SAFETY EVALUATION REPORT
Docket No. 71-9374
Model No. HI-STAR 80 Package
Certificate of Compliance No. 9374
Revision No. 0

TABLE OF CONTENTS

SAFETY EVALUATION REPORT
Model No. HI-STAR 80 Package
Certificate of Compliance No. 9374
Revision No. 0

SUMMARY

By application dated August 23, 2016, as supplemented October 20, 2016, June 16, 2017, January 10, March 1, and September 4, 2018, Holtec International requested approval of the Model No. HI-STAR 80 as a Type B(U)F-96 package. Revision No. 3 of the package application, dated September 2018, superseded in its entirety the application dated October 20, 2016.

The Model No. HI-STAR 80 package is designed for exclusive use transport of either moderate to high burnup fuel (burnup exceeding 45 GWd/MTU, up to 70 GWd/MTU) or of non-fuel waste. The authorized PWR spent fuel assemblies are in 15x15 and 17x17 arrays and must be loaded within the F-12P basket model. The authorized BWR spent fuel assemblies, loaded in the F-32 B basket model, are in 8x8, 9x9, 10x10 and 11x11 array sizes, while the MOX assembly type is in a 10x10 array size. Up to 4 MOX assemblies are allowed in the F-32B basket. The Model No. HI-STAR 80 package is also authorized to transport non-fuel waste, which consists mainly of activated or contaminated metals or ceramics, spacer grids, core grid components, control rods or control blades, and burnable absorbers. This non-fuel waste is inserted within the non-fuel waste basket, NFWB-1, which includes also an optional secondary container called the Plant Specific Stainless Steel Core Component Cassette.

The Model No. HI-STAR 80 packaging body is comprised of a nickel steel shell welded to a stainless steel lower forging at the bottom and a stainless steel forging at the top. The closure system consists of two stainless steel closure lids: (i) the inner lid seals against the upper forging flange and a tapered retainer ring connects to the upper forging flange with 36 closure bolts while providing the preload for the inner cask lid, (ii) the outer lid is secured to the upper forging flange with 36 closure bolts.

Radial shielding is provided by lead, steel, copper, and Holtite. Axial shielding is provided by the steel closure lids, and the bottom flange, supplemented by lead and Holtite. The gamma shield consists of lead between the containment shell and an intermediate steel shell supported by four radial ribs welded to both shells. The neutron shield consists of Holtite surrounded by the copper inner and outer shells that are supported by diagonal copper ribs attached to the shells.

The containment system components include: (i) the containment shell, shell cladding, upper and lower forgings; (ii) the inner closure lid including its inner seal, retainer ring, bolts and helical thread inserts, and leak test port plug and seal; (iii) the outer closure lid including its inner seal, bolts, test plug seal and helical thread inserts; (iv) the vent and drain port including its bronze plug, bushing and bushing/plug seal, inner port cover plate, and the port outer containment seal, and (v) the spray cooling port including its cap, cap inner seal, cover plate, and cover plate inner seal, bolts and helical thread inserts.

The Model No. HI-STAR 80 package is designed for maximum heat loads of 50 kW and 54 kW, respectively, depending upon the fuel loaded in the package.

The Model No. HI-STAR 80 package contains a personnel barrier to limit access to the surface areas of the package that have temperatures above 50°C. The personnel barrier is not considered a structural component of the package. Lifting trunnions, housed within the upper and lower forgings, are secured with two bolts (upper trunnions) and two locking rods (lower trunnions). The impact limiters, referred to as “AL-STAR,” are comprised of a rigid steel cylindrical core, a steel cylindrical skirt that surrounds a crushable shock absorbing material, and ductile steel fasteners.

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

NRC staff reviewed the application using the guidance in "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," NUREG-1617, March 2000.

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

References

Holtec International “Safety Analysis Report on the HI-STAR 80 Package,” Holtec Report No. HI-2146261, Revision No. 3, dated September 4, 2018.

1.0 GENERAL INFORMATION

1.1 Packaging

The HI-STAR 80 packaging consists of the following major components: the packaging body, the fuel baskets, the non-fuel waste basket, the impact limiters, and the personal barrier.

The packaging body is comprised of a nickel steel shell welded to a stainless steel lower forging at the bottom and a stainless steel forging at the top. The closure system consists of two stainless steel closure lids: (i) the inner lid seals against the upper forging flange and a tapered retainer ring connects to the upper forging flange with 36 closure bolts while providing the preload for the inner cask lid, and (ii) the outer lid is secured to the upper forging flange with 36 closure bolts. The inner seal of each lid ensures both the containment and moderator exclusion functions of the package.

Radial shielding is provided by lead, steel, copper, and Holtite. Axial shielding is provided by the steel closure lids, and the bottom flange, supplemented by lead and Holtite. The gamma shield consists of lead between the containment shell and an intermediate steel shell supported by four radial ribs welded to both shells. The neutron shield consists of Holtite surrounded by the copper inner and outer shells that are supported by diagonal copper ribs attached to the shells.

The containment system components include:

(i) the containment shell, shell cladding, upper and lower forgings,

(ii) the inner closure lid including its inner seal, retainer ring, bolts and helical thread inserts, and leak test port plug and seal,

(iii) the outer closure lid including its inner seal, bolts, test plug seal and helical thread inserts,

(iv) the vent and drain port including its bronze plug, bushing and bushing/plug seal, inner port cover plate, and the port outer containment seal, and

(v) the spray cooling port including its cap, cap inner seal, cover plate, and cover plate inner seal, bolts and helical thread inserts.

Fuel baskets are formed by a honeycomb structure of Metamic-HT plates, surrounded by an array of shaped aluminum spacers (basket shims) in the packaging cavity peripheral spaces. The aluminum basket shims are attached to the basket.

The HI-STAR 80 package includes two fuel basket designs:

(i) A 12-cell basket (F-12P) with flux traps for transporting undamaged spent PWR UO_2 fuel assemblies with a maximum enrichment of 5 wt% U-235, and

(ii) A 32-cell basket (F-32B) for transporting undamaged spent BWR UO_2 fuel assemblies with a maximum planar average enrichment of 5 wt% or undamaged spent BWR MOX fuel assemblies, with specific composition as defined in Table 7.D.1 of the application, or a mixed load of spent BWR UO_2 and BWR MOX fuel assemblies in the same basket. Hybrid MOX fuel assemblies, i.e., BWR fuel assemblies that contain both MOX fuel rods and UO_2 fuel rods, are not authorized contents.

The HI-STAR 80 also has a non-fuel waste basket, or NFWB-1, with an optional secondary container called the Plant Specific Stainless Steel Core Component Cassette, or CCC-1, that can be used with the NFWB-1. These non-fuel waste components consist of mainly activated or contaminated metals or ceramics. Table 7.D.8 authorizes the following reactor-related non-fuel Waste (“Core Components”) to be shipped within the NFWB-1: fuel channels, transition pieces, spacer grids, core grid components, core spray components, control rods or control blades, LPRM neutron monitors using fission chambers, and burnable absorbers.

The criticality safety index (CSI) of the package is 0.0, as an unlimited number of packages will remain subcritical under the procedures specified in 10 CFR 71.59(a).

The AL-STAR 80 impact limiters include crushable material enclosed in a stainless steel cup-shelled shell fitting the outside of the packaging forgings (top and bottom) with a small radial clearance.

The Model No. HI-STAR 80 package must be transported by exclusive use, with the personal barrier installed, because the temperatures of the accessible surfaces of the package exceed 50°C, while being maintained at less than 85°C in accordance with 10 CFR 71.43(g). The personal barrier is not a structural part of the packaging.

The cavity is 180 ¼ inch long with an internal diameter of 48 7/8 inch. The packaging is 212 inch long (313 inch with the impact limiters installed) and its outside diameter is 89 ¼ inch (the outside enveloping diameter is 107 inch).

The packaging body weighs approximately 157,600 lbs (without the lids). The package, as configured for transport, i.e., including impact limiters, weighs approximately from 199,500 lbs (with the NFWB basket) to 234,800 lbs (with the F-32B basket).

1.2 Contents

The contents for the Model No. HI-STAR 80 package are commercial spent fuel, i.e., fuel with the shortest possible cooling time, MOX fuel, high burnup fuel, and reactor related non-fuel waste. The Model No. HI-STAR 80 package is designed to transport moderate to high burnup fuel (burnup exceeding 45 GWd/MTU up to 70 GWd/MTU) with maximum heat loads of 50 kW and 54 kW, respectively, depending upon the fuel basket (F-12P for 12 PWR fuel assemblies or F-32B for 32 BWR fuel assemblies) loaded in the package.

The applicant specifies the allowable fuel assembly parameters in Appendix 7.D. There is a wide range of allowable spent fuel PWR, BWR and BWR-MOX assemblies, with fuel assembly limits in Table 7.D.1 of the application. Damaged fuel assemblies are not authorized contents for the Model No. HI-STAR 80 package.

The authorized PWR spent fuel assemblies are 15x15 and 17x17 arrays and must be loaded within the F-12P basket model. Control rods are authorized for transport within spent PWR fuel assemblies. Fuel assemblies may contain up to 4 irradiated stainless steel replacement rods. The assemblies are restricted to assembly burnup, enrichment, cooling time and minimum number of 1 year cycle requirements, as specified in Table 7.D.4 of the application.

The F-12P basket allows for loading of 2 fuel types (17x17 and 15x15) with a total of 7 subclass fuel designs. The maximum U-235 enrichment of PWR fuel is 5 wt%. The F-12P basket allows for loading of spent PWR fuel assemblies in all 12 cells of the basket, or only 10 cells with locations 4 and 9 empty, or alternatively 10 cells with locations 5 and 8 empty.

The authorized BWR spent fuel assemblies are UO_2 in 8x8, 9x9, 10x10 and 11x11 array sizes, and a MOX assembly type in a 10x10 array size. The maximum planar average U-235 enrichment of the UO_2 BWR fuel is also 5 wt%, with a maximum rod U-235 enrichment of 6 wt%. Their characteristics are described in Table 7.D.3 of the application. Non-fuel hardware are not authorized contents with spent BWR fuel assemblies. BWR spent fuel assemblies may contain up to 4 irradiated stainless steel replacement rods. The assemblies are restricted to assembly burnup, enrichment, cooling time and minimum number of 1 year irradiation cycle requirements, as specified in Table 7.D.5 of the application.

The F-32B basket allows for loading of 32 undamaged spent BWR UO_2 fuel assemblies or undamaged spent BWR MOX fuel assemblies or a combination of four (4) MOX and 28 UO_2 BWR fuel assemblies. Undamaged spent BWR fuel assemblies can be loaded in all 32 cells of the basket, or 28 cells with locations 13, 14, 19, and 20 empty, or 24 cells with locations 12, 13, 14, 15, 18, 19, 20 and 21 empty, or 24 cells with locations 7, 8, 13, 14, 19, 20, 25 and 26 empty.

The applicant specified the maximum burnup, minimum cooling times and minimum number of 1 year irradiation cycles, as well as the minimum enrichment for the accompanying UO_2 assemblies in Table 7.D.6 of the application. When the package is loaded with mixed UO_2 and MOX BWR fuel assemblies, the MOX fuel assemblies must be loaded in cells 6, 9, 24, and 27 in the F-32B basket, as shown in Figure 7.D.2 in Chapter 7 of the application. No other loading patterns are allowed. The MOX Pu enrichment vector is specified within Table 7.D.1 of the application.

The HI-STAR 80 is also authorized for non-fuel waste for transport within the NFWB-1. These components consist of mainly activated or contaminated metals or ceramics. Table 7.D.8 specifies the following reactor-related non-fuel Waste (“Core Components”) to be shipped within the NFWB-1: fuel channels, transition pieces, spacer grids, core grid components, core spray components, control rods or control blades, LPRM neutron monitors using fission chambers, and burnable absorbers.

1.3 Materials

The materials used in the Model No. HI-STAR 80 package have been previously reviewed by staff for the Model Nos. HI-STAR 100, HI-STAR 60, HI-STAR 180, HI-STAR 180D and HI-STAR 190 packages. The bill of materials adequately defines all construction materials, grades and mechanical properties.

The HI-STAR 80 package uses Metamic-HT fixed neutron poison plate, which is an aluminum alloy containing B_4C as neutron absorber. The applicant states that the HI-STAR 80 is designed to ensure the fixed neutron absorber will remain effective for a period greater than 40 years and that there are no credible means to cause significant loss of ^{10}B in the poison plates during this design basis package life-time. The continued efficacy of the fixed neutron absorber is assured by acceptance testing to validate the ^{10}B concentration in the fixed neutron absorber. The staff confirmed the minimum guaranteed values for Metamic-HT material properties and verified that these properties are consistent with those used in the safety analyses of the HI-STAR 80.

The staff reviewed the material properties and finds that the material properties the applicant used are consistent with the commonly available material data and are conservative. On this basis, the staff determined that the material properties of the packaging materials and the contents are appropriate and acceptable.

1.4 Criticality Safety Index

The CSI for the Model No. HI-STAR 80 package is zero, as an unlimited number of packages will remain subcritical under the procedures specified in 10 CFR 71.59(a).

1.5 Drawings

Sheets 1 through 6 of licensing drawing 9800 show the structural layout and dimensions, manufacturing tolerance, and bill of materials of the HI-STAR 80 overpack that are important to evaluating the criticality safety of the package.

Sheets 1 through 3 of licensing drawing 9796 show the structural layout and dimensions, manufacturing tolerances, and bill of materials of the of the F-12P fuel basket.

Sheets 1 through 3 of licensing drawing 9797 show the structural layout and dimensions, manufacturing tolerances, and bill of materials of the of the F-32B fuel basket.

The packaging is constructed and assembled in accordance with the following drawing Nos.:

(a) HI-STAR 80 Cask Drawing No. 9800, Sheets 1-11, Rev. 7

(b) F-12P Fuel Basket Drawing No. 9796, Sheets 1-4, Rev. 4

(c) F-32B Fuel Basket Drawing No. 9797, Sheets 1-4, Rev. 4

(d) NFWB-1 Non-Fuel Waste Basket Drawing No. 9798, Sheets 1-2, Rev. 5

(e) HI-STAR 80 Impact Limiter Drawing No. 9801, Sheets 1-7, Rev. 4

(f) HI-STAR 80 Transport Package Drawing No. 9795, Sheets 1-7, Rev. 3

1.6 Evaluation Findings

A general description of the Model No. HI-STAR 80 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 package application identifies the Holtec International Quality Assurance Program for the Model No. HI-STAR 80 package 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. HI-STAR 80 package against 10 CFR Part 71 requirements for each technical discipline.

2.0 STRUCTURAL AND MATERIALS REVIEW

2.1 Description of Structural Design

2.1.1 Discussion

The Model No. HI-STAR 80 package includes three main regions: the containment space, the inter-lid space, and the supplemental shielding.

The containment space is comprised of a stainless steel shell welded to a stainless steel lower forging at the bottom and a stainless steel forging at the top. The closure system consists of two stainless steel closure lids. The inner lid is a plug that seals against the upper forging flange. A tapered retainer ring that connects to the upper forging flange with 36 closure bolts provides the preload for the inner cask lid. The outer lid is secured to the upper forging flange with 36 closure bolts. Two concentric grooves in the inner and outer lids accommodate elastomeric seals that provide the containment between the lids and the flange.

According to the applicant, the double lid closure feature helps in establishing the necessary level of confidence to rule out moderator intrusion into the inter-lid space (the small region between the inner and outer lids) under the inner closure lid in the sequence of accidents prescribed in 10 CFR 71.73. The applicant asserts that the gap is sufficiently small enough such that the outer closure lid reinforces the inner closure lid by limiting its deflection, and both lids act in tandem in the event of a hypothetical drop accident.

The gamma shield consists of lead between the containment shell and an intermediate steel shell supported by four radial ribs welded to both shells. The neutron shield consists of Holtite-B surrounded by copper inner and outer shells that are supported by diagonal copper ribs attached to the shells. According to the applicant, the copper shells and ribs improve the heat dissipation capacity for the package.

Lifting trunnions are housed within the upper and lower forgings and are secured with two securing bolts (upper trunnions) and two locking rods (lower trunnions). Details of the containment space, lid assemblies, shielding and trunnions are in Holtec Drawing No. 9800.

The applicant stated that the function of the fuel basket and the fuel basket support (basket shims) in the transport mode is to maintain the position of the fuel in a sub-critical configuration. The fuel basket is composed of Metamic-HT panels welded together to form the basket. Details of the fuel basket and basket shims are in Holtec Drawings 9796 and 9797.

The impact limiters, referred to as “AL-STAR,” are comprised of a rigid steel cylindrical core, a steel cylindrical skirt that surrounds the crushable shock absorbing material, and ductile steel fasteners. Details of the impact limiters are in Holtec Drawing No. 9801.

The staff has reviewed the package structural design description and concludes that the contents of the application meet the requirements of 10 CFR 71.31(a)(1) and (a)(2), as well as 10 CFR 71.33(a) and (b).

2.1.2 Codes and Standards

Table 2.1.11 of the application lists the applicable codes and standards used for the various components of the Model No. HI-STAR 80 package, i.e., (i) ASME B&PV Code Section III, Division 1, Subsection NB and Appendix F, Edition 2010, for the structural components of the containment boundary, and (ii) NUREG-0612 for the design of the lifting trunnions as a special lifting device for critical loads. The applicant also used American Society of Mechanical Engineers (ASME) boiler & pressure (B&PV) Code Section III, Division 1, Subsection NF, Edition 2010, for structural components that were non-containment related.

The staff reviewed the structural codes and standards used in the package design and finds that they are acceptable because they are consistent with NUREG-1617 and NUREG/CR 3854 and therefore meet the requirements of 10 CFR 71.31(c).

2.1.3 Design Criteria

Section 2.1.2.2 of the application includes the design criteria for the containment system, the fuel basket, the dose blocker and the impact limiters.

For the containment system under NCT, the applicant used the stress intensity limits of ASME B&PV Code, Section III, Division 1, Subsection NB. For HAC, the applicant used the stress intensity limits of ASME B&PV Code, Section III, Division 1, Appendix F. The applicant summarized the design criteria limits for the containment vessel and the closure lid bolts in Tables 2.1.2 and 2.1.3 of the application.

In addition to the stress intensity limits, the applicant also stated that the overpack closure lid seals must remain functional under all events to ensure “leak tightness” of the outer containments system and the containment boundary material must not be susceptible to brittle fracture.

Because the applicant considers the package to be a special lifting device for critical loads, NUREG-0612 was used, in addition to the 10 CFR 71.45(a) regulatory requirement, as the design criteria for the lifting trunnions. 10 CFR 71.45(a) requires the yield strength of the trunnions to be 3 times greater than the calculated stress in the trunnion material. NUREG-0612 requires the yield strength of the trunnion material to be 5 times greater than the calculated stress in the trunnions and the ultimate strength of the material to be 10 times greater than the calculated stress in the trunnions.

For the basket, the applicant established a dimensionless panel deformation limit of

$$\theta = \frac{\delta_{max}}{W} \leq 0.005$$

where δ_{max} the maximum deflection of the panel and W is the nominal panel width. Additionally, the applicant stated that creep deformation must remain negligible, brittle fracture must not occur, tearing mode failure must not occur, the ^{10}B areal density for meeting subcriticality requirements must be assured, the mechanical strength and physical properties under NCT must be maintained, and the physical material properties of the plates must be maintained under neutron and gamma fluence.

For the dose blocker parts, specifically the lead gamma shield and the Holtite-B neutron shield, the applicant stated that the performance criteria is that they remain in place. The applicant did not establish a lead slump limit for the gamma shield. Instead, the applicant used the resulting lead slump from the 9 meter drop as the basis for the shielding calculations to determine the shielding performance.

The applicant stated that the impact limiters are designed to absorb the impact energy through plastic deformation during a drop event and that they should be large enough to prevent "bottoming out." Additionally, the impact limiters must stay attached during all postulated impact events to mitigate the inertial forces and keep the stresses in the containment boundary and other critical features below their respective design limits.

In Section 2.11 of the application, the applicant stated that the design criteria for the cladding strain is 1.7% for Zircaloy cladding under a vertical drop accident condition. The applicant based the Zircaloy strain limit on a study by the Pacific Northwest National Laboratory (Reference 2.11.4 of the application).

The staff reviewed the design criteria for the various components of the Model No. HI-STAR 80 package and determined that they are acceptable because they are consistent with NUREG-1617 and have been previously accepted by the staff (the Zircaloy strain limit was used for the Model No. HI-STAR180D package, Docket 71-9367).

2.1.4 Loading and Load Combinations

The applicant considered five categories of loads for the analysis of the Model No. HI-STAR 80 package. Permanent loads mostly arise from the bolt preload used to maintain the seal on the gasketed joint between the cask lids and the flange. The design condition loads include the maximum normal operating pressure, the design internal pressure, external pressure under normal conditions of transport, accident condition internal pressure and accident condition external pressure. The applicant listed these pressures in Table 2.1.1 of the application.

The handling loads include the dead weight of the various components as well as a 15% dynamic load factor.

> The NCT loads are those specified in the tests of 10 CFR 71.71 and include reduced external pressure, increased external pressure, free drop from a height of 1 foot, normal vibratory loads and normal operating conditions.
>
> The HAC loads are those specified in the tests of 10 CFR 71.73 and include the sequential application of a free drop from a height of 30 feet, a puncture test, an engulfing fire at 1475°F, and total immersion in water to a depth of 50 feet.

Based on the loads considered, the applicant determined two governing load combinations for NCT hot and cold conditions, which are designated N1 and N2. Load combination N1 includes bolt preload, design internal pressure and normal operating temperature. Load combination N2 includes the free drop from a height of 1 foot, bolt preload and maximum normal operating pressure (MNOP).

For HAC conditions, the applicant applied the above HAC loads sequentially, as required by 10 CFR 71.73.

The staff reviewed the loads and load combinations and finds them acceptable because they are consistent with Regulatory Guide 7.8, Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material. Additionally, the staff accepts the load combinations N1 and N2 as the governing load combinations for the HI-STAR 80. This is consistent with the governing load combinations for the HI-STAR 180D, which is similar to the HI-STAR 80.

2.1.5 Weights and Centers of Gravity

Table 2.1.9 of the application lists the weights of the various components of the package, and Table 2.1.10 lists the location of the centers of gravity for the package both with and without fuel.

2.1.6 Analytical Approach

The licensing basis for the Model No. HI-STAR 80 package structural performance is predicated on successful analytical modeling rather than on experimental testing. The applicant used a combination of hand calculations and finite element analyses, using LS-DYNA, to determine the adequacy of the structural design.

The applicant used a half-symmetry LS-DYNA model to represent the HI-STAR 80 package, and modeled the containment components using 6-node and 8-node solid elements. The number of layers of the elements are sufficient to adequately capture primary membrane and bending stresses as well as secondary stresses at locations of structural discontinuity. Where shell elements were used in other components of the HI-STAR 80, the applicant chose 10 integration points through the thickness of the element, which is the maximum number possible for LS-DYNA, and is sufficient to ensure convergence of the solution. Each closure lid bolt was explicitly modeled with solid elements and the gasket bearing and metal-to-metal contact interfaces were discretized in sufficient detail to capture the effects of deflection and rotations. The seals were modeled with linear-elastic solid elements to capture seal unloading.

The applicant utilized nonlinear elastic-plastic true stress-strain relationships for the key structural member materials. These relationships were developed in Appendix B of Holtec Report No. HI-2167023, Revision 0.

To qualify the containment boundary for NCT, the applicant developed a static axi-symmetric finite element model using the finite element analysis software ANSYS. The applicant used shell elements to model the through-thickness behavior of the containment shell and baseplate.

In previous transportation packages, the applicant used a two phase approach for the analysis of the dynamic drop events that involved determining the deceleration force of the package using LS-DYNA, then using that deceleration force as a static load to determine component stresses using ANSYS finite element analysis software. However, for the HI-STAR 80, the applicant proposed determining the stresses in critical cask components from LS-DYNA directly.

The applicant used this same approach for the Model No. HI-STAR 190 transportation package (Docket No. 71-9373), and presented the following benchmarking/validation material as evidence of the ability to obtain accurate simulation results of the HI-STAR 190 for impact events with LS-DYNA:

- Holtec Report No. HI-2156765, Revision 0, “Benchmark LS-DYNA for the Free Drops Involving Steel Casks Without Impact Limiters.”

- LS-DYNA was used by the applicant to predict the structural response of the HI-STORM FW dry storage cask, including stresses and strains, for non-mechanistic tip over in the HI-STORM FW FSAR (Docket No. 72-1032).

- Comparison of numerical results from the two-phase approach using LS-DYNA and ANSYS to LS-DYNA directly for the HI-STAR 60, HI-STAR 180 and HI-STAR 180D.

- Adherence to guide lines established by the ASME Section III, Division 1 Special Working Group on Computational Modeling for Explicit Dynamics (Use of Explicit Finite Element Analysis for the Evaluation of Nuclear Transport and Storage Packages in Energy-Limited Impact Events – Draft Guidance Document)

An impact event excites three dynamic responses of the package: the wave, the vibration and the quasi-static-deformation responses. All of these responses contribute to the overall response of the package and the resultant deformation. Because the HI-STAR 80 has impact limiters, the quasi-static response will dominate the deformation of the package.

The applicant's predicted deformation results of the DOE multi-canister overpack (MCO) using LS-DYNA were consistent with the actual test data (deformation measurements), as indicated in Holtec Report No. HI-2156765, Revision 0. The MCO package did not have impact limiters, but did have impact limiting qualities that served to reduce the effects of the wave and vibration responses. While the staff does not consider the comparison of the results of LS-DYNA to other FEA software to constitute benchmarking per se, it can be considered in the aggregate with other benchmarking activities.

In the analysis of previous transportation packages, the applicant used LS-DYNA to characterize the quasi-static response of the package to the impact and fed that response into ANSYS for further analysis. For the Model Nos. HI-STAR 80 and HI-STAR 190 packages, the applicant still characterized the quasi-static response of the package to the impact using LS-DYNA, but instead of using ANSYS, the applicant continued the analysis in LS-DYNA.

The staff considers LS-DYNA to be a well benchmarked finite element software package, capable of directly providing stress/strain results for impact analysis, provided a quality model is used to predict package behavior under an energy-limited impact event. The applicant demonstrated the ability to model the package behavior, similar to the HI-STAR 80, and validated their ability using physical drop test data.

Additionally, the staff determined that the applicant's LS-DYNA model is consistent with the guidance established by the ASME Section III, Division 1 Special Working Group on Computational Modeling for Explicit Dynamics, and is therefore a quality model. Because of these considerations, the staff determines that the applicant's analytical approach to impact analysis using LS-DYNA is acceptable, and therefore satisfies the requirements of 10 CFR 41(a).

Throughout the application, the applicant compares the calculated stresses in the component material with the allowable stress by calculating the Factor of Safety (FS) as shown below:

$$FS = \frac{Allowable\ Stress}{Calculated\ Stress}$$

If the FS is greater than 1.0, this indicates that the calculated stresses are less than the allowable stresses, and the structural performance of the component is adequate for that particular loading case. Conversely, if the FS is less than 1.0, the calculated stresses in the component are greater than allowed by the applicant's chosen design criteria. Based on the component, and the value of FS, this may still be acceptable, but the applicant must provide an explanation as to why this is acceptable. If the staff determines that the component can still perform the necessary function with a reasonable level of safety, then the staff may accept the deviation.

2.2 Material Properties

The staff evaluated:

(i) Materials selection, including applicable codes and standards, materials properties, weld design and specifications, bolt application, coatings, gamma and neutron shielding materials, neutron absorbing/poison materials for criticality control, and seals,

(ii) Chemical and galvanic reactions including loss of corrosion resistance and flammable gas generation, and

(iii) Cladding integrity, including temperature limits and high burnup spent fuel, spent fuel pellet oxidation.

The staff's evaluation of material properties is based on NUREG-1617, Interim Staff Guidance (ISG) – 11, Revision 3. "Cladding Considerations for the Transportation and Storage of Spent SNF", and ISG – 15, "Materials Evaluation." The evaluation of the package reflooding is covered in the thermal chapter.

2.2.1 Welding

The cask containment system welds consist of full penetration welds forming the containment shell. All containment boundary welds are fabricated and inspected in accordance with ASME Code Section III, Subsection NB (Table 8.1.6). The weld details and examinations are in the drawing package.

Important to Safety (ITS) welds on the cask (excluding containment boundary welds), Non-Fuel Waste Basket (NFWB), Fuel Spacers and impact limiters shall be examined (or repaired and examined) in accordance with ASME Code Section III, Subsection NF. Basket welds connecting Metamic-HT panels shall be examined and repaired in accordance with NDE specified in the drawing packages and with written and approved procedures developed with acceptance criteria per ASME Section V.

Basket welds connecting Metamic-HT panels to aluminum shims, and welds connecting aluminum plates surrounding the shield blocks shall be examined in accordance with NDE specified in the drawing package and with written and approved procedures, including repaired welds.

Non-Important to Safety (NITS) welds shall be examined and repaired in accordance with written and approved procedures. In the requirement for re-qualifying Friction Stir Welding (FSW) per ASME B&PV Section III, NCA-1140, the use of different code edition for different components is allowed. FSW weld procedure qualification was originally performed to the 2007 Edition of the ASME Code. The essential variables identified in the 2007 version have not changed in later code versions. As a result, there is no expectation that the applicant should re-qualify the weld procedure to a later code edition.

For non-code manufacturing, the applicant will also use ASME Code B&PV Section IX, AWS D11, D12, or equivalent. The staff finds data and procedures described on welding acceptable, based on normal and standard practices that are consistent with the NRC guidance.

The staff questioned a two part welding flagged in drawing, and determined that the applicant will be able to make the same component from one piece, by starting with a bigger part that is machined down to the same dimensions than those shown in the drawing.

The staff questioned the strain energy approach that is used in determining the weld plastic deformation, especially strain localization for structural geometry and for materials inhomogeneity. The applicant accounted for the geometric differences between the modeled connections and the actual weld material at each interface; the weld filler material is also selected to be compatible with the surrounding steel components. Therefore, it has very similar mechanical strength properties than those of the base metal. Given the maximum calculated plastic strain, any minor difference in weld versus base metal strength will not affect the conclusion that the containment remains intact. The staff finds this rationale, based on existing design and data, to be acceptable.

2.2.2 Low Temperature Impact Properties and Other Mechanical Properties of Materials

The applicant measured the Charpy impact energy at -40˚C for various steel components, which may be subject to ductile brittle transition (Tables 8.1.8 and 8.1.9 of the application). Regulatory Guide RG 7.11 requires that either fracture toughness or alternative properties be measured. The values measured are comparable with the recommended values in NUREG/CR-1815 (Holman, W.R. and R.T. Langland. ‘Recommendation for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick.” NUREG/CR-1815, 1981).

Some exemptions from the impact testing, identified in ASME B&PV, Section III (NB-2311, or NB-2333-1), are also utilized for austenitic stainless steel, alloy steel, aluminum bronze, and the impact limiter enclosure (a stainless steel). The staff confirmed the other mechanical properties presented by the applicant with ASME B&PV, Section II Part D and supplemental open literature data. The staff finds all data and procedures described above acceptable, based on normal and standard practices that are consistent with NRC guidance.

2.2.3 High Temperature Materials Properties

The applicant assessed that containment boundary integrity is maintained under HAC with 100% rod rupture, coincident with a fire, and that it is consistent with RG 7.6. Temperature data under this condition is provided in Table 3.1.3 of the application, which includes maximum temperatures under fire, i.e., 502°C for the cladding materials and 473°C for the spent fuel basket materials. Both of these temperatures remain below the HAC limits of cladding temperature, 570°C, and basket temperature, 500˚C.

The applicant also states that the containment gasket can be relaxed (softened) from a fire, as stated in Section 2.1.2.1 of the application. The applicant ensures that there is a leaktight criterion, per ANSI N14.5, under both NCT and HAC. Additional evaluations on the thermal effects on leaktightness are given in the thermal chapter of this SER, including on the seals. The maximum seal temperatures reached under NCT and HAC remain below the limits shown in Table 3.2.12.

There is a possibility that precipitation-hardened aluminum alloys may degrade in its strength over aging at elevated temperature for a prolonged time. ASME B&PV Section II, Part D, requires that time-dependent properties be used for exposure above 177˚C for many structural aluminum grades. For aluminum shims, Type B221 2219-T8511 is used and the applicant assessed the data of ASM (ASM, 2006) and of ASTM (ASTM Specifications B221M-07) for typical mechanical properties of wrought aluminum alloys at various temperatures.

For up to 10,000 hours (over a year), aluminum alloys were soaked from 0.5 to 10,000 hours from room temperature to 370˚C. Strength values at elevated temperatures (up to 290˚C, per Table 2.2.7) are factored in lower-bound values, at tested temperatures corresponding to 10,000 hours prior to soaking. The strength reduction factor is taken as the ratio of the strength value at room temperature after a previous soaking to the typical strength value at room temperature. The staff determined that over one year at up to 290˚C represents the shims condition during transportation, thus a maximum time for the authorized contents to stay in the package.

Additionally, the prior soaking conditions of temperature and time represents the annealed condition for typical aluminum alloys. However, the staff still recognizes uncertainties associated with actual testing at room temperature with prior high temperature soaking. Separate literature data (The Aluminum Association, Inc., 2003) shows that Table 2.2.7 data, up to 290˚C, is acceptable from tests of 10,000 hours at test temperatures. The staff determines that the data assessed on high temperature materials properties is appropriate based on standards, related operating experience, literature information, and conservatism used.

2.2.4 Thermal Expansion Coefficient of Holtite

The applicant claimed it was conservative to assume the same thermal expansion coefficient of Holtite between a cold temperature and 50 ˚C; however, the staff noted that the thermal expansion coefficient of Holtite increases with temperature. Therefore, the thermal expansion coefficient below 50˚C is less than that above 50˚C. Assuming that the thermal expansion below 50˚C (≤ 92 (μm/m-K) is the same as that above 50˚C is indeed less conservative, contrary to the conservatism initially claimed by the applicant.

The staff evaluated the uncertainties associated with the conservatism taken by the applicant. The thermal stress caused by thermal expansion under NCT is always small (due to low temperatures) even if a “higher” conservative thermal expansion coefficient is used under NCT. Also differential expansions of fuel and fuel basket, under NCT, is much less than the nominal cold gaps (Table 3.4.2). This should not be an issue under NCT, per the thermal evaluation.

Under HAC, the NCT bounds HAC because the package cavity (steel shell) expands more than the fuel assembly thereby increasing the gaps between the fuel and inner lid/shell. Therefore, the staff finds the assumption of the same thermal expansion coefficients acceptable but requested the applicant not to state that assuming the same thermal expansion between cold temperature and 50 ˚C is conservative, and also to revise Holtite report accordingly.

2.2.5 High Burnup Fuel

The allowed contents include high burnup (HBF) spent fuel (up to 70 GWD/MTU) with Zircaloy 2, Zircaloy 4, ZIRLO™ and M5® cladding materials. The mechanical properties of HBF cladding may be altered by the formation of radial hydrides under HAC, at cladding temperature above 400°C or with higher hoop stress causing integral fuel burnable absorber (IFBA) SNFs. In IFBA

SNFs, poison materials release may lead to additional helium generation increasing the gap gas pressure.

This pressure increase may increase cladding hoop stress, resulting in reorientation of more hydrides with higher hydrogen concentration in HBF. Cladding mechanical properties may be altered with more radial hydrides, affecting cladding mechanical performance under NCT and HAC. The applicant used appropriate mechanical properties (Pacific Northwest National Laboratory (PNNL) -17700, "PNNL Stress/Strain Correlation for Zircaloy", July 2008). Additionally, the applicant conducted a very conservative defense-in-depth consequence analysis in criticality and shielding, if a significant rod breakage occurs. The applicant obtained the required regulatory margin of safety against criticality and shielding.

The staff also evaluated independently HBF rim effects on the release fraction of radionuclides, which are needed to assess confinement function. The rim is fine-grained (0.1 to 0.3 μm) and Pu-rich microstructure on the surface of SNF fragments in HBF pellets. Previously literature presented possible enhanced release fraction due to the brittle nature and fine grains of rims (Ahn T, R. Sun, T. Wilt, S. Kamas and S. Whaley, "Source Term Analysis in Handling Canister-Based Spent Nuclear SNF: Preliminary Dose Estimate," ADAMS, ML112640440, 2011). However, the staff's assessment, with newer data and indirect previous rationale, concludes that the release fraction would not be increased with HBF.

2.2.6 Metamic-HT™

Metamic-HT™ is a neutron absorber with a structural function, being used in storage and transportation of spent fuel by Holtec International. Metamic-HT™ is a metal matrix composite consisting of an aluminum matrix reinforced by nano-particles of alumina and superfine particles of boron carbide. The staff has recognized that the fracture toughness of Metamic-HT™ used is derived from the Charpy impact energy measurements. The derivation is based on the correlation of fracture toughness and Charpy impact energy, which was developed in steels from literature. The staff assessed the fracture toughness based on an energy balance, along with the available analogue literature data analyses.

To respond to staff's RAIs, the applicant measured fracture toughness values at various operating temperatures, and based on these measurements, the applicant also determined the minimum unstable crack sizes at each temperature. The staff determined that values of both fracture toughness and minimum unstable crack size at each temperature were acceptable. However, the staff also had several questions the on measurement procedures and the use of mechanical properties, i.e., ASTM procedures and other mechanical properties measured with ASTM guides. The staff assessed the applicant's responses and determined that:

- The applicant provided acceptable standard deviations of the yield stress, with data coming from the latest version of the "Metamic-HT™ Qualification Sourcebook."

- The applicant provided an acceptable minimum unstable crack size with a formula for the minimum unstable crack size dependent on the stress demand, not the intrinsic yield stress.

- The applicant provided an acceptable explanation of the effect of crack orientation. The most critical orientation for a flaw or crack in the manufactured Metamic-HT™ panel is used. Also, the applicant noted that, under normal storage conditions, the SNF basket only supports its own dead weight inducing small compressive stresses in the Metamic-HT^{TM} panels. This is not a concern from a fracture toughness standpoint.

- The applicant provided an acceptable explanation for the use of a detectable 1/32” crack size. The codified value of this crack size is provided and under a conservative 1/16” crack size in the Source Book, the corresponding safety factor remains above 1.0.

- The applicant provided an acceptable explanation for the fracture toughness measurement at elevated temperatures. The ASTM E1820-15a uses a J-integral (i.e., energy) vs. crack growth resistance (J-R) curve. The constraints of ASTM E1820-15a allow for the evaluation of meaningful testing data at elevated temperatures (e.g., > 200˚C [392˚F]). The strain energy release rate is not influenced significantly by events within the plastic zone, if the plastic zone is relatively small and accounted for per ASTM E1820-15a guidelines.

- The applicant provided an acceptable explanation for valid fracture toughness measurements for the elastic stress regime below the minimum guaranteed value for the yield stress by extrapolation. The maximum induced primary stress (axial plus bending) in the compact specimen during tensile loading remains below the materials yield stress at 400˚C (752˚F).

- The applicant provided an acceptable explanation on potential geometric reconfigurations for non-mechanistic tip-over events and the effect of plastic deformation. The Metamic-HT^{TM} SNF baskets do not experience any gross plastic deformation, and the primary stresses in the SNF basket panel remain elastic during the non-mechanistic tip-over and 9-meter drop events. Potential localized effect may cause very limited plastic straining.

2.2.7 Aluminum Shims

The applicant assumes aluminum alloy to be effective for the short duration dynamic loading from the tip-over accident. Aluminum alloy, such as Alloy 2219 used by Holtec, is a precipitation-hardened alloy. In response to the staff’s request for information, the applicant provided yield stress and tensile stress at elevated temperatures, i.e., about 260 - 270˚C (500 – 518˚F). The provided data considered the mechanical properties of precipitation hardened aluminum alloys as a function of time at temperature due to overaging.

Based on the staff’s independent literature data review (The Aluminum Association, Inc. “Aluminum Standards and Data 2003), the staff determined that the data provided by the applicant was sufficient to determine that the mechanical properties of the precipitation-hardened aluminum alloy are appropriate.

2.2.8 Aluminum Honeycomb in Impact Limiter

The imposed temperature is low, i.e., about 100˚C, and therefore the strength reduction due to overaging is not present, as described on aluminum shims in this SER.

The applicant uses aluminum honeycomb and the staff verified that the crush strength is adequate and be held to a tolerance band of about 20%.

2.2.9 Potential Pyrophoricity

The applicant includes various material classifications for spent fuel and contents with metals of activated elements, fissile metals, etc., all in solid form and generally described as “chunk.” The structural analyses of SNF rods, in Section 2.11 of the application, show that the SNF is expected to remain essentially undamaged during HAC. Additionally, it is noted that a post-shipment spent fuel Integrity Acceptance Test, described in Section 8.1.8 of the application, is performed to identify a potential reconfiguration of the spent fuel prior to removal of contents.

However, rubbles may be of a small “chunk” size and, for these small “chunk” sizes, the applicant addressed their potential pyrophoricity under NCT and HAC. The applicant states that if SNF reconfiguration did occur resulting in the formation of metal chips or powder in the package, pyrophoricity is prevented because the oxygen is removed from the cask environment.

The applicant also stated that, if the SNF is loaded or unloaded outside spent fuel pool, as discussed in Sections 7.1.2 (Loading of Contents) and 7.2.2 (Removal of Contents) of the application, the package is backfilled with an inert gas whenever SNF is not covered with water, thus preventing a potential ignition of pyrophoric materials. The staff determines that the inerting approach taken by the applicant is acceptable as a standard method to remove moisture or oxygen that could cause pyrophricity.

For non-fuel waste, i.e., metals with activated elements, fissile metals, and plutonium, the applicant also described these as chunks and provides some size information (Table 7.D.8). The applicant classified these as process waste that is debris material representative of typical material collected from spent fuel cleanup, including material generated as a result of cladding failures. Powders (dispersible non-SNF waste per NUREG/CR-6487 [Anderson, et al., 1996]) is another category on its own, as acknowledged by NUREG/CR-6487.

The HI-STAR 80 application does not seek approval for dispersible solids. The application indicates the approved contents are limited to core components or pieces of core components that may have been cut to fit into the waste basket (NFWB-1). However, removable contamination exists but this contamination is not viewed as falling under the category of shipment of dispersible solids.

Thus, it is considered that the HI-STAR 80 non-SNF waste contents are practically free of pyrophoric material with no concern for pyrophoricity under NCT or HAC such that the backfill gas can be air or helium. The staff determines that it is acceptable that the applicant does not approve dispersible solids which is a prerequisite for pyrophricity with a large surface area.

2.2.10 Boron Depletion of Metamic-HT™ and Holtite-B, and Potential Thermolysis and Radiolysis of Holtite-B

Under NCT, the applicant claims that the depletion of the B-10 in the Metamic-HT™ and Holtite neutron shielding material is negligible, i.e., less than a fraction of 10^{-6} over 50 years. This estimate is based on calculations prepared for a similar cask model. The applicant provided testing data, calculation results, or relevant references to arrive at these conclusions for Holtite-B. The applicant calculated B-10 depletion as a result of neutron capture reaction under irradiation from the loaded SNF. The calculated depletion fraction was 9.4×10^{-9}. The staff finds it is acceptable that the calculated fraction is negligible.

The applicant also tested thermolysis and radiolysis of the polymeric Holtite-B at 157– 220˚C under a 40-year design life gamma exposure and a 40-year design life neutron fluence. Metamic-HT™ was not tested because it is metal. For Holite-B, the applicant did not observe any damage or any detectable weight loss or dimensional change due to irradiaion. The applicant measured a 1-2% weight loss under a punitive thermal environment, but this remains well within a 5% materials qualification threshold. The staff determines that it is acceptable that the weight loss is well within the 5% qualification threshold.

2.2.11 Radiation Damage

The package is composed of materials that either have a proven history of use in the nuclear industry or have been extensively tested. The radiation levels from spent fuel do not affect the package materials. Both gamma and neutron exposures are not significant with respect to the dose levels of potential damage. The staff determined that the assessment, made by the applicant based on literature information in general, is acceptable.

2.2.12 CILC Cladding

Crud-Induced Localized Corrosion (CILC) of channeled BWR SNF has the potential of corrosion-induced damage to the cladding and, therefore, cladding integrity may be uncertain. If these assemblies are not dechanneled, visual inspection or ultrasonic testing of the cladding will not be viable. The user will likely need to rely on reactor operating records and/or SNF sipping methods to reasonably demonstrate that the cladding condition is within the bounds of the CoC conditions and not grossly-breached. The staff requested that the applicant provide a plan (e.g., notes, warnings or cautions in the operating procedures) assuring that channeled BWR spent nuclear fuel, selected for loading, is undamaged in accordance with the CoC conditions.

The applicant provided an acceptable assurance process where spent fuel assembly selection and verification is performed by the user in accordance with written, and approved, procedures ensuring that only spent fuel assemblies authorized in the CoC are loaded into the Model No. HI-STAR 80 package. Spent fuel assembly selection and some aspects of assembly verification are typically performed well in advance of the actual loading date with respect to the selection and verification of the assemblies to meet the definition of undamaged spent fuel.

A typical approach to show compliance with the definition of undamaged fuel may include the following steps:

- During reactor operation, the water chemistry is monitored. If no indications of SNF leakage is detected, all assemblies unloaded from the core are considered undamaged.

- If indications of leakage are found in the water during reactor operation, the population of the assemblies in the core that may have the leak may be narrowed down by a more detailed evaluation of the leaked isotopes, or by manipulating control blades in a BWR core.

- Once unloaded, further examination, such as sipping, may be performed to clearly identifying the leaking assembly or assemblies, out of the population identified.

- Once leaking assemblies are identified, they may simply be considered not meeting the CoC conditions and excluded from the selection, or further tests are performed to identify the extent of cladding damage.

- For channeled BWR assemblies, such further tests to identify the extent of the leak, and potentially qualify them as undamaged if the leak does not exceed the requirements for undamaged assemblies, would require the removal of the channel.

Spent fuel handling shall be performed in accordance with written site-specific procedures. The applicant provided detailed procedures to ensure that only spent fuel assemblies authorized in the CoC are loaded into the Model No. HI-STAR 80 package, and unloaded when needed. The staff finds the detailed procedures acceptable.

2.2.13 SNF (Pellet) Oxidation

Per ISG-22, “Potential Rod Splitting due to Exposure to an Oxidizing Atmosphere during Short-term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel”, the applicant assessed the potential oxidation based on (i) appropriate literature information, and (ii) example calculations for time limits at various temperatures. The applicant determined that the oxidation process is highly sensitive to the temperature. Therefore, the time limit for an air/moisture exposure would increase for casks with lower heat load.

The applicant also considered a later stage of vacuum drying with less oxygen partial pressure which is likely to decrease oxidation rate. The staff finds this assessment to be acceptable based on the staff’s previous research on this topic (Anderson, B.L, R.W. Carson and L.E. Fischer. “Containment Analysis for Type B Packages Used to Transport Various Contents.” NUREG/CR-6487, 1996) and the review of updated recent literature information.

2.2.14 Coating and Corrosion

The HI-STAR 80 package exterior steel surfaces are coated with a conventional surface preservative such as Carboguard® 890 and/or an equivalent surface preservative. The applicant stated that Carboguard ® 890 and equivalent surface preservatives have provided years of proven performance on the Model No. HI-STAR 100 package. In addition, exterior surfaces of the package are easily inspected and can be recoated as necessary.

For package coatings, alternate surface preservatives are determined equivalent per the recommendation of a coating manufacturer and with Holtec approval. Carboguard 890 is the product name at the time of this SER. Chemically identical products with different names are permitted. Other coatings that can be shown to have had proven performance in similar applications and environments are permitted. The coating qualifications comply with standards. No galvanic reactions are expected.

The staff does not expect a potential loss of corrosion resistance of the exterior steel surface. This is a transportation package considered for short periods of shipment over the years and, during these short periods of time, the staff does not anticipate any noticeable corrosion, especially surface oxidation (e.g., copper). Also, galvanic corrosion is not expected. Additionally, there are proper inspections before any use of the package.

Copper is introduced in the design of the Model No. HI-STAR 80 package to increase the thermal conductivity of the package, and avoid lead melting. Copper has a unique nature in metal corrosion with a high electrode electrochemical potential compared to most other metals. Therefore, a potential galvanic corrosion of the external surface of the steel in contact with the copper needs to be considered.

However, the applicant eliminated aqueous (water) conditions and imposed significant oxygen free conditions. Considering the various processes adopted by the applicant, and the required high humidity for corrosion at temperatures above ambient conditions, the staff determines that galvanic corrosion of steel is not credible.

2.4 Lifting and Tie-Down Standards

2.4.1 Lifting Devices

In Reference 2.1.12 of the application, the applicant evaluated all devices or components related to lifting operations including the upper lifting trunnions, the trunnion bushings and the overpack forging.

The applicant evaluated the trunnions, bushings and forging using the requirements of ANSI N14.6 for special lifting devices, which requires a factor of safety on yield strength of 6 and a factor of safety on ultimate strength of 10. As noted by the staff in Section 2.1.3 of this SER, these acceptance criteria exceed those required by 10 CFR 71.45(a) which only requires a factor of safety of 3 against yielding for lifting. The applicant evaluated the bending and shear stresses of the upper trunnions as well as the bearing stress on the trunnions, the trunnion bushings and the forging.

Additionally, the applicant stated that the load carrying capacity of the trunnions are governed by the cross section of the trunnion that is external to the cask rather than by any section within the cask; therefore, the loss of the external shank of the lifting trunnion will not cause the loss of any other structural or shielding function of the cask.

The staff reviewed the analyses in Reference 2.1.12 of the application, as they applied to the yield criteria of 10 CFR 71.45(a) for the bending and shear stress of the trunnions only. Because all calculated stresses in the lifting attachments are less than the allowable stress for the associated material by a factor of 3, and because the application of excessive load will not impair the ability of the package to meet other requirements of Subpart E of 10 CFR 71, the staff finds that the package meets the requirements of 10 CFR 71.45(a).

2.4.2 Tie-Down Devices

Section 2.5.2 of the application states that there are no tie-down devices that are a structural part of the HI-STAR 80 package for transport in the United States. Additionally, in Section 7.1.3, the applicant requires the use of custom tie-down devices or other blocking ancillary devices to be used in order to render the top and bottom trunnions inoperable. Figure 7.A.2 of the application illustrates the tie-down configuration.

Because the typical custom tie-down devices shown in Figure 7.A.2 of the application render the upper and lower trunnions inoperable, and because the applicant requires trunnion covers that do the same if another tie-down configuration is used (i.e. the trunnions cannot be used as tie-down devices), the staff finds that the package satisfies the requirements of 10 CFR 71.45(b).

2.5 General Requirements for All Packages

2.5.1 Minimum Package Size

Based on Drawing 9800, the staff finds that the package satisfies the requirements of 10 CFR 71.43(a) for minimum size.

2.5.2 Tamper-Indicating Features

Section 2.4.2 of the application states that the upper impact limiter must be removed to gain access to the closure lid bolts. During transport operations, a cover is installed over one of the access tubes for the impact limiter attachment bolts and attached with a wire tamper-indicating seal with a stamped identifier. This seal will indicate whether or not any tampering with the impact limiter has occurred. Additionally, the applicant stated that the cask closure lid bolts may include holes for installation of tamper-indicating wire seals.

The staff reviewed Drawing No. 9801 and the transport procedures in Chapter 7 of the application and determined that the package satisfies the requirements of 10 CFR 71.43(b) for a tamper-indicating feature.

2.5.3 Positive Closure

In Section 2.4.3 of the application, the applicant stated that there are no quick disconnect valves in the containment boundary and that the only access to the cask cavity space is through the two closure lids which requires special handling equipment to remove. According to the applicant, the only other opening in the containment boundary is through the vent and drain port, which is sealed with a plug and cap as well as a bolted cover plate as shown in Drawing No. 9800. The applicant asserted that, based on the closure system and the analysis for normal and accident condition pressure, the package meets the requirements for positive closure.

The staff reviewed drawing No. 9800 and the applicant’s analysis for normal and accident pressure conditions and concluded that the containment system is securely closed by a positive fastening device and cannot be opened unintentionally or by a pressure that may arise within the package and therefore satisfies the requirements of 10 CFR 71.43(c) for positive closure.

2.6 Normal Conditions of Transport

2.6.1 Heat

The applicant evaluated the HI-STAR 80 package for the effects of thermal expansion as a result of hot NCT. Based on an initial temperature of 68°F, and the final temperatures listed in Table 2.6.2 of the application, the applicant calculated the thermal expansion of the various components of the HI-STAR 80 transportation package and reported the resulting gaps (initial and final) in Table 3.4.2.

The staff reviewed Table 3.4.2 as well as Drawings 9800 and 9797 and the thermal expansion calculations. Based on Table 3.4.2, the staff determines that there are no interference situations among the various components of the HI-STAR 80 transportation package due to differential thermal expansion; therefore, there are no induced stresses within the components, and the hot conditions of 10 CFR 71.71(c)(1) do not substantially reduce the effectiveness of the package.

In Section 2.6.1.4.1, the applicant analyzed the containment boundary under load combination N1 using the ANSYS model described in this SER. In Table 2.6.5 of the application, the applicant presented the calculated stresses and associated factors of safety for the components that comprise the containment boundary. Because the FS for all components is greater than 1.0, the staff determines that the structural performance of the containment boundary is adequate under load condition N1.

2.6.2 Cold

The applicant evaluated the transportation package under cold NCT (-40°F) with respect to internal pressure, allowable stresses, bolt stress, and differential thermal expansion. With respect to internal pressure and allowable stresses, the applicant concluded that the internal pressure will decline with decreasing ambient temperature while the material allowable stresses will increase under the same condition. The applicant concluded that decreasing the load and increasing the available strength of the material would result in larger margins of safety than what would be expected for a hot condition.

In Calculation 19 of Reference 2.1.12 of the application, the applicant determined the initial stress in the closure bolts at 70°F due to the preload, and computed the final stress as a result of a differential temperature change of -110°F (from 70°F to -40°F). The applicant determined that the final preload of the inner and outer closure lid bolts remained higher than the required closure lid seal seating loads and that the sealing conditions were unchanged as a result of the environmental change.

The staff reviewed the calculations and subsequent conclusions made by the applicant and determines that the cold conditions of 10 CFR 71.71(c)(2) do not substantially reduce the effectiveness of the package.

2.6.3 Reduced External Pressure

The applicant stated that the reduced external pressure equal to 25 kPa (3.5 psia) is bounded by the results of the internal pressure analysis (Load Combination N1). The staff reviewed Load Combination N1 and concludes that it bounds the reduced external pressure condition, and that the reduced external pressure conditions of 10 CFR 71.71(c)(3) do not substantially reduce the effectiveness of the package.

2.6.4 Increased External Pressure

The applicant stated that an increase in external pressure of 140 kPa (20 psia) is bounded by the external pressure of 2MPa (260 psia) required by 10 CFR 71.61 and that no additional analysis was required to demonstrate the performance of the package.

Because the requirements of 10 CFR 71.61 bound those of 10 CFR 71(c)(4), the staff concludes that the increased external pressure conditions of 10 CFR 71.71(c)(4) do not substantially reduce the effectiveness of the package.

2.6.5 Vibration and Fatigue

2.6.5.1 Vibration

The applicant calculated the natural frequencies of the HI-STAR 80 fuel basket panel and the containment shell and determined that the values exceeded those expected during normal conditions of transport. The applicant also determined the natural frequency of the fuel rod by analyzing it as a clamped beam with a length equal to the longest span between adjacent grid spacers. The applicant reported a frequency of 37.1 Hz, and considers the fuel rod to be rigid under NCT and analyzes the cladding under a 5g load. In Table 2.6.9, the applicant reports a safety factor 5.12 against bending.

The staff reviewed the applicant's calculations and determines that the large natural frequencies of the basket plates and the cask will preclude any resonance conditions during NCT and that the structural performance of the fuel cladding is adequate under the vibratory loads for NCT.

2.6.5.2 Fatigue

The applicant considers cyclic operations of the containment cask using the criteria of ASME B&PV Code Section III, Division 1, Subsection NB-3222.4(d). The applicant evaluated five conditions including (i) Atmosphere to Service Pressure Cycles, (ii) Normal Service Pressure Fluctuation, (iii) Temperature Difference at Startup and Shutdown, (iv) Temperature Difference for Normal Service, and (v) Mechanical Loads, to demonstrate that the package was exempt from detailed fatigue calculations. The staff reviewed the applicant's evaluation and, because the criteria from NB-3222.4(d) are satisfied, determined that a detailed fatigue analysis is not required on the containment cask.

In Section 2.6.1.3.2 of the application, the applicant performed a fatigue analysis on both types of closure lid bolts, the closure lid port cover bolts and threads on the containment closure flange and various other connections. The table below presents the maximum permissible number of cycles, determined by the applicant, for each of the components.

Component	Maximum Permissible Cycles
Closure lid Bolts (SA-564 630 H1100)	241
Closure Lid Bolts (SB-637 N07718)	257
Closure Lid Port Cover Bolts	588
Containment Upper Flange Closure Bolt Threads	20,000
Stainless steel threads of various plugs and caps	20,000
Vent/Drain Bronze Plugs	2,400

The applicant also considered fatigue failure of the fuel cladding due to the vibratory cyclic loading during transportation.

Using the model described in Section 2.6.5.1 of this SER (Vibration) and the 5-g load, the applicant determined that the calculated bending stress in the fuel rod was less than the fatigue endurance limit of 25,730 psi for Zircaloy (NUREG/CR-1132).

The staff reviewed the applicant's calculations and determines that the structural performance of the containment boundary and the fuel cladding are adequate under the fatigue loads for NCT.

2.6.6 Water Spray

Because the HI-STAR 80 is a large shipping cask, in accordance with RG 7.8, the staff finds that the water spray test of 10 CFR 71.71(c)(6) has no significance in the structural design of the package and will not substantially reduce the effectiveness of the package.

2.6.7 Free Drop

In Section 2.6.1.4 of the application, the applicant analyzed the HI-STAR 190 for the 1-foot free drop under hot conditions using LS-DYNA. The applicant reported a maximum deceleration of 28.9g and presented the calculated stresses and associated factors of safety for the containment boundary components subjected to a 1-foot drop (load combination N2) in Table 2.6.6. Because the factors of safety for all components are greater than 1.0, the staff determines that the structural performance of the containment boundary is adequate for the 1-foot free drop test.

2.6.8 Corner Drop

Because the HI-STAR 80 is a large shipping package (greater than 220 lbs), the staff finds that the corner drop test of 10 CFR 71.71(c)(8) is not applicable.

2.6.9 Compression

Because the HI-STAR 80 weighs more than 11,000 lbs, the staff finds that the compression test of 10 CFR 71.71(c)(9) is not applicable.

2.6.10 Penetration

Because the HI-STAR 80 is a large shipping package, in accordance with RG 7.8, the staff finds that the penetration test of 10 CFR 71.71(c)(10) has no structural significance and will not substantially reduce the effectiveness of the package.

The staff reviewed the packaging structural performance under the normal conditions of transport proscribed in 10 CFR 71.71 and concludes that there will be no substantial reduction in the effectiveness of the packaging that will inhibit its ability to satisfy the requirements of 10 CFR 71.51(a)(1) for a Type B package and 10 CFR 71.55(d)(2) for a fissile material package.

2.7 Hypothetical Accident Condition

The applicant evaluated the structural performance of the HI-STAR 80 under hypothetical accident conditions based on the sequential application of the tests specified in 10 CFR 71.73.

2.7.1 Free Drop

For the free drop, the applicant evaluated four different categories of drop orientations: vertical end drops, side drops, center of gravity (CG) over corner drops, and slap down drops where one end impacts first and the cask rotates and impacts the opposite end. Table 2.7.3 of the application lists the different drop orientations that the applicant analyzed to identify the most damaging scenario for the structural components. The applicant simulated the 30-foot drop by positioning the model of the cask over the concrete surface in the orientation to be evaluated, and imparting a velocity of 43.9 ft/sec on the LS-DYNA model.

The applicant summarized the maximum deceleration values for each of the simulated drop scenarios as well as the maximum calculated crush depth of the impact limiter in Table 2.7.3 of the application. In all cases, the maximum crush depth of the impact limiter was less than the allowable crush depth. The allowable crush depth is based on the distance from the outside edge of the impact limiter to the closest point on the steel cask or the impact limiter backbone, except for the end drop where the allowable crush depth is the distance to the outer end of the honeycomb blocks with a larger diameter.

Table 2.7.4 of the application lists the maximum calculated stress in the components that make up the containment portion of the cask along with their respective allowable stress values, the safety factor and the governing accident that produced the maximum stress in that component. All safety factors in Table 2.7.4 are greater than 1.0 with the lowest value of 1.11 due to the stress intensity in the outer closure lid bolts as a result of a 30-foot, bottom first, slap-down drop event.

In addition to the stress intensities in Table 2.7.4, the applicant reported the maximum radial lead slump in the upper/lower forging gamma shield (as appropriate) and the maximum axial lead slump in the containment shell gamma shield. These values of lead slump were used by the applicant for the shielding evaluation without an increase for conservatism. The applicant also stated that the lid seals remained sufficiently compressed after the drop accidents.

In Table 2.7.6, the applicant reported that the calculated primary effective stress in the dose blocker components was less than the ultimate strength of the material. Additionally, the applicant reported a maximum calculated deformation of the fuel basket of less than 1 mm, which satisfies their dimensionless deformation criteria.

In Table 2.11.3, the applicant reported the maximum cladding deceleration for the 30-foot drop as well as the peak principal strains in the Zircaloy. The peak principal strain for the Zircaloy cladding was below the allowable strain value and resulted in a factor of safety of 1.40.

Based on a review of the applicant's analysis in LS-DYNA, and the calculated structural results, the staff has reasonable assurance to conclude that the free drop test of 10 CFR 71.73(c)(1) will not diminish the structural performance of the HI-STAR 80.

2.7.2 Crush

Because the weight of the HI-STAR 80 is greater than 1100 lbs, the staff determines that the crush test prescribed by 10 CFR 71.73(b)(2) is not applicable.

2.7.3 Puncture

For the subsequent puncture test, the applicant retained the LS-DYNA model from the top end drop and considered puncture impact on the top end and the side wall of the cask. The applicant added a mild steel bar fixed to the ground with the appropriate dimensions to the LS-DYNA model, positioned the cask in the orientation considered and applied an initial velocity corresponding to a 1-foot drop. The applicant reported the following:

- The bolted joint maintained its integrity with a large margin of safety,

- The bar does not fully penetrate the dose blocker material surrounding the containment shell and that the penetration does not yield unacceptable shielding consequences, and

- The primary stress levels in the closure lid, containment shell and baseplate remain below their respective Level D allowable limits.

Base on a review of the applicant's analysis in LS-DYNA analysis, the staff has reasonable assurance to conclude that the puncture test of 10 CFR 71.73(c)(3) will not diminish the structural performance of the HI-STAR 80.

2.7.4 Thermal

The applicant evaluated the Model No. HI-STAR 80 package for the subsequent thermal accident in Section 2.7.4 of the application, and provided detailed calculations in Holtec Report No. HI-2156553, Revision 0. The applicant's evaluation consisted of ensuring that:

(i) the average temperature across any section of the containment boundary material remains below the maximum permissible temperature,

(ii) internal interferences due to thermal expansion do not develop among the internal components, and

(iii) the cask closure lid bolts do not unload causing leakage from the containment boundary.

In fact, the applicant reported that stress in the closure lid bolts increases, which serves to increase the clamping force of the closure lids, but remained below the Level D limit. The applicant's analysis demonstrated that there was a sufficient margin in all cases and that the containment boundary of the Model No. HI-STAR 80 package remained intact under fire accident conditions.

Based on a review of the applicant's calculations, the staff has reasonable assurance that the thermal test of 10 CFR 71.73(c)(4) will not diminish the structural performance of the Model No. HI-STAR 80 package.

2.7.5 Immersion – Fissile Material

This requirement is bounded by the deep water immersion requirement of 10 CFR 71.61; therefore, the staff concludes that the HAC test requirement of 10 CFR 71.73(c)(5) is satisfied.

2.7.6 Immersion – All packages

This requirement is bounded by the deep water immersion requirement of 10 CFR 71.61; therefore, the staff concludes the HAC requirement test of 10 CFR 71.73(c)(6) is satisfied.

The staff reviewed the packaging structural performance under the hypothetical accident conditions proscribed in 10 CFR 71.73 and concludes the packaging has adequate structural integrity to satisfy the subcriticality, containment, and shielding requirements of 10 CFR 71.51(a)(2) for a Type B package and 10 CFR 71.55(e) for a fissile material package.

2.8 Special Requirements for Irradiated Nuclear Fuel Shipments

2.8.1 Deep Immersion

In Section 2.7.7 of the application, the applicant analyzed the overpack containment boundary of the HI-STAR 80 for deep immersion. The applicant stated that the external pressure of 290 psi acts in a direction that increases the pressure on the land (the contact surface between the top flange and lid); therefore, in-leakage of water from this accident condition is not a concern.

The applicant used ASME Code Case N-284 to evaluate the stability of the containment shell in Holtec Report No. HI-2156553, Revision 0, and assumed the outer dose blocker parts do not prevent the 290 psi pressure from acting directly on the outer surface of the containment shell. The applicant's calculations indicate that the containment shell does not yield or buckle as a result of this accident condition.

The staff reviewed the packaging structural performance under an external pressure of 290 psi for not less than one hour and finds that the package does not buckle, collapse or allow the in-leakage of water and therefore satisfies the requirements of 10 CFR 71.61 for irradiated nuclear fuel shipments.

2.9 Summary of Damage

As noted in Section 2.7.8 of the application, the applicant has demonstrated that the Model No. HI-STAR 80 package is capable of maintaining its structural integrity to satisfy the requirements of 10 CFR 71.61 for deep water immersion and 10 CFR 71.51(a)(2) and 10 CFR 71.55(e) for the sequentially applied hypothetical accident tests of 10 CFR 71.73.

Specifically, the analyses show that the HI-STAR 80 containment space will remain inaccessible to the moderator under the immersion event of 10 CFR 71.73, which follow the free-drop, puncture-drop and fire tests. Both the inner and outer closure lids will maintain a positive contact load with the top flange, which enables the seals to remain functional as effective leak barriers to moderator intrusion to the containment cavity. While there is some plastic deformation due to the puncture bar test, there is no penetration of the containment barrier. Finally, the average basket deflection in the active fuel region is less than the established limit and no damage occurs to the fuel cladding.

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

3.0 THERMAL REVIEW

The objective of the thermal review is to:

(i) Verify that the thermal performance of the Model No. HI-STAR 80 package has been adequately evaluated for the tests specified under both NCT and HAC, and that the package design satisfies the thermal requirements of 10 CFR Part 71,

(ii) Determine whether the package fulfills the acceptance criteria listed in Section 3 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent

Nuclear Fuel," as well as associated Interim Staff Guidance (ISG) documents, and

(iii) Ensure that the peak cladding temperatures (PCTs) and package component temperatures are below the required limits and the temperature gradients within the fuel basket are minimized to reduce the thermal stresses.

3.1 Description of Thermal Design

The Model No. HI-STAR 80 package is designed for maximum heat loads of 50 kW or 54 kW, for the F-12P or F-32B baskets, respectively. The fuel basket, formed by a honeycomb structure of Metamic-HT plates, is surrounded by an array of shaped aluminum spacers (basket shims) in the package cavity peripheral spaces. The major functions of the basket shims are heat transfer and lateral structural support to the fuel basket. The axial holes in the basket shims serve as the passageway for the flow of the helium gas under natural convection.

The heat is dissipated in the fuel baskets principally by heat conduction in the Metamic-HT plates arrayed in two orthogonal directions. Heat dissipation in the fuel basket peripheral spaces is by a combination of contact heat transfer, helium conduction and radiation across the peripheral spacer gaps, and by conduction through the basket shims. The inner surface of the package delivers its heat across the containment shell principally by conduction in which the highly conductive copper-chromium ribs play a critical role.

The non-fuel waste basket (NFWB-1) is a square shaped compartment formed by stainless steel plates to hold non-fuel waste. The applicant stated in Section 3.1.1 that it is not necessary to backfill the package cavity with helium, to improve its thermal performance, because of the significantly low decay heat of 2 kW for a package loaded with non-fuel waste.

The staff reviewed the Model No. HI-STAR 80 package thermal design and verified that (a) the package is designed to safely dissipate heat under passive conditions and (b) the package and contents' temperatures will remain within their allowable values or criteria for NCT and HAC, as required in 10 CFR Part 71. The staff reviewed the package description and evaluation and concludes that they satisfy the thermal requirements of 10 CFR Part 71.

3.2 Material Properties and Component Specifications

The applicant presented the material properties and specifications in Section 3.2, "Material Properties and Component Specifications." The applicant provided the package component thermal properties in Tables 3.2.2 through 3.2.9 and 3.2.13 and listed the temperature limits of structural materials, fuel cladding, and package components in Tables 3.2.10, 3.2.11, and 3.2.12, respectively.

The applicant stated in Section 3.2.2, "Component Specifications," that the cladding temperature limits, specified in ISG-11 Rev. 3, are applicable to all fuel types, burnup levels and cladding materials approved for power generation. The neutron absorber material (Metamic-HT) is stable in excess of 538˚C (1000˚F).

The applicant noted in Table 3.2.13 that the copper chromium, used for the copper shells and ribs, has a lower thermal conductivity than the one of pure copper. Therefore the thermal conductivity of pure copper is adopted in the 30-minute fire to maximize heat input into the package and the thermal conductivity of copper chromium is adopted in the post-fire cooldown for less heat dissipation through the copper components.

The use of copper chromium as material for the copper shells and ribs keeps the PCT and maximum package component temperatures below the allowable limits under NCT and HAC. It also keeps the temperature of lead (used for gamma shield and enclosed within the steel container) below its melting point of 620°F under HAC.

The applicant provided surface emissivity data in Table 3.2.6 and presented the calculated package exposed surface heat transfer coefficients by accounting for both natural convection heat transfer and radiation heat transfer.

The staff reviewed Section 3.2, “Material Properties and Component Specifications,” and Tables 3.2.1 through 3.2.9, as well as Table 3.2.13, used for the material properties of contents and packaging components and finds acceptable the material properties, emissivity/absorption data and component specifications used in the thermal analysis. The staff determines that the material properties and component specifications provided in Section 3.2 of the application are sufficient to provide a basis for the thermal evaluation of the package against the thermal requirements of 10 CFR Part 71.

3.3 General Considerations

3.3.1 Grid Sensitivity Studies

The applicant stated, in Section 3.3.1.6, “Grid Sensitivity Studies,” that the grid sensitivity studies were performed with three sets of mesh sizes Mesh 1, Mesh 2 and Mesh 3, as shown in Table 3.3.5. The applicant calculated a Grid Convergence Index (GCI) of 0.69% for Meshes 1, 2, and 3 and adopted the Mesh 2 grid layout for all thermal analyses of the HI-STAR 80 package.

The staff reviewed the calculations of the GCI and confirmed that a GCI of 0.69%, i.e., less than 1%, is acceptable.

3.3.2 Fin Effectiveness

The exterior body of the Model No. HI-STAR 80 package is engineered with fins to increase the heat transfer area and enhance heat dissipation by natural convection. Such heat dissipation is characterized by a fin effectiveness (also called fin enhancement factor), which is defined as a ratio of the energy dissipated from the finned surface to the energy dissipated from the bare surface. The applicant simulated the finned surface by a cylindrical surface (or bare surface) with a fin enhancement factor for the external convective heat transfer.

The applicant used an ANSYS FLUENT computational fluid dynamics (CFD) fin model to verify the fin efficiency and the fin array effectiveness for heat dissipation. The fin model is made of the geometries of finned package surfaces with four fins and the bare package surface, while the non-physical “edge” effects at the boundary are excluded. The applicant’s CFD fin model calculated a fin enhancement factor, but adopted, to be conservative, a smaller fin enhancement factor of 2.85 in the thermal model.

1) The staff reviewed the package surface temperatures, thermal properties and Raleigh number used by the applicant to determine the flow range and the heat transfer

coefficient on the bare cylindrical surface under NCT. The staff confirmed that the heat transfer coefficient used for the bare cylindrical surface is acceptable under normal conditions of transport.

2) The staff reviewed the equations and analytical procedure for the fin effectiveness factor, as shown in Appendix A, "Fin Effectiveness Evaluation," of Holtec's Report HI-2156468. The staff checked the equations of the fin effectiveness factor and performed confirmatory calculations. The staff finds that the fin effectiveness factor is acceptable.

3) The staff checked the design drawings and confirmed that the low-profile fins are distributed on the entire outer shell surface.

4) The staff reviewed the report "CFD Evaluation of Cask Fin Efficiency for HI-STAR 80," and accepts the methodology used in the CFD fin model. The fin enhancement factor calculated by the CFD fin model is consistent with the value calculated by the analytical method.

Based on the reviews mentioned above, the staff concluded that the fin effectiveness, and its corresponding heat transfer coefficient, adopted in the licensing basis model, are acceptable for the thermal evaluation of the package.

3.4 Thermal Evaluation of Loading Operations

3.4.1 Time-to-Boil Limits

The applicant stated in Section 3.3.4, "Time-to-Boil Limits," that water inside the package cavity is not permitted to boil during fuel loading operations, in accordance with NUREG-1536. The applicant performed an adiabatic heat up calculation to determine a bounding heat-up rate, based on the package heat load, and thermal inertia of the loaded package, and then obtained the maximum permissible time duration for the fuel to be submerged in water.

The applicant determined the maximum allowable time for completion of wet transfer operations, based on the temperature rise rates of 5.16°K/hour for the F-12 basket and 5.64°K/hour for the F-32B basket. The applicant tabulated the maximum allowable time for completion of the wet transfer operations in Table 3.3.6.

The staff reviewed Section 3.3.4 and Table 3.3.6 and confirmed that:

(a) the assumptions and the methodology used to derive the time-to-boil limits are appropriate for derivation of the maximum allowable time for completion of wet transfer operations (the time-to-boil limit) for the HI-STAR 80 package, based on staff's engineering justification on physical phenomena and thermal characteristics, and

(b) the allowable time limits for completion of wet transfer operations of the F-12P and F-32B fuel baskets, as shown in Table 3.3.6, bound those for 10 assemblies in the F-12P fuel basket and 28 assemblies in the F-32B fuel basket, respectively, because of higher heat loads in the F-12P and F-32B fuel baskets.

3.4.2 Moisture Removal Operations

3.4.2.1 Vacuum Drying for SNF

At the design heat load in the HI-STAR 80 package, the PCT of high burnup fuel (HBF) cannot be maintained below 400°C (752°F), as defined in ISG-11 Rev. 3, under a vacuum condition of infinite duration. Thus, cycles of vacuum drying are to be performed until a drying criterion is achieved.

The applicant stated in Section 3.3.6.1 that:

(a) if the drying completion criteria is not met, then the package cavity must be backfilled with helium to 1 atm absolute pressure for cooldown before it reaches the ISG-11 Rev. 3 temperature limit of 400°C (752°F) for HBF,

(b) the fuel cooling under helium is evaluated until the fuel temperature decreases by 65°C (117°F) and the maximum permissible time is obtained from the transient evaluation; and

(c) if a total of 10 drying cycles fail to meet the drying criteria, then other competent means, such as a forced helium dehydrator (FHD) must be used to dry the fuel or the package must be de-fueled.

The applicant stated in Section 3.3.6.1 that:

(a) the time required for the HBF to heat up from an initial temperature of 100°C (212°F) to 380°C (716°F) is determined in a cycle to keep PCT below the allowable limit of 400°C (752°F), if the helium backfill operation is started within 30 minutes after the vacuum drying operation time limit is reached, and

(b) the time required for the (MBF) to heat up from an initial temperature of 100°C (212°F) to 550°C (1022°F) is determined in a cycle to keep PCT below the allowable limit of 570°C (1058°F), if the helium backfill operation is started within 2 hours after the vacuum drying operation time limit is reached.

The staff reviewed those statements in the application and the variations of peak cladding temperatures with time during vacuum drying, as shown in Figure 3-8.1 for the F-12P basket (containing HBF), Figure 3-8.2 for the F-32B basket (containing HBF), Figure 3-8.3 for the F-12P basket (all MBF), and Figure 3-8.4 for the F-32B basket (all MBF).

The staff determined that the PCTs of 380°C for HBF and 550°C for MBF, used for determining the vacuum drying operation time limits, are acceptable for HBF and MBF, respectively, when the fuel is not exposed to air at loading based on the temperature margins allowed for vacuum drying operations.

Regarding the “Exposure of Fuel Cladding to Air during Loading” (Vacuum Drying), the applicant noted, in Table 7.1.6, that, in the conditions for loading (vacuum drying) and unloading with air exposure, a temperature limit of 400°C is set for fuel cladding and is applied to both MBF and HBF because the temperature of 400°C is also the temperature threshold to avoid fuel oxidation due to air exposure. Therefore, the temperature limit of 570°C does not enter into the logic for MBF during air exposure conditions. After exiting the air exposure conditions, the limit of 570°C can again be applied for MBF. The applicant also stated that the time limit of 30 minutes will also apply during the air exposure conditions; therefore, this time limit does not need to be revised and does not need to be explicitly stated in Table 7.1.6. In any case, time limits have included the completion of certain steps (instead of time limits for the beginning of certain steps).

The staff reviewed Table 7.1.6 and the related analysis and finds acceptable the temperature limit of 400°C and the proposed air exposure threshold time limits (ATTL) for fuel cladding of both MBF and HBF for fuel loading (vacuum drying) and fuel unloading with air exposure. The staff finds that the fuel oxidation could be avoided with the fuel cladding below 400°C with the ATTL not exceeded.

3.4.2.2 Vacuum Drying for NFW

Packages loaded with NFW may be dried using a vacuum system to reduce the internal pressure to less than 6 torr while maintaining the pressure below 6 torr for a period of 2 hours minimum: a pressure of 6 torr represents a saturation pressure of water at approximately 4°C which will insure that any liquid water does not freeze once boiling commences.

The staff reviewed Section 3.3.6.1 and agrees with the 6-torr and 2-hour criteria for NFW, based on thermodynamics principles: the dryness criteria with a pressure less than or equal to 6 torr for a time period of not less than 2 hours for non-fuel waste package can insure that any remaining water could accumulate the heat necessary to evaporate to a gas.

Therefore, the staff determines that the cask drying method and dryness criteria for non-fuel waste, as shown in Table 7.1.2, are acceptable.

3.4.2.3 Forced Helium Dehydration (FHD)

The FHD ensures that the fuel cladding temperature remain below the HBF temperature limit of 400°C (752°F) for all combinations of SNF type, burnup, decay heat and cooling time authorized for loading in the HI-STAR 80 package. The FHD operation induces a forced convection heat transfer and has the PCT lower than the PCT in the quiescent mode of cooling under NCT.

The staff reviewed Section 3.3.6.2 and agrees that FHD method, reviewed and accepted by NRC, can be conducted to maintain the PCT below the 400°C for all SNF fuel types in the HI-STAR 80 package because of its high heat removal capability enhanced by forced convection.

3.5 Thermal Evaluation under NCT

The applicant performed the NCT thermal evaluation of the Model No. HI-STAR 80 package using the ANSYS FLUENT CFD code, as described in Section 3.3, “Thermal Evaluation under Routine and Normal Conditions of Transport.”

Since the decay heat released by the core components is significantly lower than the decay heat of the fuel assemblies, the temperature of the package loaded with the non-fuel waste basket is much lower than the temperature of the package loaded with the fuel basket. Therefore, the thermal evaluations were performed on the package loaded with the fuel baskets (F-12P and F-32B).

The staff reviewed the assumptions, boundary conditions, and parameters used in the computer model, and finds them acceptable for the thermal model used for NCT thermal evaluation.

3.5.1 Heat and Cold

The applicant described the loading patterns of the F-12P and F-32B fuel baskets in Table 7.D.1 and Holtec's Report HI-2156468, "Thermal Evaluation of HI-STAR 80 in Transport." The loading patterns of the F-32B fuel basket are categorized as 32 FAs (1.687 kW/FA), 28 FAs (1.928 kW/FA), and 24 FAs (2.35 kW/FA), all within the maximum permitted total heat load of 54 kW. The loading patterns of the F-12P fuel basket are categorized as 12 FAs with a design heat load of 50 kW and 10 FAs with a design heat load of 48.5 kW. The heat load limit of non-fuel waste (NFW) is 2 kW as shown in Table 7.D.8. Under NCT, the heat is transferred by radiation, natural convection, solar insolation, and uniform internal decay heat.

The applicant analyzed the thermal performance of HI-STAR 80 and provided the normal transport maximum temperature of 367°C (693°F) for the F-12P fuel basket and of 374°C (705°F) for the F-32B fuel basket. The heat load pattern of the F-32B fuel basket (with 32 FAs and a heat load 54 kW) is the bounding scenario for the HI-STAR 80 package.

The staff reviewed the NCT maximum temperatures of the F-12P baskets and F-32B basket, as shown in Tables 3.1.1.A and 3.1.1.B, respectively, and confirmed that the heat load pattern of the F-32B fuel basket with 32 FAs and a heat load 54 kW is the bounding case for the HI-STAR 80 package because of its high decay heat and maximum fuel and component temperatures.

The applicant stated in Section 3.1.5, "Cask Surface Temperature Evaluation," that a maximum surface temperature of 125°C (257°F) is predicted in still air at 38°C and in shade, as shown in Table 3.1.5, which is above the allowable surface temperature limit of 85°C. Section 3.1.5 states that a personnel barrier will be required to meet the accessible surface temperature limit specified in 10 CFR 71.43(g) in an exclusive use shipment.

The staff reviewed Table 3.1.5 and confirmed use of the personal barrier is required with the calculated maximum accessible surface temperature (125°C) above the limit of 85°C permitted by 10 CFR 71.43(g) for the exclusive use shipment when transported in a 38°C ambient with no insolation.

The applicant presented the maximum fuel cladding and package component temperatures of the F-12P and F-32B fuel baskets, respectively, in Tables 3.1.1.A and 3.1.1.B. The applicant adopted the F-32B fuel basket with 32 FAs as the bounding pattern for the thermal analysis because of its highest cladding temperature of 374˚C (705˚F) calculated by the applicant.

The staff reviewed Tables 3.1.1.A and 3.1.1.B and confirmed that the PCTs are below the 400°C limit, consistent with ISG-11 Rev. 3, and the maximum temperatures of fuel basket, basket shim, neutron shield, gamma shield and containment seals at lid, vent port and drain port are within the HI-STAR 80 structural materials, fuel cladding and component temperature limits as shown in Tables 3.2.10, 3.2.11 and 3.2.12, respectively.

The staff confirms that the Model No. HI-STAR 80 package meets the thermal requirements of NCT, in compliance with 10 CFR 71.71.

3.5.2 Maximum Normal Operating Pressure (MNOP)

The applicant stated in Section 3.3.3, "Maximum Normal Operating Pressure (MNOP)," that the HI-STAR 80 package is de-moisturized and backfilled with dry helium after fuel loading and prior to lid closure. The MNOP evaluation is based on the initial maximum backfill of 29.0 psia, water vapor, and helium form the radioactive decay, and generation of flammable gases. The applicant provided MNOPs of the F-12P and the F-32B fuel baskets in Tables 3.1.2.A and 3.1.2.B, respectively.

The applicant predicted MNOPs of (a) 51.4 psia for F-12P fuel baskets and 52.2 psia for F-32B fuel baskets under NCT with 0% rods rupture, and (b) 53.6 psia for F-12P fuel baskets and 55.2 psia for F-32B fuel baskets under NCT with 3% rods rupture, as shown in Tables 3.1.2.A and Table 3.1.2.B. The applicant predicted the pressures in the inter-lid space of 22.5 psia for F-12P fuel baskets and of 22.7 psia for F-32B fuel baskets under NCT.

The staff reviewed the MNOPs shown in Tables 3.1.2.A and 3.1.2.B and the pressure limits listed in Table 2.1.1, and determined that (a) the calculated MNOPs for both the F-12P and F-32B fuel baskets (with 0% or 3% rods rupture) are bounded by the cavity space MNOP of 80 psia and the design internal pressure of 100 psia for the cavity space, and (b) the calculated inter-lid space pressures of the F-12P and F-32B fuel baskets are below the allowed pressure of 34.7 psia (or 20 psig), as specified in Table 2.1.1. The staff confirmed the calculations of the cavity space MNOP and inter-lid space pressure under NCT.

3.5.3 Maximum Thermal Stress

The HI-STAR 80 package uses (a) high conductivity materials (Metamic-HT and low alloy steels) to minimize temperature gradients, and (b) large fit-up gaps to allow unrestrained thermal expansion of the package internals during NCT.

The applicant presented the thermal expansion of the fuel and the fuel basket during NCT in Table 3.4.2:

(1) the nominal cold gaps in the design are much greater than the differential expansions of the fuel basket in axial and radial directions,

(2) the gaps in NCT bound the gaps in HAC because of the expansion of the package body under direct fire heating, and

(3) the package cavity (steel shell) expands more than the fuel assembly, thereby increasing the axial gap between the fuel assemblies and the inner lid.

The staff reviewed Section 3.4.4 and Table 3.4.2 and finds that the thermal stress is not a significant issue under the NCT thermal evaluation, based on an engineering justification on the material expansion coefficients and the calculated fuel cladding and package component temperatures.

3.5.4 Personnel Barrier Evaluation

A personnel barrier will be used for the HI-STAR 80 package to prevent access to the package hot surfaces. The applicant performed the thermal calculation, with personnel barrier characteristics provided in Table 3.3.7, to evaluate the impact of a personnel barrier to the package heat removal capability and tabulated the results in Table 3.3.8 for the F-32B fuel basket.

The staff reviewed the personnel barrier characteristics, the calculated temperatures tabulated in Table 3.3.8, and compared these temperatures to those of Table 3.1.1.B (without a personnel barrier) for the bounding F-32B fuel basket scenario. The staff confirmed that:

(a) the maximum fuel cladding and package component temperatures for the F-32B fuel basket (with a personnel barrier) remain below their corresponding limits as shown in Tables 3.2.10 to 3.2.12, and

(b) the deployment of the personnel barrier has no significant impact onto the package heat removal capability because the flow resistance caused by the personnel barrier is negligible and the package component temperatures, essentially unchanged, still remained below the required limits.

3.5.5 Fuel Reconfiguration under NCT

The applicant performed thermal analyses for fuel reconfiguration under NCT as described in Section 3.3.7, "Fuel Reconfiguration under Normal Conditions." With a 3% fuel failure postulated under NCT, the applicant assumed 3% of the design heat load on the bottom surface of the package inner closure lid and 97% of the design heat load in the intact fuel region and provided the maximum NCT temperatures in Table 3.3.10 and the maximum NCT cavity and inter-lid space pressures in Table 3.3.11.

The staff reviewed Tables 3.3.10 and 3.3.11 and confirmed that maximum fuel cladding and package component temperatures are below their respective temperature limits, as shown in Table 3.2.10. The cavity pressure of 55.0 psia and the inter-lid space pressure of 24.3 psia are also below the cavity design internal pressure (114.7 psia) and the package inter-lid space maximum operating pressure (< 34.7 psia), respectively, as reported in Table 2.1.1.

3.6 Thermal Evaluation under HAC

3.6.1 HAC Fire

The applicant selected the F-32B fuel basket loaded with 32 fuel assemblies as the bounding pattern for the HAC thermal analysis and evaluated the HAC thermal performance of the HI-STAR 80 package using ANSYS FLUENT CFD code. The applicant presented the PCT and the maximum package component temperatures of the F-32B fuel basket in Table 3.1.3.

The F-32B fuel basket is the bounding pattern for the HAC thermal analysis because of its maximum heat load of 54.0 kW and the maximum fuel cladding and package component temperatures under NCT, as compared to the F-12P basket and other configurations.

The applicant noted in Section 3.4.2, "Fire Conditions," that the thermal gaps between the outer shield cylinder (OSC) made of copper chromium and the intermediate shell made of carbon steel are first ignored during the 30-minute fire transient to increase the inflow of heat from the fire to the inside and then are assumed to resume, during the post-fire cooldown, to decrease the outflow of heat from inside to the ambient.

The staff confirmed that the modeling of the air gaps under a 30-minute fire transient and its post-fire cooldown, as described by the applicant in Section 3.4.2, will increase the calculated temperatures of the fuel cladding and package components, and therefore the modeling of the air gaps between the OSC and the intermediate shell is acceptable.

For the HAC thermal evaluations, instead of natural convection simulated in NCT, the transfer of heat from the fire source to the package comes from a combination of radiation with a minimum fire emissivity (0.9) and a lower-bound package absorptivity (0.8) and forced convection with heat transfer coefficient of 25.5 W/m^2-K during the 30-minute HAC fire with solar insolation. During the post-fire cooldown, the transfer of heat between the package and the ambient combines radiation with a package surface emissivity of 0.66, natural convection and solar insolation.

The staff reviewed the initial conditions, fire conditions, and the model for the HAC thermal evaluation, and confirmed that the initial conditions and fire conditions used in the HAC thermal evaluation are acceptable as they are in accordance with the guidance in NUREG-1617.

The staff reviewed the maximum fuel cladding and package component temperature under the HAC fire and confirmed that the maximum temperatures of the fuel cladding and containment components (e.g., containment shell, closure lid, containment seals) are below the allowable limits under the HAC fire.

The staff also verified that there is no lead melting with a maximum HAC temperature of 588°F, which is below the melting point of 620°F when the outer shield cylinder made of cooper chromium is used to contain lead. The staff checked assumptions, boundary conditions, and component/material properties, and heat transfer parameters used in the applicant’s thermal model, and the fuel cladding and package component temperatures calculated by the applicant’s thermal model.

The staff confirmed that the Model No. HI-STAR 80 package meets the HAC thermal requirements, in compliance with 10 CFR 71.73.

3.6.2 Fuel Reconfiguration under HAC

The applicant performed thermal analyses for fuel reconfiguration under HAC: with a 100% fuel failure postulated under HAC, the applicant performed thermal analyses by assuming that 100% of the design heat load is uniformly distributed in 75% of the original fuel length and the fuel is shifted to the package inner lid to maximize the predicted temperatures of the seals which constitute part of the containment boundary.

The applicant reported the HAC maximum fuel cladding and package component temperatures, due to fuel reconfiguration, in Table 3.4.4.

The staff verified that (a) the PCT and the maximum package component temperatures, including the containment seals, are below their respective HAC temperature limits as reported in Table 3.2.10 and (b) the cavity pressure and the inter-lid space pressure under HAC are bounded by the HAC pressure limit of 176.2 psia (Table 3.1.4).

3.7 Evaluation Findings

According to the statements and representations in the application, the staff concludes that the thermal design has been adequately described and evaluated, and both NCT and HAC thermal evaluations of the Model No. HI-STAR 80 package meet the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT REVIEW

The objective of the review is to verify that the containment performance of the Model No. HI-STAR 80 package has been adequately evaluated for the tests specified under both NCT and HAC, and that the package design satisfies the containment requirements of 10 CFR Part 71.

4.1 Description of the Containment System

The containment system for the Model No. HI-STAR 80 package consists of the following components: shell, upper and lower forgings, inner closure lid (including inner seal and leak test port with plug and seal), outer closure lid (including inner seal and test plug seal), vent and drain ports (including inner port cover plate, port inner seals and port outer seal), and spray cooling port (including cover plates and seals).

The staff reviewed Section 4.1, "Description of Containment System," and confirmed that a complete description of the containment boundary is clearly depicted in Figure 4.1.1.

4.1.1 Containment Vessel

The containment vessel is represented by the containment shell, containment lower forging, containment upper forging, and inner and outer closure lids and no valve or pressure relief device is specified on the HI-STAR 80 containment system. The staff reviewed Section 4.1.1 and Figure 4.1.1 and confirmed that (a) the containment vessel is the primary containment system, which creates an enclosed cylindrical cavity for the containment of the enclosed radiological contents, and (b) there is no pressure relief valve on the containment vessel of the HI-STAR 80 package.

4.1.2 Containment Penetrations

The containment system penetrations include the spray cooling port, the outer closure lid test plug, the inner closure lid leak test port and the vent and drain ports, with elastomeric seals in each penetration. The containment penetrations are designed and tested to ensure that permitted activity release limits, specified in 10 CFR 71.51, will not be exceeded.

The staff reviewed the discussion of the containment penetration provided by the applicant, in Section 4.1.2 "Containment Penetrations," and verified that all containment boundary penetrations, and their methods of closure, are described in detail.

4.1.3 Seals and Welds

4.1.3.1 Containment Seals

The applicant stated that the containment seals are designed and fabricated to meet the design requirements specified in Chapter 2 of the application. The containment seals include the following:

(a) an inner seal in the inner closure lid,
(b) an inner seal in the outer closure lid,
(c) two inner seals in the vent port (one is located between the forging and the bushing and another between the bushing and the bronze plug),

(d) two inner seals in the drain port (one is located between the forging and the bushing and another between the bushing and the bronze plug), and

(e) a cap inner seal and a cover plate inner seal in the spray cooling port.

The applicant stated that all of the inner closure lid, the outer closure lid, and the vent and drain ports contain the inner seal(s) and outer seal. The inner seal is the containment seal and the outer seal provides redundant closure. The inner closure lid containment boundary and redundant boundary sealing surfaces are not subject to corrosion due to the presence of the outer closure lid and inter-lid cavity helium backfill.

The applicant also stated that the outer closure lid containment sealing surfaces are not subject to corrosion due to the presence of redundant closure features that prevent exposure to the environment external to the package. The seal materials of construction are highly corrosion resistant.

The staff reviewed the information in Sections 4.1.3.1.1 and 4.1.3.1.2, as well as the engineering drawings provided by the applicant. The staff agrees that the containment seals are highly corrosion resistant due to the presence of (a) redundant closure features that prevent exposure to the ambient air, and (b) the helium backfill that provides an inert environment in the inter-lid cavity. Based on those redundant closure features, the staff determined that the resistance of the containment seals to corrosion, as described in Section 4.1.3.1, is appropriate and acceptable.

4.1.3.2 Containment Welds

The applicant stated that the cask containment system welds consist of full penetration welds forming the containment shell, the full penetration weld connecting the containment shell to the containment upper forging, and the full penetration weld connecting the containment lower forging to the containment shell. All containment system boundary welds are fabricated and inspected in accordance with ASME Code Section III, Subsection NB.

The staff reviewed Section 4.1.3.2 and finds that the containment system welds are adequately shown in the drawings. The staff confirmed that all containment system boundary welds would be fabricated and inspected in accordance with ASME Code Section III, Subsection NB, as described in Chapter 2, “Structural Evaluation.”

4.2 Containment under NCT

The applicant stated that the HI-STAR 80 package containment components are maintained below their peak temperature and pressure limits under NCT; therefore, the design basis leakage rate will not be exceeded during NCT, as defined in 10 CFR Part 71.

The applicant also stated that the HI-STAR 80 package is leaktight for all containment system leakage tests, when loaded with spent fuel, and has an allowable leakage rate for all containment system leakage tests when loaded with non-fuel waste.

The staff reviewed Tables 3.1.1.A, 3.1.1.B, 3.1.2.A and 3.1.2.B and verified that:

(a) the package cavity MNOPs of both the F-12P and F-32B baskets, with less than 3% rod rupture, are below the bounding package cavity MNOP of 94.7 psia,

(b) the maximum pressures in the inter-lid space for both the F-12P and F-32B baskets are below the bounding pressure of 34.7 psia, and

(c) the peak containment seal and component temperatures for both the F-12P and F-32B baskets are below the allowable NCT limits, as shown in Tables 3.2.11 and 3.2.12.

Therefore, the staff confirms that the containment effectiveness of the containment seals is maintained under NCT, and the HI-STAR 80 package meets the containment requirements in compliance with 10 CFR 71.71 and 71.51(a)(1).

The staff reviewed Section 8.1.4, "Leakage Tests" of the application, and verified that the leakage rate testing on the cask containment system is to be performed in accordance with ANSI N14.5 – 2014.

As such, the leakage rate testing procedures shall be approved by an ASNT Level III specialist, and leakage rate testing shall be performed (i) by personnel who is qualified and certified in accordance with the requirements of SNT-TC-1A and (ii) in accordance with a written quality assurance program.

4.3 Containment under HAC

The applicant stated that the HI-STAR 80 package containment components are maintained below their peak temperature and pressure limits under HAC; therefore, the design basis leakage rate will not be exceeded during HAC, as defined in 10 CFR Part 71. The applicant also stated, in Section 4.2.1 "Containment Criteria," that the HI-STAR 80 package is leaktight for all containment system leak tests when loaded with spent fuel and has an allowable leakage rates for all containment system leakage tests when loaded with NFW.

The staff reviewed Table 3.1.3 for the PCT and maximum component temperatures under HAC with no fuel reconfiguration, Table 3.1.4 for the maximum cavity pressure with 100% fuel rods rupture under HAC, and Table 3.4.4 for the PCT and maximum component temperatures for fuel reconfiguration under HAC for both the F-12P and F-32B baskets.

The staff verified that, for both the F-12P and F-32B baskets, (a) the maximum package cavity pressures under 100% fuel rod rupture, are below the HAC limit of 214.7 psia and (b) all containment components have peak temperatures below the required HAC limits shown in Tables 3.2.10 and 3.2.12.

Therefore, the staff confirms that the containment effectiveness of the containment components, including containment seals, is maintained under HAC and the Model No. HI-STAR 80 package complies with 10 CFR 71.73(c)(4) and 71.51(a)(2).

4.4 Calculation of Allowable Leakage Rate

4.4.1 Leakage Rate Tests for HI-STAR 80 Used as Fuel Package or NFW Package

(a) Fabrication Leakage Rate Test

The applicant stated that:

(i) the fabrication leakage rate test demonstrates that the containment system, as fabricated, provides the required level of containment, and

(ii) the entire containment boundary, including base material, welds, seals, closures, valves, etc., will be leak tested during the fabrication process.

As such, this will demonstrate that the containment system, as fabricated, provides the required level of containment, in accordance with ANSI N14.5 (2014).

The applicant specified the allowable leakage rates in Table 8.1.1 and the type of fabrication leakage rate test for each containment component in Table 8.1.2 for both a spent fuel package and a NFW package, in accordance with ANSI N14.5 (2014).

The staff reviewed Section 4.4.1 and agrees on the test types, for each containment component shown, and the fabrication leakage rates of 1×10^{-7} ref-cm^3/sec (leak-tight) on the entire containment boundary of the HI-STAR 80, when used as a spent fuel transport package and 5×10^{-7} ref-cm^3/sec for the entire containment boundary of the HI-STAR 80, when used as a NFW transport package, as shown in Tables 8.1.1 and 8.1.2, to meet the test criteria, consistent with ANSI N14.5 (2014).

The staff confirmed that the fabrication leakage rate testing should be performed on the entire containment boundary, and the package should be leak tested to a leak-tight criterion, when used to transport spent nuclear fuel, and be leak tested to a criterion of $\leq 5 \times 10^{-7}$ ref-cm^3/sec, when used to transport NFW.

(b) Pre-shipment Leakage Rate Test

The applicant stated that the pre-shipment leakage rate test demonstrates that the containment system closure has been properly performed, and is performed by the user before each shipment, after the contents are loaded and the containment system is assembled, and that the pre-shipment leakage rate test remains valid for one year.

The applicant specified the allowable pre-shipment leakage rate for each containment component in Table 8.1.1 and the type of leakage rate test in Table 8.1.2, for both cases of the package transporting either spent fuel or NFW, in accordance with ANSI N14.5 (2014).

The staff reviewed Section 4.4.1 and finds acceptable the allowable pre-shipment leakage rate and the test type for each containment component shown in Tables 8.1.1 and 8.1.2, respectively, for the HI-STAR 80 used as a spent fuel package or NFW package.

(c) Periodic Leakage Rate Test and Maintenance Leakage Rate Test

The applicant stated that (a) the periodic leakage rate test demonstrates that the containment system closure capabilities have not deteriorated over an extended period of use, and is performed by the user before each shipment if the previous leakage rate test has expired, and (b) the periodic leakage rate test remains valid for one year. The applicant specified the allowable leakage rate and the test method for each containment component in Table 8.1.1 and Table 8.1.2, respectively.

The applicant stated also that the maintenance leakage rate test demonstrates that the containment system provides the required level of containment after undergoing maintenance, repair, and/or containment component replacement; and shall be performed prior to returning package to service.

The staff reviewed Sections 4.4.3 and 4.4.4 and finds that the allowable leakage rates and the types of leakage rate testing for the periodic leakage rate test and maintenance leakage rate test, shown in Table 8.1.1 and Table 8.1.2 of the application, are acceptable to meet the criteria of leakage tests, consistent with ANSI N14.5 (2014).

4.4.2 HI-STAR 80 Package Used as a NFW Package

The applicant presented the assumptions used in determining the allowable leakage rates for the specific NFW basket (NFWB-1) in Section 4.5.1, "Assumptions," and the analysis and results in Section 4.5.2, "Analysis and Results." As stated in Section 4.5, "Requirements for Normal and Hypothetical conditions of NFWB-1 Transport," the allowable leakage rates for NCT and HAC are determined in accordance with the requirements of ANSI N14.5 (2014) and the guidelines of NUREG/CR-6487, "Containment Analysis for Type B Packages Used to Transport Various Contents."

The applicant noted that the non-fuel inventory for HI-STAR 80 is limited to irradiated non-fissile core components in solid form; the waste is considered as non-dispersible solids with radioactive surface contamination; the contents are non-fissile or fissile exempted and is mainly consisting of metal or ceramics; and the radioactive gases are not associated with the waste contents and considered negligible.

The applicant presented the isotopic inventory for the non-fuel inventory in Table 4.5.3, the parameters from NUREG/CR-6487 used in the source term calculations in Table 4.5.4, and the total source term results in Table 4.5.5 of the application.

After reviewing the assumptions provided in Section 4.5.1 "Assumptions," the sources terms described in Section 4.5.2.1 "Source Terms for NFWB-1," and NUREG/CR-6487, the staff agrees that:

(a) the source-terms from releasable activity arise from surface CRUD,

(b) the containment analysis for the HI-STAR 80 assumed non-dispersible solids with no fines made of the bulk radioactive material,

(c) the radioactive gases can be neglected if present because they are not associated with waste contents, and

(d) the assumed surface area of the activated core components, used in the calculation, is greater than the surface area of 13 typical PWR fuel assemblies taken from NUREG/CR-6487.

Based on the source term of the non-dispersible solids and releasable activity described in NUREG/CR-6487, the staff determined that the assumptions and parameters used in leakage-rate calculations are acceptable.

The applicant performed the containment analysis and calculated the allowable leakage rates of 5.18×10^{-7} ref-cm^3/sec for NCT and 1.90×10^{-4} ref-cm^3/sec for HAC (Table 4.5.8) for the HI-STAR 80 package loaded with NFWB-1, as shown in Section 4.5.2 "Analysis and Results," and Report HI-2167204 "Containment Analysis for HI-STAR 80."

In Report HI-2167204, the applicant used the methodology described in ANSI N14.5 and the source terms of NFWB-1 to calculate the release rates and show they are below the allowable release rates, in compliance with 10 CFR 71.51(a)(1) under NCT and 10 CFR 71.51(a)(2) under HAC, respectively.

After reviewing the methodology and calculations provided in Section 4.5 "Requirements for Normal and Hypothetical Accident Conditions of NFWB-1 Transport," and Table 4.5.8, and Holtec's Report HI-2167204, the staff confirmed that (a) the methodology and calculations are consistent with ANSI N14.5 (2014) and are acceptable and (b) calculated volumetric leakage rates of 5.18×10^{-7} ref-cm^3/s (NCT) and 1.90×10^{-4} ref-cm^3/s (HAC) are less than 10^{-6} A_2 per hour for NCT and less than A_2 per week for HAC.

The staff determined that:

(a) the limiting leakage rate acceptance criteria of 5.0×10^{-7} ref-cm^3/sec and its sensitivity of 2.5×10^{-7} ref-cm^3/sec, selected by the applicant, are acceptable for HI-STAR 80 package loaded with NFW (NFWB-1),

(b) the calculated leakage rates of HI-STAR 80 loaded with NFW (NFWB-1) do not exceed the allowable leakage rates for NCT and HAC, respectively, in compliance with 10 CFR 71.51(a)(1) and 71.51(a)(2), and

(c) the measured leakage rates for individually tested components shall be summed when the HI-STAR 80, loaded with NFW, is not leaktight.

4.5 Evaluation Findings

The staff reviewed the containment features section of the Model No. HI-STAR 80 package and concludes that: (1) the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71, and (2) the package is leak tested to the leak-tight criterion when spent fuel assemblies are loaded and (3) the package has a leakage rate below the allowable limits when NFW is loaded.

The Model No. HI-STAR 80 package meets the requirements of 10 CFR 71.51(a)(1) for NCT and 10 CFR 71.51(a)(2) for HAC.

5.0 SHIELDING REVIEW

The package is designed for exclusive use, with proposed contents of spent UO_2 PWR fuel, and spent UO_2 or MOX BWR fuel, as well as non-fuel wastes contained in the NFWB-1, a non-fuel waste basket. These fuels all have short cooling time and high burnup. The fuel baskets, loaded directly into the cask overpack, are made of METAMIC-HT. The NFWB-1 has an optional secondary container, called the Plant Specific Stainless Steel Core Component Cassette or CCC-1, that can be used with the NFWB-1. The CCC-1 is not credited for its shielding within the evaluations to calculate dose rates.

The Model No. HI-STAR 80 package contains a personnel barrier for purposes of limiting access to the surface areas of the package that have temperatures above 50°C. However, the personnel barrier is not considered a structural component of the package and the applicant has not credited its presence in the dose rate evaluations.

The HI-STAR 80 has two impact limiters installed at the axial ends of the package, credited to some extent for the dose rate evaluations under both NCT and HAC, as prescribed in 10 CFR 71.71 and 71.73 respectively.

5.1 Description of the Shielding Design

5.1.1 Packaging Design Features

The staff reviewed the general information chapter in the application, including the drawings in Section 1.3, and determined that the applicant specified all dimensions and tolerances of all components considered within the shielding evaluation. The staff also reviewed the additional information on the shielding design in Chapter 5 of the application and determined that all figures, drawings, and tables describing the shielding features are sufficiently detailed to support an in-depth technical evaluation.

The shielding design features of the HI-STAR 80 package include the fuel basket and overpack structures:

> The overpack includes the two lids, the concentric layers of steel, lead, steel and neutron shield and the heat rejection fin that form the body, lower forging, and the central steel structures of the impact limiters.
>
> The gamma shielding is mostly performed by the cask body steel, lead and copper shells, the lids, and lower forging.
>
> The neutron shielding is mostly performed by the Holtite, a borated polymer material neutron absorber embedded in those parts and in the radial direction. The neutron absorber is located outside of the lead region and inside of the copper enclosure shell. The neutron shield has copper structural ribs that are angled to provide additional gamma shielding.

5.1.2 Summary Table of Maximum Radiation Levels

For spent fuel, the applicant analyzed each burnup and cooling time combination required for the loading patterns specified in Appendix 7.D of the application to ensure that they meet regulatory dose rate limits. In Tables 5.1.1 through 5.1.8 of the application, the applicant displayed the results of the bounding fuel loadings that produce the highest dose rates at the surface, at 2 m under NCT, and at 1 m under HAC.

The applicant additionally calculated the distance for the dose rate limit for the normally occupied space. These tables include dose rate contributions from neutrons, gammas, fuel assembly hardware, n-gamma reactions and control rod assemblies (CRAs) (applicable to PWR fuel packages only), and include the accumulated contributions to the external dose rates from all factors that increase dose rates within the application, namely source term input uncertainties, irradiated stainless steel rods, manufacturing tolerances on the materials, and axial burnup and void distributions, for both NCT and HAC.

The applicant calculated the maximum dose rates for the NFWB-1 and showed the results in Tables 5.1.9 through 5.1.11 of the application. The tables show maximum dose rates at the surface and at 2 m under NCT, and at 1 m under HAC.

The staff reviewed Tables 5.1.1 through 5.1.11 of the application and found that the calculated dose rates meet the requirements in 10 CFR 71.47 and 10 CFR 71.51. Since the applicant states that the HI-STAR 80 will be operated under “exclusive use,” the staff verified that the evaluated radiation levels do not exceed those specified in 10 CFR 71.47(b).

The staff verified that the limit of 200 mrem/h will not be exceeded on the external surface of the package. The maximum calculated surface dose rate for the HI-STAR 80 is 177.4 mrem/h for the F-32B (BWR fuel basket) at the side adjacent to the lower forgings, and 190 mrem/hr for the NFWB-1. This meets the regulatory limits in 10 CFR 71.47(b)(1).

The applicant showed that calculated dose rates at the outer package surface are less than 200 mrem/h, therefore in compliance with 10 CFR 71.47(b)(2) which requires that the dose rate be limited to 200 mrem/h at the vehicle surface.

The staff verified that the limit of 10 mrem/h will not be exceeded at any point 2 meters from the outer lateral surface of the vehicle. The maximum calculated dose rate at this location was 9.1 mrem/h at the side of the package for the F-32B (BWR fuel basket). The staff finds that this meets the requirement in 10 CFR 71.47(b)(3).

In Tables 5.1.5 and 5.1.6 of the application, the applicant shows the calculated distance from the package necessary to comply with the 2 mrem/hr limit in the regulation 10 CFR 71.47(b)(4) for normally occupied space. The staff found that this meets the regulation by requiring personnel to wear radiation dosimetry at this distance or closer. For a package loaded with the NFWB-1 basket, the applicant shows that the dose rates at 2 meters from the package are all below 2 mrem/hr; therefore, the package is in compliance with 10 CFR 71.47(b)(4) at this distance.

The staff verified that the radiation dose rate under HAC does not exceed 1 rem/h at 1 meter from anywhere on the external surface of the package. The staff finds that this meets a requirement of 10 CFR 71.51(a)(2).

5.2 Source Specification

The applicant specifies the allowable fuel assembly parameters in Appendix 7.D. There are a wide range of allowable spent fuel PWR, BWR and BWR-MOX assemblies with fuel assembly limits in Table 7.D.1 of the application. Damaged fuel assemblies are not authorized contents for the Model No. HI-STAR 80 package.

The authorized PWR spent fuel assemblies are 15x15 and 17x17 arrays. The PWR spent fuel assemblies must be loaded within the F-12P basket model. The authorized PWR spent fuel assembly characteristics are described in Table 7.D.2 of the application. Control rods are authorized for transport within spent PWR fuel assemblies. Fuel assemblies may contain up to 4 irradiated stainless steel replacement rods. This is discussed in further detail in Section 5.2.3.1 of this SER.

Spent PWR fuel assemblies may be loaded in any location within the F-12P basket with the exception of fuel assemblies with steel replacement rods, as discussed further in Section 5.2.3.1 of this SER. The assemblies are restricted to assembly burnup, enrichment, cooling time and minimum number of 1 year cycle requirements as specified in Table 7.D.4 of the application. This table has two sub-tables, one for a full basket and one for up to 10 assemblies where there are two empty cells in the center of the basket. There are minimum cooling times within this table that are less than one year; however, Note 2 of the Table states that all SNF must have a minimum cooling time of 1 year per 10 CFR 71.4.

The authorized BWR spent fuel assemblies are UO2 in 8x8, 9x9, 10x10 and 11x11 array sizes, and a MOX assembly type in a 10x10 array size. The authorized BWR spent fuel assembly characteristics are described in Table 7.D.3 of the application.

Non-fuel hardware are not authorized contents with spent BWR fuel assemblies. BWR spent fuel assemblies may contain up to 4 irradiated stainless steel replacement rods as discussed further in Section 5.2.3.1 of this SER.

Spent BWR fuel assemblies may be loaded in any location within the F-32B basket with the exception of fuel assemblies with steel replacement rods, as discussed in Section 5.2.3.1 of this SER. The assemblies are restricted to assembly burnup, enrichment, cooling time and minimum number of 1 year irradiation cycle requirements as specified in Table 7.D.5 of the application. Note 6 to Table 7.D.1.II states that for BWR assemblies with axial blankets the minimum enrichment excludes that of the blankets. The staff found this acceptable since these areas are on the ends of the assemblies and have much lower source term.

This table includes three sub-tables: up to a full basket, up to 28 assemblies where there are 4 empty cells in the center of the basket, and up to 24 assemblies where there are 8 empty cells in the center of the basket. There are minimum cooling times within Table 7.D.5 of the application that are less than one year; however, Note 2 of the Table states that all SNF must have a minimum cooling time of 1 year per the definition in 10 CFR 71.4.

There are up to 4 MOX assemblies allowed with the F-32B basket. The applicant specified the maximum burnup, minimum cooling times and minimum number of 1 year irradiation cycles as well as minimum enrichment for the accompanying UO2 assemblies in Table 7.D.6 of the application. The MOX assemblies must be loaded within 4 specific locations on the interior of the basket. The MOX Pu enrichment vector is specified within Table 7.D.1 of the application.

The HI-STAR 80 is also authorized for non-fuel waste for transport within the NFWB-1. These components consist of mainly activated or contaminated metals or ceramics. Table 7.D.8 authorizes the following reactor-related non-fuel Waste ("Core Components") to be shipped within the NFWB-1: fuel channels, transition pieces, spacer grids, core grid components, core spray components, control rods or control blades, LPRM neutron monitors using fission chambers, and burnable absorbers.

5.2.1 Spent Fuel Source Term

As discussed in Section 5.2 of the application, the applicant calculated the gamma and neutron source term from radioactive fission products using the TRITON and ORIGAMI/ORIGEN modules of the SCALE 6.2.1 system using the ENDF/B-VII 252-group library. ORIGEN is considered acceptable to the staff per the guidance in Section 3 of NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages." TRITON is a transport theory based code, released after the publication of NUREG/CR-6802; however, it is mentioned in Section 3.2.6 of NUREG/CR-6802 as acceptable for this type of analysis. It represents more advanced and detailed 2-D reactor physics solution method compared to SAS2H (which is recommended and used in NUREG/CR-6802) and the staff found the use of this code acceptable for the HI-STAR 80. The applicant used TRITON to calculate the cross section libraries for the spent fuel so that the libraries match the specification for the fuel and operating conditions required for the HI-STAR 80, and did not use any of the predetermined libraries. The staff found this acceptable.

The applicant selected the 17x17S2 assembly as the design basis assembly for generating PWR spent fuel assembly source terms. This is based on this assembly having the highest UO_2 mass per assembly and UO_2 specific mass (mass per length), as shown in Table 5.2.17 of the application. The staff finds this assumption conservative and bounds all the proposed PWR fuels because the higher UO_2 mass and UO_2 specific mass should create the highest source term. The applicant evaluated two additional assembly types and discussed this in Section 5.4.9 of the application.

The applicant compared the dose rates under NCT and HAC and showed the results in Tables 5.4.24, 5.4.26 and 5.4.28 of the application. For most of the comparisons, the design basis assembly produced dose rates within the statistical uncertainty of the calculations; however, for NCT surface calculations near the lower forgings, the applicant found that there was a significant difference between that of the design basis fuel and the other two fuel assemblies and included this difference as an adjustment factor on the dose rates calculated at this location. There were other locations where the other fuel assemblies produced significantly higher dose rates than with the design basis fuel; however, the dose rates at these locations were so low that they are not close to being limiting and, therefore, no adjustment factors were included. Based on these reasons, the staff found that the applicant's selection of the design basis PWR assembly, including adjustment factors, is acceptable and would reasonably represent all PWR assemblies authorized for shipment within the HI-STAR 80.

The applicant selected the 10x10S2 assembly as the design basis assembly for generating BWR spent fuel source terms. This is based on this assembly having the highest UO_2 mass and the second highest UO_2 specific mass, as shown in Table 5.2.18 of the application. The higher UO_2 mass and UO_2 specific mass should create the highest source term. The applicant evaluated three additional assembly types with a comparable specific mass and discussed this in Section 5.4.9 of the application.

The applicant compared the dose rates under NCT and HAC at all locations around the package, at all distances required by the regulations, and showed the results in Tables 5.4.23, 5.4.25, and 5.4.27 of the application. For most of the comparisons, the design basis assembly produced dose rates within the statistical uncertainty of the calculations; however, for NCT and HAC surface calculations, near the upper forgings, the applicant found that there was a significant difference between that of the design basis fuel and the other three fuel types; therefore, it included this difference as an adjustment factor on the dose rates calculated at this location. Based on this discussion, the staff found that the applicant's selection of the design basis BWR assembly, including adjustment factors, identified the maximum dose rate and is acceptable and would reasonably represent all BWR assemblies authorized for shipment within the HI-STAR 80.

There is only one type of BWR-MOX assembly authorized for shipment, which is represented as the 10x10S2 in Table 7.D.3 of the application; so, there is no need to determine a representative assembly as this assembly can be modeled explicitly.

The staff found differences when comparing the design basis fuel assemblies used within the analysis, described in Tables 5.2.1 and 5.2.2 of the application, to the allowable specifications for spent fuel assemblies in Tables 7.D.2 and 7.D.3 of the application. The assemblies analyzed for the shielding evaluation use nominal dimensions and therefore do not match exactly the specifications in Tables 7.D.2 and 7.D.3 of the application. Thus, the applicant has added a mass and specific mass restrictions on all spent PWR and BWR assemblies.

The staff verified that the mass and specific mass in Tables 7.D.1 and 7.D.2 of the application do not exceed that of the assemblies that were used to produce the source term. The staff finds that the mass and specific mass restrictions for the UO_2 assemblies would prevent assemblies from being loaded that could produce a source term leading to dose rates higher than what was analyzed by the applicant as the heavy metal mass is the characteristic that most significantly affects the source term within a fuel assembly. The applicant did not include a mass nor specific mass restrictions for the MOX assemblies: the staff found it acceptable because there are no more than 4 MOX assemblies allowed within an F-32B basket, and that these assemblies will be loaded within the 4 cells in the intermediate zone and this receive shielding from the outer UO_2 assemblies.

The applicant stated that it analyzed all burnup, cooling time and enrichment combinations of spent fuel allowed for transport, as specified in Appendix 7.D. Due to the large number of source terms generated, it only included source terms for a few selected burnup, cooling time and enrichment combinations to display within the application. The applicant presented the total gamma source term for a BWR spent fuel assembly, PWR spent fuel assembly, and BWR MOX spent fuel assembly in Tables 5.2.4, 5.2.5 and 5.2.6 of the application, respectively. The applicant included gammas in the range of 0.45 MeV to 3.0 MeV. The staff found this acceptable as independent staff calculations show that gamma source strength from spent fuel above 3 MeV is at least one order of magnitude smaller than the gammas at or below 3 MeV, and gammas below 0.45 MeV would not penetrate the cask shielding. This conclusion is consistent with the principles of radiation absorption, as reflected in the flux-to-dose rate conversion factors for gammas, i.e., high energy gammas will not be absorbed by human organs as effectively as lower energy gammas and the gammas with energy below 0.45 MeV will not penetrate the layers of shielding materials of the package.

The applicant presented the total neutron source term for a BWR, PWR and BWR-MOX spent fuel assembly in Tables 5.2.11, 5.2.12, and 5.2.13, respectively, of the application. The staff reviewed the total gamma and neutron source for spent fuel and found that it is within the range typically seen for spent nuclear fuel and found it appropriate.

The power density affects the source term generated by the depletion analysis. If a spent fuel assembly is burned with a larger power density than what was assumed within the analysis, it would have a larger source term and the analysis would be non-conservative. The power density is directly related to the number of cycles because, if a fuel assembly achieves a certain burnup level in fewer cycles, then this means it was burned at a higher power density. Therefore, the applicant specified the minimum number of irradiation cycles as a loading requirement in Table 7.D.4 of the application for PWR fuel, 7.D.5 of the application for UO2 BWR fuel, and 7.D.6 of the application for MOX fuel and the UO2 BWR fuel that can be transported with the MOX fuel.

Table 5.2.1 of the application specifies the number of irradiation cycles used within the shielding evaluation for each burnup level for PWR and UO2 BWR and MOX BWR fuel. There are additional conditions (burnup, enrichment, cooling time, and minimum number of irradiation cycles) that are allowed for loading under Tables 7.D.4 and 7.D.5 of the application that are not listed in Table 5.2.1 of the application. The applicant discusses these aspects in Section 5.4.7.2 of the application by stating that higher cooling time requirements are established for assemblies that reach their burnup within a smaller number of cycles.

The applicant displayed the results of the additional calculations to cover the additional conditions in Tables 5.4.37 and 5.4.38 of the application which show that the dose rates are comparable to that of the design basis values from Table 5.2.1 of the application; therefore, the staff finds the additional conditions represented by burnup, enrichment, cooling time and minimum number of cycles, acceptable for the Model No. HI-STAR 80 package.

There is only one set of conditions in Table 7.D.6 of the application for MOX fuel and the staff confirmed that they are equivalent to the conditions specified in Table 5.2.1 of the application and therefore found this acceptable.

The staff found the downtime assumed between cycles of 15 days is reasonable, as seen for a typical refueling operation.

5.2.1.1 Spent Fuel Source Term Input Uncertainties

(a) Uncertainty Related to Depletion Operating Parameters

To perform the fuel depletion analysis, the applicant varied the reactor operating parameters, stated in Table 5.F.1 of the application, to determine if a change to the individual operating parameters has an appreciable effect on dose rates. The applicant showed the results of varying these parameters on dose rates in Table 5.F.2 of the application. The applicant calculated the changes in dose rates at 2 meters from the side of the package under NCT and at 1 meter from the surface of the package under HAC as these locations exhibit the smallest margin to the regulatory dose rate limits.

The applicant varied the coolant temperature, coolant density, fuel temperature, soluble boron concentration (PWR), fuel density, and studied the effects of control blade insertion (BWR) and gadolinia rods (BWR).

To determine if the nominal value and the range in which the parameter was varied was reasonable, the staff compared the values used by the applicant to those found in Table B.1 in NUREG/CR-6802 and to values found throughout NUREG/CR-7240, “Impact of Operating Parameters on Extended BWR Burnup Credit.” The staff found that the range of values used in Table 5.F.1 of the application does not encompass the range of all parameters discussed in the above references; however, the resulting changes in external dose rates due to these parameter variations, displayed in Table 5.F.2, show mostly insignificant changes to dose rates and, therefore, the staff finds that increasing the range of variation any further would have only insignificant changes on dose rates and staff found the selected depletion parameters in Table 5.F.1 of the application acceptable. The staff found that the direction and magnitude of the changes on dose rates are consistent with its previous experience.

The only parameter whose variation resulted in a more than 1% increase in dose rates is coolant density. For PWR fuel, the applicant's calculations show that the largest difference is below the available margin to the regulatory limit, as shown in Tables 5.1.2, 5.1.4 and 5.1.8 of the application. There is a larger change in dose rates due to the variation of the coolant density for BWR fuel and this is discussed in the following section of this SER on axial burnup profiles.

The applicant did not perform any evaluations of uncertainty related to depletion parameters for MOX fuel. Since MOX is limited to only 4 assemblies per package, and receives shielding from the surrounding UO2 assemblies, the contribution from MOX fuel to dose rates is less significant.

(b) Uncertainty Related to Axial Burnup Profile

The applicant uses the axial burnup distribution from Table 1.2.6 of the application when generating source terms. For each axial section, the applicant inputs the source term as calculated at the adjusted burnup for that axial node. In Appendix 5.E of the application, the applicant justifies the use of the axial burnup profiles from Table 1.2.6 of the application. For conditions (burnup range and assembly specific characteristics) where the applicant could not justify that the axial burnup distribution from Table 1.2.6 of the application was a reasonable representation, the applicant derived a penalty, which has been included as an increase in the source terms used to generate allowable loading limits for the applicable conditions. The applicant notes that the analyses performed within this Appendix use an earlier design version of the HI-STAR 80 that used aluminum to contain the neutron absorber. The staff found that, since these evaluations are used to perform sensitivity studies and the previous version of the HI-STAR 80 package had very similar shielding performances, it is acceptable to use such analyses for these studies.

The applicant examined the effects of burnup profile on dose rate by comparing the dose rates evaluated using the design basis burnup profile to the dose rates evaluated using other burnup profiles. The applicant performed this comparison by dividing up the assemblies into burnup ranges and categories, discussed in Section 5.E.0.3 and Section 5.E.0.4 of the application, respectively. The applicant determined these categories based on assembly specific characteristics that would cause an assembly to have a unique or more limiting burnup profile, such as blanketed vs. non-blanketed fuel and exposure to ASPRs. For each category, the applicant compared the appropriate burnup profiles to that of the design basis burnup profile. For categories where the applicable burnup profile produced higher dose rates than the design basis burnup profile, the applicant applies a dose rate penalty, when determining loading limits for the fuel assemblies within the category.

(c) PWR Burnup Profiles

The applicant used burnup profile data for PWR spent fuel from commercial reactor critical state points and data from YAEC-1937, “Axial Burnup Profile Database for Pressurized Water Reactors,” May 1997, which the NRC has recognized as a source of representative data that can be used for establishing profiles to use in the licensing basis safety analysis in SFST-ISG-8, Rev. 3, “Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks.”

To account for blanketed PWR fuel, the applicant showed that the standard burnup profile (for non-blanketed fuel) bounded the blanketed fuel as long as the average burnup of the non-blanketed fuel is calculated without the blanketed nodes. The applicant included a condition in Note 8 of Table 7.D.1 that states that assembly average burnup for blanketed assemblies has to be calculated excluding the blankets. This would cause the assembly to be treated as a higher burnup assembly, which is conservative and therefore acceptable to the staff.

For all other categories of PWR assemblies, the applicant discussed in Section 5.E.1.1.2 of the application that the results from the different profiles produce essentially identical results to that of the design basis reference cases and that the design basis standard burnup profile is adequate for all PWR assemblies. This is shown in Table 5.E.1 of the application. Based on the applicant's results, the staff found the use of the burnup profile in Table 1.2.6 of the application acceptable for all PWR assemblies, conditions and burnup ranges.

(d) BWR Burnup Profiles

For BWR spent fuel, there is no equivalent set of burnup profiles; therefore, the applicant used data from BWR commercial reactor criticals, and some from NUREG/CR-7224, "Axial Moderator Density Distributions, Control Blade Usage, and Axial Burnup Distributions for Extended BWR Burnup Credit," August 2016 (ADAMS Accession No. ML16237A100) which was work performed under NRC contract to investigate BWR burnup credit. NUREG/CR-7224 also recognizes the lack of publicly available data for BWR burnup profiles. Therefore based on staff's engineering judgment, the conservative nature of this evaluation, and the knowledge that the burnup profile for the categories of fuel, as determined by the applicant, would have similar characteristics (also discussed in NUREG/CR-7224), the staff determined that the burnup profiles used as a basis of comparison to the design basis burnup profile are acceptable.

The staff also took into consideration the fact that the applicant did not have any burnup profiles over 45 GWd/MTU. Fuel burned longer tends to have a flatter burnup profile (less peaking) which would distribute the source more uniformly. The applicant used a flattened profile for the BWR case at 70 GWd/MTU so that there would not be excessive burnup in the peak node. In the applicant's model at 70 GWd/MTU, the peak node is limited to about 80 GWd/MTU. The staff found this to be realistic and acceptable.

All of the BWR profiles contain axial blankets. As seen from the results for PWR fuel, this is conservative over an assembly that does not have axial blankets and is therefore found acceptable by the staff.

The results in Table 5.E.2 of the application show that, when the applicant considers the more limiting profiles, the highest dose rate increase for any category is about 15% for the lowest burnup range. The applicant did not state that it included any uncertainty in its dose rate evaluations to account for this increase and the staff found this to be non-conservative. However the staff still found this acceptable for this application of the Model No. HI-STAR 80 package, because this is applicable only to the lowest burnup group, which also would produce the lowest source term. The staff also considered the margin to the limit for the calculated dose rates for BWR fuel shown in Tables 5.1.1, 5.1.3, 5.1.5 and 5.1.7 of the application, which shows about a 10% margin to the limits.

The applicant did not perform any studies to justify the use of the burnup profile for MOX fuel. The neutron source term is less strongly correlated to burnup as compared to UO_2 according to Figure 5.4.1 of the application and staff calculations, because the depletion of MOX fuel produces less actinides [Yoshihira ANDO and Hideki TAKANO, "Estimation of LWR Spent Fuel Composition," Japan Atomic Energy Research Inst., Tokai, Ibaraki (Japan). Tokai Research Establishment), Japan Atomic Energy Research Inst., Tokyo (Japan), 1999]. Also the 4 MOX assemblies, allowed within the HI-STAR 80, must be loaded within the cells in the intermediate regions of the fuel basket, and the source from these assemblies will be shielded by the surrounding UO_2 assemblies. Thus, the uncertainties related to the source term will have less of an impact on dose rates.

(e) Uncertainty Related to Axial Void Fraction Profile (BWR Fuel)

For BWR spent fuel, the applicant depletes the fuel assuming a uniform void fraction. The applicant compared the dose rate results generated using the uniform void fraction to that from dose rates evaluated using an axially varying profile from data from NUREG/CR-7224. The applicant found that the results were nearly identical, as shown in Table 5.E.3 of the application, and therefore did not add any kind of uncertainty or penalty to the dose rates evaluated. Based on the applicant's results, the staff found acceptable the applicant's use of a uniform axial void fraction, as well as the value used for the HI-STAR 80.

5.2.2 Spent Fuel Hardware Activated Metal

The primary sources of activity in the non-fuel regions of the fuel assemblies are from Co-60, which is created mainly by neutron activation of Co-59 in the steel and Inconel material. The applicant states, in Table 5.2.3 of the application, that it uses a Co-59 impurity level of 0.3 g/kg for the PWR fuel assemblies, and 0.5 g/kg Co-59 for the BWR and BWR-MOX assemblies. These values are requirements for maximum Co-59 impurity in Table 7.D.1 of the application for PWR and BWR fuel, respectively, and therefore its use within the shielding evaluations is acceptable to the staff.

The mass of the non-fuel regions of the assembly used for Co-60 activation calculations are also in Table 5.2.3 of the application. The staff used DOE/RW-0184, Volume 3 of 6, Appendix 2A, "Physical Descriptions of LWR Fuel Assemblies," December 1987, to verify the hardware masses for the different zones for the spent fuel hardware and found that the assumed values in Table 5.2.3 of the application are consistent and therefore acceptable to the staff.

The applicant calculated the activity of the Co-60 from the Co-59 activation using the ORIGAMI code and ENDF/B-VII cross sections using the in-core region flux at full power. The applicant modified the activity for each region using scaling factors listed in Table 5.2.7 of the application. The process of using the scaling factors is discussed in NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages," May 2003, (ADAMS Accession No. ML031330514). These scaling factors are from PNL-6906, "Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal," June 1989, which is referenced in Section 5.5.2.1 of NUREG-1617 and also in Section 3.3.2 of NUREG/CR-6802. On these bases, the staff found that the applicant's approach meets the acceptance criteria as specified in NUREG-1617 and is acceptable for determining source term from activated fuel hardware.

The applicant presented the total calculated Co-60 source from fuel hardware for BWR and PWR and BWR-MOX spent fuel assemblies in Tables 5.2.8, 5.2.9 and 5.2.10, respectively, of the application. The staff reviewed the tables and found that the magnitude of the source term is representative of the Co-60 source for fuel hardware and was thus acceptable.

5.2.3 Non-fuel Hardware

The only non-fuel hardware authorized for shipment with spent nuclear fuel assemblies are the PWR control rods which are allowed to be stored within PWR assemblies within the F-12 basket. The control rods are only allowed within the central four basket locations and must have a cooling time of at least 1 year. The control rods must be operated in a fully withdrawn position during full power operations and must have an absorber material of AgInCd with mass fractions of Ag, In, and Cd of 0.8, 0.15 and 0.05 respectively. These are reflected as requirements in Note 1 of Table 7.D.1 of the application and are thus acceptable as analysis assumptions, when evaluating the source term.

The applicant assumed the control rods were operated in a fully withdrawn position with respect to the active fuel region. Even when fully withdrawn, the bottom ends of the control rods are present within the upper portion (gas plenum and upper tie plate) of the fuel assembly and are activated. The applicant assumed the control rod cladding is stainless steel, with a Co-59 impurity of 1 g/kg (0.1%), consistent with impurity values in PNL-6906 and, thus, found acceptable by the staff.

The applicant calculated the source term of the control rods using TRITON and ORIGEN, then scaled them to the appropriate mass and neutron flux factors in Table 5.2.14 of the application. The applicant does not include a maximum permissible burnup of CRAs in Table 7.D.1 of the application. The staff found the burnup range from 0 to 360,000 MWd/MTU to be reasonably representative of the life of a CRA. The staff also found that the CRAs being limited to only the middle four assemblies would largely be shielded by the assemblies around them and any uncertainty in the source term, due to longer burnup times, would have an insignificant contribution to dose rates on the outside of the package.

The staff reviewed the values presented in Table 5.2.14 of the application and found the mass factors are reasonable as compared to the data in DOE/RW-0184, Volume 5 of 6, Appendix 2E, “Physical Descriptions of LWR Nonfuel Assembly Hardware,” December 1987. The flux factors are consistent with those used for the fuel hardware in Table 5.2.7 of the application, which the staff found acceptable, as they are consistent with those from PNL-6906.

The applicant shows the bounding source terms in Tables 5.2.15 and 5.2.16 of the application. The staff reviewed these tables and found the magnitude of the source term to be consistent with its previous experience and found them acceptable.

5.2.3.1 Irradiated Stainless Steel Rods

As discussed in Section 5.2 of this SER, the HI-STAR 80 package is authorized to ship fuel assemblies with irradiated stainless steel rods. These are steel rods that would replace a fuel rod that has failed in an assembly and the assembly is burned again within the reactor core, therefore irradiating the steel rod. These steel rods become activated producing a gamma source term primarily due to the production of Co-60. Although the contribution to dose rate may be less than that from an actual fuel rod, when determining the overall impact on dose rate the applicant should also consider the change in self-shielding when spent UO_2 is replaced with a steel rod.

PWR assemblies within the F-12P basket may contain up to 4 irradiated stainless steel replacement rods. Assemblies with stainless steel replacement rods must be located within the central 4 F-12P basket locations. Fuel assemblies with up to 25 irradiated stainless steel rods are permitted for the 17x17S1 assembly class but are limited to two assemblies within the F-12P basket and must be within the central 4 basket locations. These requirements are reflected as Note 4 to Table 7.D.1.I of the application.

BWR spent fuel assemblies within the F-32B basket may contain up to 4 irradiated stainless steel replacement rods. Assemblies with stainless steel replacement rods must be located within the central 12 basket locations of the F-32B basket. These requirements are reflected as Note 3 to Table 7.D.1.II of the application.

To justify the inclusion of irradiated stainless steel replacement rods, the applicant performed an analysis and discussed it in Section 5.4.6 of the application.

> For the BWR fuel in the F-32B basket, the applicant added Co-60 source from the equivalent cross sectional area of 4 rods to the inner 12 basket locations.
>
> Similarly for the PWR fuel in the F-12P basket, the applicant added the Co-60 source term from the equivalent area of 4 rods to the inner 4 basket locations to determine the effect on dose rates, except for the 17x17S1 assembly model where it added the Co-60 source term from the equivalent area for 25 rods to the inner 2 basket locations.

The applicant included the contribution to the dose rate evaluations used to determine loading limits, as summarized in Section 5.1 of the application.

To address the effect of changing the self-shielding provided by the spent UO_2 rods to that of stainless steel rods, the applicant performed a sensitivity study where it reduced the density of all of the fuel assemblies within the F-32B and F-12P to that of the stainless steel rods, and showed that there was a negligible change in the external dose rate. The staff therefore found that the applicant's analysis is a conservative representation of the stainless steel replacement rods and was adequately accounted for within the evaluation.

5.2.4 Non-Fuel Waste Content in the NFWB-1

The source term for the non-fuel waste content consists of activated metals and ceramics. The dominant source term is Co-60 from activated Co-59 present within the stainless steel; however, there are many other radioisotopes present from activation and contamination. The applicant states that it is not practical to specify a radionuclide content limit in terms of individual nuclides. The applicant has instead specified limits in terms of gamma energy. Each waste shipment will have different contents so the user will need to translate the isotopic composition of each payload into a gamma energy distribution. This can be done using SCALE or other similar programs.

The contents of the NFWB-1, as described in Table 7.D.8 of the application can be activated throughout or have contamination only on the surface. Therefore the applicant analyzed, as discussed in Section 5.4.8.4 of the application, and created loading limits for both a volume source and a surface source. The NFWB-1 allowable source energy and activity are located in Table 7.D.9 of the application. The source is described as a limit as either a mass specific (photons/sec/kg) or a surface specific source strength limit (photons/sec/m2) for each energy group, with a separate group for Co-60. This table also includes a total limit in photons/sec for each group, and Co-60, based on the total allowable mass of the content.

The contents of the NFWB-1 are primarily gamma emitters however there will be some small amount of neutron emitting material present as fission chambers and LPRMs are allowable contents as well as 2 grams of fissile material (including special nuclear material, SNM) and is expected primarily as crud or surface contamination on the surfaces of the activated metals. The applicant did not analyze the neutron source from this amount of fissile material; however, it is the staff's judgment that the neutron source from this amount of material would not significantly contribute to the dose rate outside of the package and therefore the staff found neglecting the neutron source from this material is acceptable.

5.3 Model Specification

The staff reviewed the structural and thermal evaluations to determine the effects of NCT and HAC on the packaging and its contents.

5.3.1 Normal Conditions of Transport

The model of the HI-STAR 80 package under NCT consists of the neutron shield and the impact limiter, neglecting the crush material. The applicant credits the dimensions of the impact limiter, i.e. it calculates dose rate for surface and at 2 meters in relation to the outer dimensions of the impact limiter cover plates in the axial direction. The staff found this acceptable because, under the NCT tests prescribed in 10 CFR 71.71, the package does not experience any damages that significantly affect the shielding of the package. Any deformities to the impact limiter due to the tests in 10 CFR 71.71 are inconsequential to the dose rate.

5.3.2 Hypothetical Accident Conditions

As discussed in Section 5.3 of the application, under HAC, the applicant's model takes into account the damage to the neutron shield and outer shell as a result of the design basis fire. The applicant models the neutron shield material as void. The staff found this conservative and acceptable.

There is lead slump in the lead components of the package under the 9-meter (30 foot) drop test. To account for this potential reduction of shielding, in the model for evaluating HAC dose rates the applicant replaced some of the radial lead, inner lid lead, and lead in the bottom forgings with void. The amount of lead removed from the top and bottom of the radial lead shield is consistent with the amount calculated within the simulations and shown in Table 2.7.4 of the application. For the lower lead forgings the applicant modeled this as a reduction in radial lead by half of the amount calculated in Table 2.7.4 of the application. For the lid, the applicant reduced the radius, but the staff could not verify the value as it was not evaluated in Chapter 2 of the application.

Although the staff is unable to determine if the representation of the lead slump within the dose rate analysis is conservative within the top lid and bottom forgings, the staff found it acceptable for this application based on the following reasons.

> For the bottom forgings, the slump calculated in Table 2.7.4 of the application would not manifest as a radial removal of lead and would likely be slumped in the geometry of a chord. Although removing the lead in the geometry of the chord using the slump

dimensions from Table 2.7.4 would result in a larger streaming path in a single location, reducing the radius of the entire shield by half, this amount actually results in more overall lead removed from the bottom forgings and at 1 meter from the surface (the location of the HAC dose rate limits), may result in a higher dose rate. There would be some contribution to the radial dose rate; however, the largest contribution would be to the bottom dose rate.

The bottom of the cask is not the limiting dose rate location from Tables 5.1.7 and 5.1.8 of the application. There is a significant margin (over one order of magnitude) to the limit at this location and the staff finds that its uncertainty in the applicant's lead slump assumptions would likely not result in dose rates increasing beyond the regulatory limit. In addition, the staff found it conservative that the applicant is replacing the lead with void, when in reality the lead would not be removed from the cask.

Similarly for the top lid, the staff does not have enough information to determine that the amount of the radial reduction in lead due to lead slump is conservative. However, the staff still found it acceptable for the HI-STAR 80 based on similar reasons it found the assumptions applied to the bottom forgings acceptable (margin to the limits, more overall lead removed). In addition the geometry of the top lid would produce less of a streaming path due to lead slump in this location because the lead extends further beyond the radius of the cavity than it does at the bottom.

The applicant did not perform an evaluation on the effects of lead slump when applied to the radius of the radial lead shield. The staff found that the uncertainty due to this effect is likely bounded by the effect of removing lead from the top and bottom of the radial shield and, if there is any increase in dose rate, there is enough margin to the HAC dose rate limits in Tables 5.1.7 and 5.1.8 of the application (roughly 15%) so that the dose limits would not be exceeded.

There is damage to the impact limiters as a result of the 9-meter (30 foot) drop. To account for this in the model for evaluating HAC dose rates, the applicant has excluded the crush material within the impact limiters and the outer skin surrounding the material. The applicant has retained the impact limiter backbone, which is directly attached to the cask. This is consistent with the structural analyses within Chapter 2 of the application; therefore, the staff found this assumption acceptable.

The applicant includes localized reduced lead thickness due to the puncture event. The amount of lead removed as a cylindrical reduction in lead thickness at a location located at approximately the mid-plane of the fuel. The amount of lead removed locally is shown in Table 5.3.7a of the application and is reasonable when compared to the amount of penetration calculated in Table 2.7.4 of the application given the conservative nature of the model, i.e. removing lead rather than displacing it, and the staff found it acceptable. The staff also found the location of the penetration to be reasonable and found it acceptable.

The results from Chapter 2 show that the F-12P and F-32B baskets remain largely unaltered; therefore, the applicant did not include any changes to the model of the baskets and the staff found this acceptable.

The NFWB-1 content does not take credit for any shoring of contents and therefore the NCT and HAC models of the contents are the same. The staff found this acceptable.

5.3.3 Configuration of Source and Shielding

As discussed in Section 5.4.1 of this SER, the applicant uses MCNP to model the HI-STAR 80 package to calculate external dose rates. Since this is a 3-dimensional code capable of modeling complex geometries, the HI-STAR 80 package can be modeled explicitly without gross approximations.

The staff examined some of the input files of representative shielding calculations. The staff verified that the dimensions were consistent with the package drawings presented in Section 1.3 of the SAR. The applicant used the minimum dimensions for all of the package components expected to have a significant effect on dose rates. The applicant has a list of components with the dimensions used in Table 5.3.9 of the application. The staff reviewed this table and found the list of components acceptable as the applicant included all components that provide shielding and consequently effect dose rates. This is based on the staff's review of the drawings in Section 1.3 of the application.

The staff also reviewed the value of the component thicknesses and determined that it is consistent with the minimum dimensions specified in the drawings within Section 1.3 of the application and found this acceptable. For the neutron shielding material, the minimum cold dimensions of the radial neutron shielding blocks are specified within the drawings in Section 1.3 of the application; however, within the MCNP calculations the applicant used dimensions representing the thermal expansion of these blocks and justifies the assumed gaps in the Holtec Report HI-2177580. The staff reviewed this information and verified that the gaps assumed within the MCNP model are appropriate based on the calculation of the maximum gap in Appendix B and Figure B.1.2 of Holtec Report HI-2177580.

The applicant did not consider the personnel barrier in performing the shielding evaluations. The staff found this to be appropriate as the personnel barrier is not structurally qualified.

5.3.3.1 Spent Nuclear Fuel and Hardware Modeling

The applicant homogenized the fuel assemblies rather than model them explicitly (i.e. pin by pin) and placed the homogenized fuel assemblies within the basket which was modeled explicitly. It neglected the mass of the grid spacers, which is conservative. The staff finds that modeling fuel assemblies as homogenous rather than an exact geometry is acceptable, based on Section 4.2 of NUREG/CR-6802 which states that explicit pin by pin modeling produces statistically equivalent results as homogenized fuel assemblies.

The applicant also modeled the end fittings and plenum regions as homogenous regions of steel and/or zircaloy. The applicant shows the axial lengths of the fuel assembly components in Tables 5.3.1 and 5.3.2 of the application. The staff found these acceptable based on fuel design information from DOE/RW-0184, Volume 3 of 6, Appendix 2A, "Physical Descriptions of LWR Fuel Assemblies," December 1987. The applicant shows the homogenized densities in Tables 5.3.5 and 5.3.6 of the application. The staff's evaluation of the materials used in the applicant's model is discussed in Section 5.3.4 of this SER.

5.3.3.2 Fuel/Source Modeling for HBF

The applicant discusses its approach for addressing considerations related to high burnup (HBF) in Section 1.4 and Appendix 1.A of the application. HBF is defined as fuel experiencing burnup above 45 GWd/MTU. The applicant's structural analyses of the fuel show that fuel rod breakage under NCT is not credible and that fuel remains undamaged during HAC. However, the material data is insufficient to support such a definitive conclusion for the high burnup fuel under HAC.

The staff is still investigating the effects of NCT and HAC on HBF and therefore is unable to confirm if the applicant's conclusions are consistent with expected effect. To address this uncertainty in the mechanical performance of the fuel assembly of high burnup fuel, the applicant performed calculations simulating reconfiguration consistent with the recommendations in NRC Draft Regulatory Issue Summary (RIS) 2015-XX, "Considerations in Licensing High Burnup Spent Fuel in Dry Storage and Transportation," (ADAMS Accession No. ML14175A203) to demonstrate that the package can still meet the regulatory dose rate requirements under NCT and HAC. Although this RIS is preliminary, it represents the staff's understanding and recommendations for addressing HBF reconfiguration at this time and therefore the staff found its use acceptable.

The applicant's analysis is documented in Section 5.4.5 of the application. The details on how the applicant interprets the NCT and HAC reconfiguration scenarios are discussed in the following subsections.

(a) Fuel/Source Modeling for HBF under NCT

The NRC's Draft RIS on high burnup fuel states that a 3% failure under NCT is considered acceptable. The NRC's Draft RIS is not prescriptive on how this 3% is to be applied. The applicant applied a [[10% axial collapse scenario by reducing the fuel height by 10% and increasing the source strength by the same amount equally]]. The staff does not necessarily find this to be an accurate representation of reconfigured fuel. If fuel were to fail, it would not recollect in a uniform way but most likely collect in localized spaces, probably at the bottom of the package.

Although the staff finds that a more realistic analysis would better represent fuel reconfiguration, based on the applicant's analysis of the fuel as documented in Section 2.11 of the application and based on the staff's studies on HBF reconfiguration to date, the staff has reasonable assurance that HBF will not reconfigure under NCT, and that this analysis, performed as defense-in-depth, is representative enough to show that minor fuel reconfiguration would not cause external dose rates to increase beyond regulatory limits. The staff does not have as much data on HBU MOX fuel and finds that there is likely more uncertainty in the performance of HBU MOX fuel; however, considering that there are only 4 MOX assemblies allowed during a single shipment and they are to be placed in the intermediate region of the F-32B basket, it is the staff's judgment that a more realistic fuel reconfiguration analysis would not have any significant effects on the external dose rate.

The results of the reconfiguration analysis are shown in Tables 5.5.4 and 5.4.5 of the application. These tables show that the dose rate increases for the surface of the package is about 10-15% near the lower forgings and that dose rates evaluated at all other locations are essentially the same at the surface and at 2 meters from the package. Although the applicant did not include this increase in the evaluations used to determine loading limits, the staff found that there is enough margin to the limit to account for this effect. Further, as stated in the previous paragraph, the staff does not expect the fuel to reconfigure under NCT and as a "defense in depth" analysis, the staff finds that there is reasonable assurance that the dose rates will remain under the regulatory dose rate limits under NCT.

(b) Fuel/Source Modeling for HBF under HAC

To account for possible degradation and reconfiguration of HBF during HAC, the applicant analyzed collapse scenarios consistent with the recommendations in the staff's draft RIS on high burnup fuel which states that 100% fuel failure is a conservative assumption for shielding analyses and therefore the staff finds this acceptable.

The applicant analyzed 3 collapse scenarios. The applicant performed these evaluations modeling both the F-32B (BWR) and the F-12P (PWR).

The results in Table 5.4.2 for the F-32B and 5.4.3 for the F-12P of the application show that the maximum increase is around the bottom location of the package. The increase on the bottom for both PWR and BWR models is expected as compressing the source will bring it closer to the bottom. The limiting dose rate location of the HI-STAR 80 under HAC is on the side of the package. Even with the consideration of the potential reconfiguration of fuel, the dose rates on the bottom of the package still do not exceed those of the side of the package.

For this reason, the staff found it acceptable that the applicant did not include reconfiguration in its design basis calculations that determine loading limits. It is the staff's judgment that the applicant applied the recommendations for fuel failure from the staff's draft RIS in a reasonably bounding manner and staff has found that the applicant adequately addressed the dose rate increase from possible fuel reconfiguration under HAC in its dose rate evaluations.

(c) Non-Fuel Waste Source Modeling

The applicant modeled the steel core basket from the NFWB-1, which provides additional gamma shielding, under NCT. The basket is not modeled under HAC. The applicant ignored the support plates, guide bars, and lifting lugs. This is conservative. The applicant did not model the optional secondary packaging, CCC-1. This is appropriate as this packaging is optional and may not be present.

The allowable contents for the NFWB basket are varied in material and geometry and are going to be different for each shipment. Therefore the applicant needs to model the source contents in a bounding way for all allowable materials and geometries. Section 5.2.4 of this SER discusses the gamma source modeling for this content and Section 5.3.4.3 of this SER discusses the materials modeled for this content. This section of the SER discusses the applicant's modeling of the source geometry. The geometry of the source is important, as a concentrated source of a given activity would contribute more to dose rate at a specific point than a source of equal activity distributed over a larger volume.

As discussed in Section 5.4.8.4 of the application, the applicant models two types of sources, and these are reflected as having different limits within Table 7.D.9 of the application. The "mass specific source" limits are to be applied to sources that are activated throughout, whereas the "surface specific source" is to be applied to sources that have surface contamination.

For the "mass specific source" model, the applicant filled the cavity with material at its full density. This exceeds the allowable amount of mass or activity for the package however this may not be necessarily conservative as self-shielding increases with source. It is appropriate given the limits are specified as activity per mass and therefore represents the maximum amount of source allowed. This content has limiting dose rates at the surface which are more affected by localized changes in source and therefore filling the entire package with the maximum activity/mass would allow the applicant to evaluate the location of highest dose rates. The staff found this modeling acceptable.

For the “surface specific source” model, the applicant concentrated it on the surface toward the outside of the cavity. This brings the source closest to the outside of the package and is a conservative modeling assumption.

The staff finds that homogenous/uniform source distribution can be non-conservative when modeling a component that has non-uniform activation. For example a control rod would have more activation at the bottom where it was closest to the reactor core during power operations. Modeling this component as uniformly distributed would be non-conservative; therefore, the applicant has added a Note to Table 7.D.9 of the application that states that the user, when evaluating the most activated or contaminated section of the component, must limit the mass or surface area to that of the most activated or contaminated section. The staff found that it had reasonable assurance that the applicant’s analysis would bound components loaded with this restriction.

5.3.4 Material Properties

The applicant lists the material properties it used within the dose rate evaluations in Tables 5.3.3 of the application. All of these materials are represented in the MCNP dose rate evaluation model. The following subsections of this SER provide the staff’s evaluations of the material properties used in the shielding models.

5.3.4.1 Packaging Materials

With the exception of Holtite-B, discussed below, the staff verified the density and composition of the cask components in Table 5.3.3 of the application to be consistent with typical material properties as used in shielding and criticality code material libraries and therefore acceptable.

The derivation of the Holtite-B density used in Table 5.3.3 of the application is discussed in Holtec Report HI-2177580. It states that the density in Table 5.3.3 of the application considers the worst case weight loss due to aging and the highest thermal expansion. There are 2 Holtite-B densities specified in Table 5.3.3 of the application. One for the radial neutron shielding and one for all other regions.

The radial region density is based on the expansion of the minimum allowable dimensions for the Holtite-B blocks within the package drawings. For all other regions the density is calculated considering volumetric fill ratios of the Holtite-B block volume reduced due to cold conditions and manufacturing tolerances to that of the minimum pocket volume]]. The staff reviewed this evaluation and confirmed the density of cold Holtite-B in HI-2177580 is equal to that of the required density within Table 8.1.10 of the application. The staff also found the applicant’s adjustment to the Holtite-B density under operating (hot) conditions acceptable.

5.3.4.2 SNF Materials

The applicant modeled the fuel using nuclides representative of spent fuel. The applicant used the nuclides from the depletion calculations it performed to determine the composition of each node using the TRITON and ORIGAMI/ORIGEN codes, as discussed in Section 5.2 of the application. This approach is [[more accurate, and more conservative. This is because the actinides and fission products within spent fuel nuclides generate more (n,2n), ($_{\gamma}$,n) reactions and also generate more secondary gammas than fresh fuel (B. L. Broadhead, M. D. DeHart, J. C. Ryman, J. S. Tang, and C. V. Parks, Investigation of Nuclide Importance to Functional Requirements Related to Transport and Long-Term Storage of LWR Spent Fuel, ORNL/TM-12742, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, June 1995). The applicant shows the compositions and densities for the spent fuel for a representative burnup, enrichment and cooling time combination in Table 5.3.8 of the application. The staff finds that using spent fuel composition to represent the fuel is acceptable.

Most of the BWR assemblies have part-length rods that would have a lower fuel density in the upper part(s) of the assembly where the part-length rods are not present. The applicant provides a justification for the modeling of part-length assemblies as full length in Section 5.E.1.2 of the application in the discussion on axial burnup profiles. The applicant modeled the assembly 10x10S2 using its part-length rods, and compared it to the design basis assembly with the standard axial burnup profile for several different burnup/enrichment/cooling time combinations. The applicant shows in Table 5.E.3 of the application that the two models produce essentially the same dose rates and, therefore, the staff found that modeling part-length BWR assemblies using the material properties of the full length model is acceptable for the HI-STAR 80 package.

5.3.4.3 NFWB-1 Non-Fuel Waste Materials

The material and density of the contents allowed within the NFWB-1 are challenging to model as the allowable contents will vary from shipment to shipment. The applicant's assumptions need to be applicable and conservative with respect to all allowable contents. As discussed in Section 5.4.8.5 of the application, the applicant expects that the predominant material of these components will be stainless steel however other materials with lower density, and lower self-shielding capability, may also be a part of the content.

The "mass specific source strength" limits specified in Table 7.D.9 of the application are based on activity per mass, therefore the density of the self-shielding material is accounted for within the loading limits. The applicant performed an analysis for various materials to show the difference between the attenuation properties for different materials of a given density. For the "mass specific" content, in addition to stainless steel the applicant also modeled Inconel, zirconium, and aluminum. In determining the activity limits for each energy group the applicant used the material that produced the lowest limit. The staff found this conservative.

The applicant included a note on Table 7.D.9 of the application that states that only the mass of stainless steel, Inconel, zirconium and aluminum can be used when evaluating the "mass specific source strength." Neglecting the mass of other materials is conservative as these materials will in reality provide some self-shielding. The staff found that the applicant's analysis accounts for all possible self-shielding materials allowed within the NFWB-1 for the "mass specific" source evaluations.

In determining the "mass specific source strength," Table 7.D.9 of the application also has a note that the user must neglect the mass of any non-activated handling or support material. The staff found this to be conservative as this material may provide some shielding; however, it is appropriate to neglect it within the dose rate evaluations as there are no placement or loading requirements for any handling or support material and its presence during shipment cannot be predicted.

For the surface source (contamination) and HAC source, the applicant only performed calculations with stainless steel and did not model the other materials. For the surface source, the inner material does not provide as much self-shielding and therefore would not affect the external dose rate as much and, therefore, the staff found using stainless steel for the surface source acceptable.

The applicant chose to use stainless steel for HAC evaluations because the content limits for the NFWB-1 are limited by the NCT evaluations, and there is more margin to the regulatory limit under HAC. The staff reviewed the applicant's evaluations using the different materials in Table 5.4.33 of the application and found that the difference in dose rates using different materials for any of the energy groups does not exceed that of the margin to HAC dose rate limits from Table 5.1.11 of the application and found the use of stainless steel acceptable for HAC evaluations.

5.4 Evaluation

5.4.1 Codes

The applicant calculated the gamma and neutron source terms from radioactive fission products using the TRITON and ORIGAMI/ORIGEN modules of the SCALE 6.2.1 system using the ENDF/B-VII 252-group library provided by the developer of the code. This is also discussed in Section 5.2.1 of this SER.

The applicant uses the MCNP-5 1.51 code to calculate dose rates. MCNP is a transport theory based three dimensional code that employs the Monte Carlo solution method. This code has been widely used across a wide range of applications and is well benchmarked and tested. The applicant used cross section libraries from ENDF/B-VI and ENDF/B-VII data. The applicant states that these are the default libraries for the MCNP code.

The applicant did not perform benchmarking studies with this code specific to the HI-STAR 80, however the applicant referenced some reports of general benchmarking performed on this code by Los Alamos National Laboratory. The staff finds that this provides reasonable assurance that the code is capable of predicting the dose rates.

5.4.2 Calculation Process

The applicant evaluated the dose rates using what it calls a "two-step" process. This is described in Section 5.4.1 of the application. The staff reviewed the applicant's evaluation method described by Equation 5.4.1 in the application and found it acceptable for this application. The staff found that this method is capable of considering the neutron, gamma and Co-60 contributions to dose rates and allows the applicant flexibility in being able to evaluate dose rates from multiple loading patterns and assemblies.

Since MCNP is a transport theory based Monte Carlo method code, the calculated values have an uncertainty associated with them. This uncertainty needs to be considered in the dose rate evaluation. The applicant states that it added this uncertainty to all design basis results. Given the two-step process, discussed above, including this uncertainty is not straight forward. The applicant shows how it statistically combined the values related to determining this uncertainty in Section 5.4.1 of the application and it is represented by Equation 5.4.5 in the application. The staff reviewed this information and found it acceptable.

5.4.3 Flux-to-Dose-Rate Conversion

The applicant states that it uses the ANSI/ANS 6.1.1-1977 flux-to-dose rate conversion factors in all the shielding evaluations. The staff finds this acceptable per Section 5.5.4.3 of NUREG-1617.

5.4.4 Tallies

To demonstrate that the package design meets the external dose rates at locations prescribed for the NCT vehicle surface, 2 meters, and HAC, the applicant is required to determine the maximum dose rate considering all points on the surface. The presence of neutron absorber and the copper ribs in the cask wall could result in azimuthal variations in dose rate. The axial burnup profile causes axial variations in dose rate. The applicant must use sufficiently small tally bins such that a maximum is not reduced to an average considering these features. The applicant's discussed its tally specifications in Section 5.3.4 of the application.

The staff reviewed the applicant's tally specifications and found that it considered dose rates around every location of the package. The surface tallies were positioned in line with the ribs and Holtite from a previous design of the package that contained aluminum ribs and are not aligned with the current locations of the ribs and Holtite. The current design has angled ribs, which improves shielding and reduces streaming paths as every location will have some neutron shielding material and some additional gamma shielding from the ribs. The applicant did an evaluation, as discussed in Section 5.3.1.3 of the application where it compared the dose rate from misaligned tallies with that of tallies aligned with the rib's location on the outside of the package and found that there was essentially no noticeable streaming effects. The staff therefore found the locations of the surface tallies acceptable for this application.

For the tallies, for the radial 1 and 2 meter locations, the staff considered the height of the axial burnup profile cells and found this sufficiently small to be able to discern the maximum dose rates from average dose rates.

For the axial tallies the applicant divided the tally volumes into circular disks. The staff also found that the radius was sufficiently small to be able to discern the maximum dose rates from average dose rates considering the width of fuel assemblies that would occupy each ring.

5.5 Evaluation Findings

The staff reviewed the package shielding design, calculated dose rates, material specifications, and models for dose rate calculations. The staff found the applicant used the dimensions and material compositions consistent with the package drawings and bill of materials. The applicant's dose rate calculations, including source term and shielding model assumptions are conservative. The calculated dose rates meet the dose rate limits prescribed in 10 CFR 71.47 for a package under NCT and 10 CFR 71.51 for a package under HAC respectively.

The staff followed the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," March 2000, and Supplement 1 to NUREG-1617, "Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel," in its review. Based on its review of the information and representations provided in the application, the staff has reasonable assurance that the proposed package design and contents satisfy the shielding requirements and external dose rate limits in 10 CFR Part 71.

6.0 CRITICALITY REVIEW

6.1 Review Objective

The HI-STAR 80 is designed to transport intact high burnup PWR or BWR UO_2 fuel, BWR MOX fuel, mixed BWR UO_2 and BWR MOX fuel assemblies, or non-fuel radioactive waste and hardware. PWR fuel and associated non-fuel hardware will be transported in the F-12P basket, and the BWR fuel and BWR MOX fuel will be transported in the F-32B basket. The non-fuel waste and some non-fuel hardware will be transported in the non-fuel waste basket (NFWB-1) and the total weight of the fissile materials in the NFWB-1 shall not exceed the exempted quantity of fissile materials as defined in 10 CFR 71.15. Damaged fuel is not an authorized content for transportation with the HI-STAR 80 packaging system.

The objective of this review is to determine that the Model No. HI-STAR 80 package, loaded with the spent fuel assemblies as specified in the Certificate of Compliance, meets the regulatory requirements of 10 CFR 71.55, 71.59, and 71.87, i.e., the package remains subcritical under NCT, HAC, as well as during loading and unloading operations.

6.2 Criticality Safety Evaluation

The NRC staff reviewed the design of the HI-STAR 80 package to determine whether this package meets the regulatory requirements, with respect to criticality safety, as prescribed in 10 CFR 71.55, 71.59, and 71.87. The staff's review results are documented below. The staff did not perform a criticality safety evaluation for the package containing NFWB-1 basket because the fissile material in the NFWB-1 basket is limited to the exempted quantity per 10 CFR 71.15.

In its review, the staff followed the guidance and the acceptance criteria provided in NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel” and Supplement 1 to NUREG-1617, “Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel”

6.2.1.1 Packaging Design Features

The HI-STAR 80 packaging system consists of an overpack and a fuel basket, or non-fuel waste and hardware basket, is inserted inside the cavity of the overpack. The overpack employs two independent bolted closure lids. The structure design of the overpack ensures that each bolted lid joint is engineered to meet the leaktight standard of ANSI N14.5 under NCT and HAC, as prescribed in 10 CFR 71.71 and 71.73 respectively. Both closure lids are designated as containment boundary components. Based on the results of the structural review, each containment boundary closure meets the water exclusion criterion with significant safety margin because moderator exclusion requires only no ingress of water into the cavity of the package, which is a much less demanding requirement than a leaktight design. Based on this fact, the staff determined that the package design meets the criterion for moderator exclusion in accordance with the acceptance criteria specified in Interim Staff Guidance number 19 (ISG-19).

The Model No. HI-STAR 80 package includes two fuel basket designs: (1) a 12-cell basket (F-12P) and (2) a 32-cell basket (F-32B). The F-12P basket is designed for transporting undamaged spent PWR UO_2 fuel assemblies with a maximum enrichment of 5 wt% U-235. The F-32B is designed for transporting undamaged spent BWR UO_2 fuel assemblies with a maximum planar average enrichment of 5 wt%, or undamaged spent BWR MOX fuel assemblies with specific composition as defined in Table 7.D.1 of the application, or a mixed load of spent BWR UO_2 and BWR MOX fuel assemblies in the same basket. Hybrid MOX fuel assemblies, i.e., BWR fuel assemblies that contain both MOX fuel rods and UO_2 fuel rods, are not authorized contents.

The applicant states in the SAR: "The F-12P basket is designed to be loaded with *fresh* or spent PWR UO_2 fuel assemblies, and the F-32B basket is designed to be loaded with *fresh* or spent BWR UO_2 or BWR MOX fuel assemblies." The staff verified that this is an incorrect statement: the package is designed for transporting only spent PWR UO_2 fuel or BWR UO_2 and MOX fuel.

The fuel basket is a stainless steel cylinder shell and its internal cavity is compartmentalized into square fuel cells by Metamic-HT plates that contain B-10 as neutron poison. The Metamic poison plate is 0.79 inches thick with a minimum areal density of 0.0441 g/cm^2 of B-10 for the F-12P PWR fuel basket. For the F-32B BWR fuel basket, the Metamic poison plate is 0.59 inches thick with a minimum areal density of 0.0441 g/cm^2 of B-10. The B-10 loading requirements are shown in Table 7.1 of Holtec Report HI-2084122.

Fuel assemblies are loaded and held in the fuel cells to maintain their geometric locations under NCT and HAC. Metamic-HT, which is an aluminum and B_4C composite material, is also used as the structural material for the spent fuel basket. Therefore, all assemblies in the basket are completely surrounded by neutron absorbing material to assure criticality safety of the package under NCT and HAC.

In addition, the F-12P basket design also includes flux traps. This special design feature reduces the neutronic coupling between the adjacent fuel assemblies in the fuel basket to reduce the potential nuclear chain reactions. This is another criticality safety feature that reduces the reactivity, k_{eff}, of the package.

The F-12P basket allows for loading of spent PWR fuel assemblies in all 12 cells of the basket, or 10 cells with locations 4 and 9 empty, or 10 cells with locations 5 and 8 empty. The F-32B basket allows for loading of 32 undamaged spent BWR UO_2 fuel assemblies or undamaged spent BWR MOX fuel assemblies or a combination of four (4) MOX and 28 UO_2 BWR fuel assemblies. Undamaged spent BWR fuel assemblies can be loaded in all 32 cells of the basket, or 28 cells with locations 13, 14, 19, and 20 empty, or 24 cells with locations 12, 13, 14, 15, 18, 19, 20 and 21 empty, or 24 cells with locations 7, 8, 13, 14, 19, 20, 25 and 26 empty.

The staff notes that the reduced fuel loading configurations are intended to meet the limit decay heat of the content in the package. These partially loaded baskets are bounded by the fully loaded basket in terms of criticality safety because they are much less reactive in comparison with fully loaded packages for two reasons: (1) there is less fuel, and (2) the empty cells reduce neutron communication between the fuel assemblies in the baskets, functioning similar to flux traps. For this reason, the staff's criticality safety evaluations are focused on packages with a fully loaded basket.

The F-12P basket allows for loading of 2 fuel types (17x17 and 15x15) with a total of 7 subclass fuel designs. The F-32B basket allows for loading of 3 fuel types (8x8, 9x9, and 10x10) with a total of 13 subclass fuel designs. The maximum U-235 enrichment of PWR fuel is 5 wt%. The maximum planar average U-235 enrichment of the UO_2 BWR fuel is also 5 wt%, with a maximum rod U-235 enrichment of 6 wt%. The material compositions of the MOX fuel, including the plutonium vector, are defined in Table 7.D.1 of the application. The characteristics of the allowable fuel are provided in Table 7.D.2 and Table 7.D.3 of the application for UO_2 PWR and BWR fuel respectively.

When the package is loaded with mixed UO_2 and MOX BWR fuel assemblies, only four (4) MOX fuel assemblies are authorized to be loaded per package and these MOX fuel assemblies must be loaded in cells 6, 9, 24, and 27 in the F-32B basket as shown in Figure 7.D.2 in Chapter 7 of the application. No other loading patterns are allowed.

The applicant includes the licensing drawings in the application that show (i) the structural layout and dimensions, manufacturing tolerance, and bill of materials of the HI-STAR 80 overpack that are important to evaluating the criticality safety of the package, (ii) the structural layout and dimensions, manufacturing tolerances, and bill of materials of the of the F-12P and of the F-32B fuel baskets.

In summary, the HI-STAR 80 uses neutron poison plates or a combination of neutron poison plates and flux trap (F-12P basket only) to ensure criticality safety of the package under NCT. It also employs double closure lids to meet the criterion of moderator exclusion in accordance with the guidance provided in ISG-19.

6.2.1.2 Summary Table of Criticality Evaluations

The applicant performed criticality safety analyses for packages containing 12 PWR fuel assemblies in an F-12P basket, package containing 32 BWR fuel assemblies in an F-32B, and package containing 28 UO_2 BWR fuel assemblies and 4 MOX fuel assemblies in the F-32B basket respectively. The applicant evaluated a single package with internal water flooding and reflected with 30 cm water to demonstrate that the package is in compliance with the regulatory requirements of 71.55(b) and 71.55(d). For the package under HAC, the applicant took credit for the double closure design feature of the overpack and evaluated the criticality safety without moderator because the package meets moderator exclusion design criterion. The applicant presents its summary of criticality safety analysis results for the package in Table 6.1.1 of the application.

This package is designed to transport high burnup fuel (burnup exceeding 45 GWd/MTU up to 70 GWd/MTU). For high burnup fuel, the structural analyses demonstrated the fuel will retain its geometry under the tests of 10 CFR 71.71. However, there is no sufficient data to assure that the fuel cladding will be able to withstand the impacts without deformation under the tests prescribed in 10 CFR 71.73. For this reason, the applicant evaluated criticality safety with the assumption that the fuel geometry would change under HAC. The applicant performed criticality safety analyses for the HI-STAR 80 package under HAC with the assumption that fuel has lost its geometric shape and assumes the package cavity will remain dry because the double closure design will prevent water from leaking into the cavity of the cask. The applicant demonstrated that any potential reconfiguration of high burnup fuel under HAC would be inconsequential from the criticality perspective because of lack of moderator inside the package. This is consistent with the acceptance criteria as specified in ISG-19 and the design is acceptable to the staff as a moderator exclusion package.

The applicant assumed that the fuels are unirradiated, i.e., the applicant took no credit for the fuel burnup. This assumption provides a significant additional safety margin with respect to criticality safety because the net loss of fissile materials and accumulation of fission products in the light water reactor fuels both lead to a reduction of the reactivity of the fuel.

The applicant provides a summary of the calculated k_{eff} values in Table 6.1.1 of the application. The data presented in Table 6.1.1 of the application include the k_{eff} values for a single package under NCT and HAC as well as an array of packages under NCT and HAC. The maximum k_{eff} value for the package containing the fully loaded F-12P basket with the maximum U-235 enrichment of 5 wt % is 0.9485. The maximum k_{eff} value for the package containing the fully loaded F-32B is 0.9438. The calculated k_{eff} values show that the package under NCT remains subcritical with all uncertainties in the calculation and include an adequate administrative safety margin. This result demonstrates that the package meets the requirements of 10 CFR 71.55(b) and 71.55(d).

The k_{eff} values in Table 6.1.1 also demonstrate that the package under HAC remains subcritical even with considerations of hypothetical fuel reconfigurations. The k_{eff} values for the packages under HAC are much lower than for the same package under NCT because the HAC criticality safety model assumes there is no moderator inside the package. The applicant calculated the k_{eff} value for a package containing each subclass of fuel and provided the results in Tables 6.1.2 and 6.1.3 of the application for PWR fuel and BWR fuel respectively. This data demonstrates that the reactivity provided in Table 6.1.1 of the application is the maximum, based on the comparison of the k_{eff} values for different fuel types.

As discussed earlier in this section of the SER, based on the evaluation results of the package's structural performance under the tests prescribed in 10 CFR 71.73, the staff finds that the applicant's criticality safety analysis for the package under HAC is consistent with the damaged condition and the assumptions used in the modeling of the package are conservative and meet the acceptance criterion provided in NUREG-1617.

The applicant's criticality safety evaluations show that an infinite array of undamaged or damaged packages remains subcritical. The applicant calculated the criticality safety index (CSI) of this package following the procedures as prescribed in 10 CFR 71.59. The CSI for this package is determined to be 0.

The staff reviewed the applicant calculation of the CSI and modeling of arrays of packages under NCT and HAC. The staff finds that the applicant's criticality safety analyses for arrays of packages under NCT and HAC are consistent with the package performance under their respective condition, and the applicant's calculation of the CSI value followed the method prescribed in 10 CFR 71.59 and is therefore acceptable.

6.2.2 Spent Nuclear Fuel Contents

The HI-STAR 80 radioactive material transportation package is designed to transport intact high burnup spent PWR and BWR UO_2, or BWR MOX fuel, or a mixed load of UO_2 and MOX fuel assemblies in the same fuel basket. The spent PWR UO_2 fuel assemblies must be loaded in the F-12P basket. Mixed loading of spent BWR UO_2 and BWR MOX fuel assemblies must be loaded in the F-32B basket. Only four MOX fuel assemblies are authorized to be loaded in the F-32B basket and the MOX fuel assemblies must be loaded in cells 6, 9, 24, and 27 as identified in Figure 7.D.2, "Cell Numbers for F-32B Basket" of the application. Damaged fuel is not authorized for transport with the HI-STAR 80 transportation system.

6.2.3 General Considerations for Criticality Evaluations

6.2.3.1 Model Configuration

The applicant explicitly modeled the fuel assembly, the Metamic-HT fuel basket fuel cell structure, which is a neutron poison material, and other structural and overpack components of the package that are important to criticality safety. Impact limiters are not included in the model. The applicant used fresh fuel composition for the spent fuel assemblies and therefore takes no credit for the loss of fissile materials and accumulation of fission products and non-fissile transuranic materials that are physically present in the spent fuel assemblies, i.e., no burnup credit is taken in the criticality safety analyses for the package. The staff finds that this is a significant conservative assumption in the criticality safety analysis.

In the criticality safety model, the applicant took credit for 90% of the boron in the Metamic-HT neutron poison plates. The staff finds that this is consistent with the guidance provided in Interim Staff Guidance-23, “Application of ASTM Standard Practice C1671-07 when performing technical reviews of spent fuel storage and transportation packaging licensing actions,” and therefore acceptable.

The applicant also included package and fuel basket manufacturing tolerances in its criticality safety analyses. These tolerances are consistent with the package’s allowable tolerances as shown in the drawings of the package. The applicant performed sensitivity studies on parameters that affect reactivity, such as fuel density and water temperature in the cask, using the CASMO-5 code.

The results, presented in Table 6.3.4 of the application, show that using the maximum fuel density and the minimum water temperature (corresponding to the maximum water density) provides bounding conditions for criticality safety analyses. Therefore, the applicant used these conditions in all of its criticality safety analysis models.

The applicant stated that the fuel temperature sensitivity analyses are not used to demonstrate compliance with regulations but only to determine the sensitivity of the system to changes in temperatures of the moderator, fuel, and structure materials and determine the bounding conditions. The staff finds that this sensitivity study, though not directly used in the criticality safety analysis of the package, does show that the applicant has considered these additional factors and their potential impact on criticality safety.

Through these analyses, the applicant identified the bounding parameters for the criticality safety analyses in the anticipated ranges of those parameters.

6.2.3.2 Material Properties

The applicant provides material compositions of the various components of the HI-STAR 80 package in Table 6.3.5 of the application. The data in the table includes the nuclide identification number (ZAID) for each nuclide, the atomic number, mass number, and the cross-section evaluation identifier, which are consistent with the ZAIDs in the MCNP manual.

The HI-STAR 80 package uses Metamic-HT fixed neutron poison plate, which is aluminum alloy contain B_4C as neutron absorber. The applicant states that the HI-STAR 80 is designed to ensure the fixed neutron absorber will remain effective for a period greater than 40 years and that there are no credible means to cause significant loss of B-10 in the poison plates during this design basis package life-time. The continued efficacy of the fixed neutron absorber is assured by acceptance testing, documented in Section 8.1.5.5 of the application, to validate the ^{10}B concentration in the fixed neutron absorber. In addition, based on its own calculations for a similar cask model, the staff finds that loss of ^{10}B atoms, due to neutron irradiation from the content in the fixed neutron absorber by neutron absorption during the service life, is negligible (less than 10^{-8} percent of the original loading). Therefore, it is not necessary to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber. The applicant provides a detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of the fixed neutron absorber to demonstrate that the minimum requirement B-10 concentration is assured. This determination is confirmed by the staff’s review, as documented in the material review chapter of this SER.

The material compositions and properties of the other packaging materials are consistent with the specifications commonly used in criticality safety analyses.

The only packaging materials that may be affected under hypothetical accident conditions are the Holtite neutron shield on the outside of the cask, and the impact limiters. However, these components are not included in the criticality model. The staff finds this modeling simplification acceptable because ignoring this material neglects the absorption of boron in the neutron shield layer and provides a more conservative reflector. The combinations of these two factors produces more conservative results in criticality safety analyses. Therefore, the staff did not review the material properties of these components because they are not included in criticality safety evaluations.

The staff reviewed the material properties and the assumptions used in the criticality safety analysis. The staff finds that the material properties the applicant used are consistent with the commonly available material data and the material properties, such as density, temperature, and amount of boron-10 used in the criticality safety analysis models are conservative. On this basis, the staff determined that the material properties of the packaging materials and the contents are appropriate and acceptable.

6.2.3.3 Computer Codes and Cross Section Libraries

The applicant used the three-dimensional Monte Carlo code MCNP5, Version 5.1 and ENDF/B-VII cross section library in the package criticality safety analyses. In the MCNP criticality safety analysis models, the applicant used 10,000 simulated histories per cycle, a minimum of 400 cycles were skipped before averaging, a minimum of 400 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies).

The applicant explicitly examined the Shannon entropy index, which is part of the MCNP5 model output, to ensure convergence of the calculations by confirming that both the k_{eff} value and fission source distribution have properly converged at the end of the calculations. The staff reviewed the Shannon Entropy index provided by the applicant in the sample output file and finds that calculations have adequately converged to provide accuracy and reliability of the resultant k_{eff} values.

The staff reviewed the applicant's criticality safety evaluation method, including the computer code and cross section library as well as the assumptions used in the models. The staff finds that the MCNP code version is one of the codes recommended by NUREG-1617 for criticality safety evaluation and the cross section library represents the most up-to-date measurement data. The examination of the Shannon Entropy assured the adequate conversion of the calculations and therefore the accuracy and reliability of the resultant k_{eff} values. On this basis, the staff finds that the computer code and cross section library are adequate for this application.

6.2.3.4 Demonstration of Maximum Reactivity

The applicant calculated the neutron multiplication factors, k_{eff}, for the HI-STAR 80 package with each of allowable fuel types as specified in Tables 7.D.2 and 7.D.3 of the application and Chapter 1 of the application. The applicant searched the maximum reactivity with considerations of moderator density and rod pitch changes, which are the two most important parameters that may cause the system k_{eff} to change for the package under NCT for a fixed enrichment.

Regarding the effect of low moderator density, the applicant determined that, with a neutron absorber present (i.e., the neutron poison integral to the walls of the storage compartments), the phenomenon of a peak in reactivity at a hypothetical low moderator density (also called "optimum" moderation) does not occur to any significant extent based on a definitive study performed by Cano, et al. "Supercriticality through Optimum Moderation in Nuclear Fuel Storage," Nucl. Technol., 48, 251-260, (1980). The staff reviewed the cited publication and finds that the study is applicable to the HI-STAR 80 package design because the system is loaded with heavy neutron absorbing materials, namely B-10, in the Metamic plate.

Based on this publication, the applicant did not evaluate the reactivity of a partially flooded package. Instead, it studied the effect of moderator level on the reactivity of the package. Based on the applicant's analyses, as shown in Table 6.3.12 of the application, in all cases, for both the F-12P and F-32B baskets, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded. Based on the study referenced by the applicant, the staff finds the applicant's justification for not performing criticality analyses for partially flooded package to be acceptable because the analyses for the fully flooded package bounds the "partial flooding" condition.

The applicant demonstrated that UO_2 BWR fuel assembly design with 5 wt% U-235 enrichment bounds the MOX fuel assembly design with total plutonium mass not to exceed 14 kg per assembly and the Pu composition vector not to exceed the limit as specified in Table 6.2.6 of the application. On this basis, the staff finds that the applicant's determination that the package containing MOX fuel as specified is bounded by the UO_2 fuel is acceptable.

By its analyses, the applicant demonstrated that the package containing 12 PWR UO_2 fuel assemblies in the F-12P basket is the most reactive configuration of the HI-STAR 80 package. Table 6.1.1 of the application shows the reactivity of the package at the most reactive condition under NCT. Because the package is demonstrated to be leak-tight under HAC, there is no moderator in the package under HAC. Based on these analyses, the applicant demonstrated that the package under HAC is much less reactive than a fully flooded package even with consideration of highly unlikely fuel reconfiguration.

The staff finds the applicant's analysis for the package under HAC and the results are consistent with the well understood nuclear physics (much lower reactivity without moderator for low enrichment system) and to be acceptable. On this basis, the staff determined that the applicant has identified the most reactive configuration of the package and the package meets the regulatory requirements of 10 CFR 71.55(b).

6.2.3.5 Confirmatory Analyses

The staff performed confirmatory analysis for the most reactive configuration, i.e., water flooded and reflected single package containing the F-12P basket loaded with bounding 15x15S3 PWR assemblies. The staff used the SCALE 6.1 computer code with continuous energy cross sections derived from the ENDF/B-VII cross section library. The results confirm the applicant's calculated k_{eff} value for the bounding package design.

6.2.4 Single Package Evaluation

The applicant performed criticality safety analyses for packages containing the F-12P or the F-32B basket under NCT. The applicant modeled the fuel assembly assuming flooded fuel cladding gap and the inter-cavity of the package. The applicant explicitly modeled the fuel basket structure, which also serves as neutron poison plates, and the overpack in the models. The applicant ignored the impact limiters. Table 6.1.1 of the application provides the results of the criticality evaluation. The results of the analyses show that the package meets the criticality safety requirements of 10 CFR 71.55(b) and 71.55(d).

The applicant also evaluated the criticality safety of a single package under HAC with the assumptions that the package internal remains dry, i.e., there is no moderator intrusion. The applicant surrounded the package with 30 cm of water reflector. The applicant ignored the impact limiters in the model. Table 6.1.1 of the application provides the results of the criticality calculations. The results show that the package remains subcritical when subjects to the tests prescribed in 10 CFR 71.73.

Based on the result of structural review as documented in Chapter 2 of this SER, the staff determined that the package meets the acceptance criteria of moderator exclusion design as defined in NUREG-1617 and ISG-19. Therefore, the criticality safety analyses based on moderator exclusion is acceptable. The applicant presents in Table 6.1.1 of the application the maximum k_{eff} values for the packages under HAC. The result demonstrates that the package meets the regulatory requirements of 10 CFR 71.55(e).

The staff reviewed the applicant's calculations for the most reactive configurations for the HI-STAR 80 package containing both the F-12P and F-32B basket and finds that the applicant has appropriately analyzed the criticality safety of the package under NCT as well as HAC. On this basis, the staff determined that the applicant has demonstrated that the package meets the regulatory requirements of 10 CFR 71.55(b), 71.55(d), and 71.55(e).

6.2.5 Evaluation of Array of Packages under NCT and HAC Conditions

The applicant performed criticality safety analyses for an array of packages under NCT and HAC separately. Based on its calculations, an array of packages under NCT or HAC remains subcritical with considerations of all potential uncertainty and an administrative safety margin of Δk_{eff} = 0.5. In these analyses, the applicant assumed there is no moderator in the cavity of the package under HAC. Based on the results of the criticality safety analyses for the infinite array of packages under NCT and HAC, the applicant calculated the Criticality Safety Index of the array of packages in accordance to the method prescribed in 10 CFR 71.59 and determined that the CSI for this package is 0.0.

The staff reviewed the applicant's analyses of the criticality safety of arrays of packages under NCT and HAC and the calculation of the CSI value. The staff finds that the assumptions used in the models for an array of packages under NCT as well as an array of packages under HAC are conservative and are acceptable based on the acceptance criteria provided in NUREG-1617 and Supplement 1 to NUREG-1617.

On these bases, the staff finds that the package design with pure UO_2 fuel assemblies, MOX fuel, or mixed load of UO_2 and MOX fuel assemblies with the specific limits on contents and loading configurations as defined in the CoC and Operating Procedures.

6.2.7 Computer Code Benchmarking

The applicant performed code benchmarking analyses for the MCNP5 computer code together with the selected cross section library, ENDF/B-VII. The applicant selected a total of 562 critical experiments from the International Handbook of Evaluated Criticality Safety Benchmark

Experiments and French HTC experiment program. The selected critical experiments encompass the ranges of fuel types (PWR and BWR), material composition (UO_2 and MOX), enrichment, rod pitch, and soluble boron this application resides. Table 6.A.1 of the application lists all of the selected critical experiments.

The applicant performed trending analyses against six parameters: enrichment, fuel rod pitch, fuel assembly separation distance, soluble boron concentration, moderator to fuel volume ratio, and EALF (energy of average lethargy causing fission). Based on its code benchmarking and trending analyses, the applicant determined biases against EALF and Pu enrichment and applied them to the calculated k_{eff} values for the applicable package contents.

The staff verified the applicant's selection of critical experiments and trending analyses (including the selected trending parameters). Based on its review and verification, the staff determined that the applicant has correctly performed code benchmarking analyses and the results are conservative.

However, the applicant recognized that the available critical experiments for code benchmarking analyses for packages containing mixed UO_2 and MOX BWR fuel assemblies are very limited. To alleviate this problem, the applicant limited the number of MOX fuel assemblies that are allowed to be loaded only 4 MOX fuel assemblies at cells 6, 9, 24, and 27 of the F-32B basket as specified in figure 7.D.2 of the application. The applicant states that, with this loading pattern, the MOX fuel assemblies are effectively separated from each other so that the neutronic characteristics of the package will not be significantly influenced by the MOX fuel.

The staff reviewed the applicant's proposed solution to the lack of critical benchmark experiment problems for the mixed load of MOX and UO_2 fuel and determined that the applicant's justification is acceptable because the effective separation of the MOX fuel and the limit on the number of allowed MOX fuel assemblies will prevent the aggregation of the MOX fuel to form a dominating region. This arrangement will reduce the influence of the MOX fuel on the reactivity of the package to a minimum.

On this basis, the staff has reasonable assurance that the applicant has demonstrated that the package with mixed MOX and UO_2 fuel in the specific loading pattern meets the regulatory requirements of 10 CFR 71.55(b), 71.55(c), 71.55(d), 71.55(e), and 71.59.

6.3 Conclusions

The staff reviewed the information provided in the application and the applicant's responses to the staff's requests for additional information. Based on its review, the staff finds that the applicant made conservative assumptions in the criticality safety analyses, including maximum allowable quantity of fissile materials (assuming fresh fuel), conservative tolerance of cask geometry, reduced credit of B-10 in poison plates in the cask and the calculated maximum neutron multiplication factor, k_{eff}, with appropriate code benchmarking analyses.

Based on the review of the information presented by the applicant and its independent confirmatory analyses, the staff determined that the HI-STAR 80 package meets the regulatory requirement of 10 CFR 71.55 and the acceptance criteria specified in NUREG-1517 on criticality safety with the following conditions:

1. No damaged fuel is authorized for transportation in this package;

2. MOX fuel composition must meet the plutonium material vector as defined in Table 7.D.1 of the application;
3. For BWR MOX assemblies, the total plutonium mass does not exceed 14 kg per fuel assembly; and
4. The maximum number of MOX fuel assemblies allowed per package is 4 and these four assemblies shall be loaded into the cells 6, 9, 24, and 27 of the basket, as identified in Figure 7.D.2, “Cell Numbers for F-32B Basket” of the application.

7.0 PACKAGE OPERATIONS

The package operations descriptions contain the essential elements of operations for using the package. Where the use of alternatives to described sequences or operations is acceptable, the operations descriptions include a description of these alternate sequences and operations.

The staff reviewed the applicant’s description of package operations to ensure (i) consistency with its technical evaluation, and (ii) compliance with the shielding design specified in the technical drawings and appropriate regulatory external dose rate limits. The staff finds that, based on its review, the operations descriptions in the application are consistent with the technical design and shielding analysis.

The applicant stated in Section 7.1.2, “Loading of Contents,” that the user needs to perform a site-specific Time-to-Boil evaluation, using a methodology described in Section 3.3.4, “Time-to-Boil Limits,” to determine a time limitation to ensure that water boiling will not occur in the package prior to the start of draining operations.

As described in Section 7.1.2.1, “Fuel Loading Operations,” if it appears that the Time-to-Boil limit will be exceeded prior to draining operations, the user shall take appropriate action to either replace the water in the package cavity with an inert gas, circulate water through the package cavity to reset the Time-to-Boil clock, or return the package to the spent fuel pool and remove the lid to allow for natural water circulation.

The applicant also stated in Section 7.1.2 that users shall refer to Tables 7.1.2 and 7.1.3 for vacuum drying criteria. As the water is drained from the package, an inert gas is used if the package is loaded with the fuel and the air may be used if the package is loaded with a non-fuel waste.

The applicant provided package backfill requirements for shipment of fuel in Table 7.1.4 and shipment of non-fuel waste in Table 7.1.5, respectively. The staff reviewed Section 7.1.2 and confirmed that the loading operations of time-to-boil limit and vacuum drying criteria, as described, are appropriate.

The applicant stated in Section 7.1.3, “Preparation for Transport,” that

(a) The surface temperatures of the accessible areas of the package are measured to confirm temperatures are within 10 CFR 71.43 requirements, if the personnel barrier will not be used.

(b) The accessible surfaces of the transport package shall not exceed the exclusive use temperature limits.

(c) For packages containing high burnup fuel, surface temperatures are measured as required by the post-shipment fuel integrity acceptance test specified in Section 8.1.8.

The staff reviewed Section 7.1.3 and finds that the measurement of the package surface temperatures, as described, is acceptable.

The applicant provided, in Chapter 7 of the application, instructions for spent fuel handling (including assembly selection and verification) to be performed in accordance with written site-specific procedures, and detailed procedures to ensure that only fuel assemblies authorized in the CoC are loaded into the HI-STAR 80 package. The staff found acceptable the procedures provided by the applicant, based on its own regulatory and inspection experiences.

Spent fuel assembly selection and some aspects of assembly verification are typically performed well in advance of the actual loading date with respect to the selection and verification of the assemblies to meet the definition of undamaged spent fuel. A typical approach to show compliance with the definition of undamaged spent fuel may include the following steps:

- During reactor operation, the water chemistry is monitored. If no indications of spent fuel leakage is detected, all assemblies unloaded from the core are considered undamaged.
- If indications of leakage are found in the water during reactor operation, the population of the assemblies in the core, that may have the leak, may be narrowed down by a more detailed evaluation of the leaked isotopes, or by manipulating control blades in a BWR core.
- Once unloaded, further examination, such as sipping, may be performed to clearly identifying the leaking assembly or assemblies, out of the population identified.
- Once leaking assemblies are identified, they may simply be considered not meeting the CoC requirements and excluded from the selection, or further tests are performed to identify the extent of cladding damage.
- For channeled BWR assemblies, such further tests to identify the extent of the leak, and potentially qualify them as undamaged if the leak does not exceed the definition of undamaged fuel assemblies, would require the removal of the channel.

The operations chapter indicated that, during the loading and unloading procedures, the fuel will be either covered in water or exposed to an inert gas. During water removal and vacuum drying, however, fuel would or could be potentially exposed to air. During unloading, with the re-flooding of the cask, air would potentially again be present. The applicant indicated that in 30 years of fuel loading and unloading, there has been adequate protection against fuel oxidation during operations.

In compliance with ISG-22, the applicant assessed the potential oxidation based on (i) appropriate literature information and (ii) example calculations for time limits at various temperatures. The applicant determined that the oxidation process is highly sensitive to the temperature. Therefore, the time limit for air/moisture exposure would increase for casks with lower heat load.

The applicant also considered later stage of vacuum drying with less oxygen partial pressure, which is likely to decrease oxidation rate. The staff finds this assessment to be acceptable, based on the staff's previous research on this topic and the review of updated recent literature information.

The staff requested the applicant to include, in the operating procedures, cautionary notes or statements for the users with specific times, and/or temperature limits, to prevent fuel oxidation, and that the user shall develop or modify existing instructions and procedures to account for such fuel handling procedures.

With the clarifications and conditions imposed by NRC staff in the updated operating procedures, as included in Revision 3 of the application, the staff determined that the handling of the potential fuel oxidation is acceptable. In particular:

> If the dryness criteria is not met, inert gas will be maintained and air will not be reintroduced.
>
> For packages with fuel contents that will be vacuum dried, the cask must be filled with inert gas after the first full vacuum cycle, applicable to either standard vacuum drying or cyclic vacuum drying, even if the air exposure "time limit" permits additional air exposure time and even if the Air Exposure Threshold Time Limit (ATTL) does not apply. Air shall not be reintroduced after the first full vacuum cycle.
>
> The reserve Time Limit (RTL) is a contingency time to provide the user with enough time to cover the fuel with either water or inert gas. The RTL is independent of any ATTL.
>
> The operating procedures allow only transient evaluations for unloading, even though 400°C is used in the Appendix 3.A example.
>
> For spent fuel loading, it is now clearly stated that a cladding cutoff temperature of 380°C is built into the vacuum drying process, via the thermal analysis. The operating procedures now require that if the ATTL is exceeded, the cask must be filled with water or an inert gas within the default RTL, i.e., 1 hour, and prior to exceeding 400°C.
>
> For spent fuel unloading, a time limit, referred to as the Fuel Cladding Temperature Threshold Time Limit (FTTL), must be established with a maximum peak cladding temperature criterion of 400°C. The FTTL time limit is linked to 400°C fuel cladding temperature.
>
> Some typical time limits for water draining, and for achieving 180 Torr, have also been included for information.
>
> The 400°C temperature limit is set for fuel cladding and is applied to both MBF and HBF, as a threshold to prevent air oxidation in all cases. Once the air exposure conditions no longer apply, 570°C can again be applied for MBF.

For spent fuel, the applicant uses the drying criterion of 3 torr 30 minutes in ASTM Guide (ASTM). Packages with non-fuel waste may be dried using less than 6 torr for 2 hours minimum. The 6 torr pressure was selected because it represents a saturation pressure of water at approximately 4°C. A drying time of 2 hours was selected to ensure there was sufficient time under boiling conditions to allow the liquid water to boil off. Alternatively, the applicant can use Forced Helium Drying (FHD), if necessary. The staff finds all procedures acceptable.

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. 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 proposed design basis fuel in accordance with 10 CFR Part 71.

Further, the Certificate of Compliance states that the package must be prepared for shipment and operated in accordance with the Operating Procedures specified in Chapter 7 of the application and includes also additional conditions stemming from the staff's technical review.

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 Model No. HI-STAR 80 package and verifies their compliance with the requirements of 10 CFR Part 71.

The applicant stated in Section 8.1.3.2 "Pressure Testing," that:

(a) all containment boundary closures must be pressure tested, and

(b) the containment boundary shall be pressure tested at a test pressure of not less than 150% of the package cavity maximum normal operating pressure (MNOP) per 10 CFR 71.85(b) or at hydrostatic or pneumatic test pressure of 125% or 110%, respectively, of the package cavity design internal pressure in accordance with the ASME Code Section III, Subsection NB-6000, whichever is greater.

The applicant clarified, in an RAI response, that:

(a) the package cavity, including the inter-lid space and space between port closures, must be pressure-tested, and
(b) all containment boundary closures must be pressure tested and each containment boundary closure should be pressure tested at least once.

The staff reviewed the pressure testing of the containment boundary, including all containment closures, as described in Section 8.1.3.2 of the application, and finds it acceptable.

The applicant stated in Section 8.1.4 "Leakage Tests," that:

(a) the leakage rate testing on the containment system shall be performed in accordance with ANSI N14.5 (2014),

(b) leakage rate testing procedures shall be approved by an ASNT Level III specialist, and

(c) leakage rate testing shall be performed by personnel qualified and certified in accordance with the requirements of SNT-TC-1A and shall be performed in accordance with a written quality assurance program.

The applicant provided the containment system performance specifications and the package components tested and the type of leakage rate test in Tables 8.1.1 and 8.1.2 respectively. The applicant noted in Table 8.1.2 that:

(1) for packages containing high burnup fuel, pre-shipment and periodic leakage rate testing shall be performed on all containment boundary closure components to demonstrate that the containment system, as fabricated, provides the required level of containment, in accordance with ANSI N14.5 (2014), and

(2) for packages containing moderate burnup fuel or NFW, pre-shipment and periodic leakage rate testing shall be performed on either the inner or outer containment boundary closure seals (single barrier, non-moderator exclusion function).

The applicant specified the allowable leakage rates in Table 8.1.1 and the type of fabrication leakage rate test for each containment component in Table 8.1.2 for both a spent fuel package and a NFW package, in accordance with ANSI N14.5 (2014). The fabrication leakage rates are 1×10^{-7} ref-cm^3/sec (leak-tight) on the entire containment boundary of the HI-STAR 80, when used as a spent fuel transport package, and 5×10^{-7} ref-cm^3/sec for the entire containment boundary of the HI-STAR 80, when used as a NFW transport package. Staff noted that the calculated leakage rates of the HI-STAR 80 loaded with NFW do not exceed the allowable leakage rates for NCT and HAC, respectively, in compliance with 10 CFR 71.51(a)(1) and 71.51(a)(2), and

The pre-shipment leakage rate test is performed by the user, before each shipment, after the contents are loaded and the containment system is assembled, and the pre-shipment leakage rate test remains valid for one year. The applicant specified the allowable pre-shipment leakage rate for each containment component in Table 8.1.1 and the type of leakage rate test in Table 8.1.2, for both cases of the package transporting either spent fuel or NFW, in accordance with ANSI N14.5 (2014).

The periodic leakage rate test demonstrates that the containment system closure capabilities have not deteriorated over an extended period of use, and is performed by the user before each shipment if the previous leakage rate test has expired. The periodic leakage rate test remains valid for one year. The applicant specified the allowable leakage rate and the test method for each containment component in Table 8.1.1 and Table 8.1.2, respectively. The maintenance leakage rate test demonstrates that the containment system provides the required level of containment after undergoing maintenance, repair, and/or containment component replacement; and shall be performed prior to returning package to service.

The applicant stated in Section 8.1.7, "Thermal Tests," that the first fabricated HI-STAR 80 package shall be tested to confirm its heat dissipation capability. The thermal test is considered acceptable if the measured heat rejection capability is greater than the design basis minimum heat rejection capacity. Based on review of the statements in Section 8.1.7, the staff concludes that the description of the thermal test required for the HI-STAR 80 package is adequate.

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. Further, the Certificate of Compliance specifies that each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application while including also the conditions described above.

CONDITIONS

The following conditions are included in the Certificate of Compliance:

The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.

The package must be tested and maintained in accordance with Chapter 8 of the application.

Damaged fuel assemblies and fuel debris are not authorized for transportation.

Maximum allowable times, based on design basis maximum heat load, for the completion of wet transfer operations are defined in Table 3.3.6 of the application.

The package shall be transported exclusive use only, with the personnel barrier installed during transport.

Transport of fissile material by air is not authorized.

The package may be used in the U.S. for shipment of UO_2 fuel and Non-Fuel Waste meeting the above conditions.

CONCLUSION

Based on the statements and representations contained in the application, and the conditions listed above, the staff concludes that the Model No. HI-STAR 190 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. 9373, Revision No. 0,
on September 18, 2018.