemblies contains moderator embedded in the fuel matrix. The 21 benchmark experiments were divided into three groups for computing trends:

- A group consisting of all 21 benchmarks
- A group consisting of the 10 HEU benchmarks and the 2 TRIGA benchmarks

- A group consisting of the 9 LEU benchmarks and the 2 TRIGA benchmarks

The USL selected was the minimum of all three benchmark sets.

The staff reviewed the benchmark comparisons and agrees that the Monte Carlo computer program MCNP5 v1.40 used for the analysis was adequately benchmarked to representative critical experiments relevant to the package design and contents specified.

6.8.3.2 Bias Determination

The applicant used the method recommended in NUREG/CR-6361 to calculate a USL. Separate USL ranges were calculated for the three parameters of (1) EALF, (2) U-235 number density, and (3) H/U-235, which matched the values of the fuel assemblies to be transported in the TN-LC. All parameters were evaluated for trends and to determine the minimum USL for each parameter using (1) the full benchmark set of all 21 benchmarks, (2) a subset of the 10 HEU benchmarks and two TRIGA benchmarks, and (3) a subset of the 9 LEU benchmarks and two TRIGA benchmarks. The applicant then selected the minimum value among these three parameters, which resulted in an overall USL of 0.9301. The criticality evaluation used the same cross section set, fuel materials, and similar material/geometry options that were used in the benchmark calculations.

6.8.4 TN-LC-1FA Basket

6.8.4.1 Experiments and Applicability

The applicant used the SCALE 6 code with the cross section data from the ENDF/B-V library. The applicant performed benchmark comparisons on 150 selected critical experiments that were chosen to bound the variables in the TN-LC package design for LWR fuel. The important selection parameters were LWR or MOX fuel with a thermal spectrum. The benchmark parameters bounded the parameters in the analysis with respect to (1) energy of the average neutron lethargy causing fission (EALF), (2) U-235 enrichment, (3) H_2O/UO_2 volume ratio, (4) fuel rod pitch and (5) PUO_2 content.

The criticality benchmark analysis uses 118 LWR critical experiments and 32 MOX experiments. All 32 MOX experiments were selected for developing the MOX USL functions. The USL selected was the minimum of these experimental sets.

The staff reviewed the benchmark comparisons and agrees that the Monte Carlo computer program MCNP5 v1.40 used for the analysis was adequately benchmarked to representative critical experiments relevant to the package design and contents specified.

6.8.4.2 Bias Determination

The applicant used the method recommended in NUREG/CR-6361 to calculate a USL. Separate USL ranges were calculated for the five parameters of (1) EALF, (2) U-235 enrichment, (3) H_2O/UO_2 volume ratio, (4) fuel rod pitch and (5) PUO_2 content, which matched the values of the fuel assemblies to be transported in the TN-LC. All parameters were evaluated for trends and to determine the minimum USL for each parameter using the full benchmark set of 150 benchmark experiments and on the subset of 32 MOX benchmarks. The applicant then selected the minimum value among these five parameters, which resulted in an overall USL of 0.9420. The criticality evaluation used the same cross section set, fuel materials, and similar material/geometry options that were used in the benchmark calculations.

6.9 Burnup Credit

The applicant does not request credit for burnup. All analyses are modeled using fresh, un-irradiated fuel.

6.10 Evaluation Findings

Based on the review of the statements and representations in the application, supplemental information supplied by the applicant, and staff confirmatory analyses, the staff has reasonable assurance that the nuclear criticality safety design has been adequately described and evaluated by the applicant and that the package meets the criticality safety requirements of 10 CFR Part 71.

7.0 PACKAGE OPERATIONS

Chapter 7.0 of the application provides a description of package operations, including package loading and unloading operations, and preparation of an empty package for shipment. Loading and unloading procedures described in this chapter only show a general approach to perform operational activities because site-specific conditions may require the use of different equipment and loading or unloading steps.

7.1 Package Loading

Package loading includes (1) preparation of the empty package for use, (2) verification that the fuel assemblies to be loaded in the appropriate fuel-specific basket meet all of the criteria set in the CoC, (3) installation of a basket into the package, and (4) loading fuel, or placing already loaded fuel baskets or pin cans in the package with the appropriate fuel-specific basket.

Procedures for the preparation of an empty package for use are identical whether the package is loaded wet or dry. In particular, the cask cavity atmosphere is sampled prior to removing the lid, contamination smears are taken on the outside surface of the package and the package is decontaminated, if necessary, and O-ring seals are discarded after each use. Precautionary measures shall be taken to minimize doses to operating personnel during underwater loading of the package because radioactive particulate matter may float to the surface of the water. Depending on the basket design being loaded, a spacer of appropriate height shall be placed at the bottom of the cavity of the package and/or bolted to the underside of the lid. If a BWR fuel assembly is loaded in a TN-LC-1FA basket, a BWR sleeve shall be placed with a hold down ring in the basket; if fuel rods are loaded in the TN-LC-1FA basket, a 25 pin can shall be placed inside the BWR sleeve with a hold down ring in the basket. A radiological survey is performed on the package as it is raised from the pool (wet loading) or disengaged from the hot cell portal (dry loading).

After being loaded, the package is placed horizontally onto the transportation skid. A verification that the surface removable contamination levels meet the requirements of 10 CFR 71.87 and 49 CFR 173.443 is performed and the assembly verification leakage testing is then completed prior to place the package onto the transportation conveyance.

7.2 Package Unloading

Package unloading includes performing a receipt inspection, removing the tamper-indicating seals, sampling the package cavity atmosphere, taking contamination smears on the outside surfaces of the package, rotating the package to a vertical orientation, lifting it for transfer to the fuel pool or the hot cell interface.

Procedures for dry unloading of the package to a hot cell are highly dependent on the hot cell design. However, in any case, the package cavity is vented to the site radwaste system for sampling prior to fuel unloading. Note that a 25 pin can in a 1FA basket is unloaded horizontally.

Unloading of the fuel in a fuel pool also requires venting of the package cavity to the site radwaste system, and flushing the cavity if necessary, filling the cask with clean water, slowly lowering the package into the pool until the lid is just above the water surface. As for wet loading operations, precautionary measures shall be considered to minimize radiation dose to personnel during these operations because radioactive particulate matter may float to the surface of the water. When the package is filled with water, the vent and supply lines are removed, the lid is lifted from the package and fuel specific unloading procedures are followed.

7.3 Preparation of an Empty Packaging for Shipment

Preparation of an empty package for transport shall be done in accordance with the requirements of 49 CFR 173.427. The applicant has detailed the procedure for leakage testing of the package containment boundary prior to shipment in Section 7.4.1 of the application. If the leakage rate is greater than 10^{-7} ref.cm^3/s, the vent port seal shall be replaced and retested.

7.4 Evaluation Findings

The staff reviewed the Operating Procedures in Chapter 7 of the application to verify that the package will be operated in a manner that is consistent with its design evaluation. The staff also reviewed the vent port plug seal leakage testing procedures, the lid O-ring leakage testing procedures, the drain port plug seal and the bottom plug O-ring leakage testing procedures. On the basis of its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures provide adequate measures and reasonable assurance for safe operation of the package in accordance with 10 CFR Part 71.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the application identifies the inspections, acceptance tests, and maintenance programs to be conducted on the package and verifies their compliance with the requirements of 10 CFR Part 71.

8.1 Acceptance Tests

A pressure test will be performed on the package at a pressure between 45 and 50 psig which is well above 1.5 times the MNOP of 16.0 psig, in compliance with 10 CFR 71.85(b). The test pressure is held for a minimum of ten minutes, in accordance with ASME B&PV Code, Section III, Subsection NB.

Fabrication verification leakage tests include five tests as follows: (i) package leakage integrity, (ii) vent port plug seal integrity, (iii) drain port plug seal integrity, (iv) lid seal integrity, and (v) bottom plug seal integrity. Prior to lead pouring and final machining of the inner shell, the package containment boundary, including the containment boundary base metal and joining welds, is leak-tested in accordance with the requirements of ANSI N14.5, using a temporary closure lid and seal for the bottom plug and lid, to ensure the measured leakage rate is less than $1x10^{-7}$ ref-cm^3/sec.

As part of the acceptance tests for the packaging, each impact limiter container will be pressurized to a pressure between 2.0 and 3.0 psig and the weld seams and penetrations will be tested for leakage using a soap bubble test (if bubbles are detected, the weld will be repaired and the soap bubble test re-performed). However, thermal acceptance testing is not required for the Model No. TN-LC package, due to its large margin of safety: gaps were modeled in the thermal analysis to account for possible gaps expected during fabrication, also assumed to be present during NCT and HAC post-fire cases, and also assumed closed when calculating heat flow into the cask (i.e., during the HAC fire). The calculated cladding temperatures are much lower than the cladding temperature limit, thus assuring a large margin of safety.

The poured lead (for gamma shielding) is inspected via gamma scanning at the intersections of a grid no larger than 6x6 inches on the outside of the shell prior to installation of the neutron shield.

Section 8.1.6.2 of the application provides the acceptance values for density and chemical composition of the neutron shield resins. Staff determined that statements such as “resin composition or density test results which fall outside of this range will be evaluated to ensure that the shielding regulatory dose limits are not exceeded” were not appropriate since the staff could not evaluate neutron shielding materials that do not meet the acceptance criteria for specific tests of the packaging. Such statements were removed from the acceptance tests program. Neutron absorbers shall be 100% visually inspected in accordance with the Certificate Holder’s Quality Assurance procedures and blisters shall be treated as non-conforming.

8.2 Maintenance

Accessible welds will be inspected prior to package loading for signs of degradation and brought into compliance with the licensing drawings before shipment, as stated in Section 8.2.3. All threaded fasteners and port plugs shall be inspected at the time of use for deformed or stripped threads. Damaged parts shall be evaluated for continued use and replaced as required. At a minimum, the lid and bottom plug bolts shall be replaced at least every 75 round trip shipments to ensure adequate fatigue strength is maintained. The elastomeric seals shall be replaced each time the package is used. The staff finds the maintenance procedures for the TN-LC packaging to be consistent with the guidance in NUREG-1617 and therefore acceptable.

As stated in Section 8.2.3.2, before each shipment the impact limiters will be visually examined for degradation. The applicant states that if there is no evidence of weld cracking or other damage that could result in water in-leakage wood performance will be maintained. The staff finds this engineering assumption reasonable. Impact limiters will be leak tested once every five years, using the same acceptance criteria than the original leak tests, to ensure that water has not entered the impact limiter and that the impact limiter has maintained its performance.

If there is evidence of weld cracking or any other damage that could result in water in-leakage, the entire block of wood in the affected damaged segment(s) will be replaced, and the entire impact limiter will be leak-tested using the same acceptance criteria as the original leak tests.

Density and moisture content are tested for each wood species on a minimum of five samples per lot of wood. Compression tests per the applicable sections of ASTM D143 are performed for each wood species on a minimum of five samples per lot of wood.

The staff agrees that no leakage tests are required prior to shipment of an empty packaging, but the applicant should verify the package is empty, and ensure that external and internal contamination levels meet the requirements of 49 CFR 173.443 and 49 CFR 173.428. The staff also agrees that periodic tests for neutron shielding are not necessary.

8.3 Evaluation Findings

The staff reviewed the acceptance tests and maintenance programs and found them acceptable. Based on the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71.

CONDITIONS

The following conditions are included in the Certificate of Compliance:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter No. 7 of the application,

(b) Each packaging must meet the Acceptance Tests and Maintenance Program of Chapter No. 8 of the application,

(c) Transport by air of fissile material is not authorized,

(d) Prior to the first shipment, the package shall be tested for the entire containment boundary, e.g., all base metal, all joining containment welds, vent port plug seal, drain port plug seal, lid seal, and bottom plug seal, in accordance with ANSI N14.5, by helium leakage testing to meet the leaktight criteria of 1.0×10^{-7} ref-cm^3/sec for fabrication leakage tests.

(e) Poison Rod Assemblies, required for shipment of PWR assemblies, shall be installed such that the active fuel length is covered by the absorber and measures shall be taken against their inadvertent removal from the fuel assembly.

CONCLUSION

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff concludes that the Model No. TN-LC 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. 9358, Revision No. 0,
on December 31, 2012.