NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9367	2	71-9367	USA/9367/B(U)F-96	1	OF	4

2. PREAMBLE

   a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, *Code of Federal Regulations*, Part 71, "Packaging and Transportation of Radioactive Material."

   b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Holtec International
1 Holtec Blvd.
Camden, NJ 08104

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Holtec International Report No. HI-2125175, *Safety Analysis Report on the HI-STAR 180D Package,* Revision No. 5, dated April 16, 2020.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HI-STAR 180D

(2) Description

The HI-STAR 180D package is designed for transportation of undamaged irradiated Uranium Oxide (UO_2) fuel assemblies. The fuel basket provides criticality control and the packaging body provides the containment boundary, helium retention boundary, moderator exclusion barrier, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the HI-STAR 180D packaging is approximately 2712-mm without impact limiters and approximately 3250-mm with impact limiters. The maximum gross weight of the loaded HI-STAR 180D package is 125 Metric Tons.

Fuel Basket

Metamic-HT, a metal matrix composite of aluminum and boron carbide, is the principal constituent material of the fuel basket, both as structural material and neutron absorber material. Two interchangeable fuel basket models, designated F-32 and F-37, contain either 32 or 37 Pressurized Water Reactor (PWR) fuel assemblies respectively, in regionalized and uniform loading patterns. The fuel basket features flux traps between some, but not all, cells.

5.(a)(2) Description (Continued)

Packaging Body

The cylindrical steel shell containment system is welded to a bottom steel baseplate and a top steel forging machined to receive two independent steel closure lids, with each lid being individually designated as a containment boundary component. The outer surface of the cask inner shell is buttressed with a monolithic shield cylinder for gamma and neutron shielding. Each closure lid features a dual metallic self-energizing seal system designed to ensure its containment and moderator exclusion functions. For this package, the inner closure lid inner seal and the inner closure lid vent/drain port cover inner seals are the containment boundary components on the inner lid; the outer closure lid inner seal and the outer closure lid access port plug seal are the containment boundary components on the outer lid.

Impact Limiters

The HI-STAR 180D package is fitted with two impact limiters fabricated of aluminum crush material completely enclosed by an all-welded austenitic stainless steel skin. Both impact limiters are attached to the body of the packaging with 16 bolts.

(3) Drawings

The packaging shall be constructed and assembled in accordance with the following Holtec International Drawing Numbers:

(a) HI-STAR 180D Cask — Drawing No. 8545, sheets 1-13, Rev. 8

(b) F-37 and F-32 Fuel Baskets — Drawing No. 8553, sheets 1-6, Rev. 7

(c) HI-STAR 180D Impact Limiter — Drawing No. 8552, sheets 1-5, Rev. 3

5.(b) Contents

(1) Type, Form, and Quantity of Material

(a) Only undamaged UO_2 fuel assemblies, with a Zr cladding type, meeting the specifications and requirements provided in Conditions 5.b(1)(b) through 5.b(1)(i), and with the characteristics listed in Table 7.D.1 of the application, are authorized for transportation.

5.(b)(1)(a) Continued

(b) Damaged fuel assemblies, i.e., assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks and which cannot be handled by normal means, as well as fuel debris, non-fuel hardware and neutron sources are not authorized contents.

(c) The maximum initial enrichment of any UO_2 assembly is 4.55 wt.% ^{235}U.

(d) The post-irradiation minimum cooling time, maximum burnup, maximum decay heat load, and minimum initial enrichment per assembly are listed in Tables 7.D.2 and 7.D.3 of the application. The cell numbering for the F-32 and F-37 fuel baskets is depicted in Figures 7.D.1 and 7.D.2 of the application, respectively.

(e) Regions and cells for regionalized loading of the F-32 and F-37 baskets are identified in Table 7.D.5 of the application. Table 7.D.6 of the application provides the minimum burnup requirements for the F-37 basket, based on initial enrichment. Fuel assemblies with a maximum of 50 cm of control rod insertion during irradiation may be transported in the F-37 basket.

(f) In-core operating limits for those assemblies that need to meet the burnup requirements in Table 7.D.6 of the application are listed in Table 7.D.7 of the application. In addition, the maximum average fuel temperature during irradiation is 1251°K (977.9°C).

(g) For those spent fuel assemblies that need to meet the burnup requirements specified in Table 7.D.6 of the application, a burnup verification shall be performed in accordance with Appendix 7.E of the application.

(h) Allowable loading patterns and fuel specifications for each basket region are referenced in Tables 7.D.2 and 7.D.3 of the application. Alternative fuel specifications for each regional loading pattern are presented in Table 7.D.4 of the application.

(i) The maximum decay heat is 33.08 kW for the F-32 basket and 36.4 kW for the F-37 basket.

5.b.(2) Maximum Quantity of Material Per Package

32 or 37 PWR fuel assemblies, as described in 5(b)(1), in the F-32 or F-37 basket, respectively.

5.(c) Criticality Safety Index (CSI)= 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

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

(b) The package shall meet the acceptance tests and be maintained in accordance with Chapter 8 of the application.

7. The personnel barrier shall be installed and remain installed while transporting the package, if necessary, to meet package surface temperature and/or package dose rates requirements.

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

9. Transport by air of fissile material is not authorized.

10. Expiration Date: June 30, 2025

REFERENCES:

Holtec International application "Safety Analysis Report on the HI-STAR 180D Package," Revision No. 5, dated April 16, 2020.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

John B. McKirgan Digitally signed by John B. McKirgan
Date: 2020.06.12 13:24:25 -04'00'

John McKirgan, Chief
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety
and Safeguards

Date: June 3, 2020

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

SAFETY EVALUATION REPORT
Model No. HI-STAR 180D Package
Certificate of Compliance No. 9367
Revision No. 2

SUMMARY

By letter dated April 25, 2019, Holtec International (Holtec or the applicant) submitted an amendment request for Certificate of Compliance (CoC) No. 9367 for the Model No. HI-STAR 180D package. On November 27, 2019, Holtec responded to staff's request for additional information letter dated October 22, 2019. The applicant provided Revision No. 5 of the application by letter dated April 17, 2020.

The structural changes made to the design include changes affecting the structural integrity of the neutron shielding components, changes to the inner closure lid design, changes to the design of the trunnions and the lifting analysis, changes to the specifications of the impact limiter materials, changes to the containment boundary and bolting analyses, changes to evaluations of certain normal conditions of transport (NCT), and changes to fuel basket welds. The applicant also proposed alternatives to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code requirements for package construction.

The thermal-related changes associated with this certificate of compliance revision included replacing lead with steel in the inner closure lid and analyzing the changes in Time-to-Boil results, changes in engineered gaps, changes in properties (thermal) to Holtite-B, analyzing the effect of burnup profile and decay heat profile, analyzing the effect of performing a transient thermal analysis due to time-varying insolation, analyzing the effect of impact limiter thermal conductivity, analyzing the effect of asymmetric geometry, and analyzing thermal effects of the monolithic shield cylinder surface enhancement features.

Regarding containment, changes were made to the design of the closure lid region and seal options, specifically: the revised inner closure lid design removes the lead shielding, and seal part drawing numbers, seal / groove dimensions, seal seating load, and seal jacket material have all been changed.

The applicant also added alternative fuel specification sensitivity analyses to determine dose rates from all alternative loading patterns and revised the application to include an additional evaluation scenario of potential fuel reconfiguration under HAC.

Based on the statements and representations in the application, and the conditions listed in the CoC, the U.S. Nuclear Regulatory Commission staff (the staff) concludes that the package meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

EVALUATION

1.0 GENERAL INFORMATION

The applicant made several changes to the design of the packaging, including:

- Removing the lead gamma shielding and replacing some Holtite-B neutron shielding with a steel gamma shielding component for the inner closure lid. Adjusting thicknesses of some shielding components, e.g., the bottom lead gamma shield, the bottom neutron shield, etc., to optimize shielding performance.

- Adding engineered gaps for lead and Holtite components.

- Specifying the thermal expansion of lead and Holtite shielding components. Evaluating the temperature dependence of impact limiter crush material properties and reducing the temperature limit for the impact limiter crush material from 260°F to 250°F.

- Specifying the minimum B-10 Areal Density for finished Metamic-HT panels. The minimum hydrogen density of Holtite-B is specified as a limiting material property.

- Revising the Metamic-HT sourcebook to (i) introduce a fracture toughness program to study the crack propagation in the material assumed to contain flaws or defects, and (ii) correct the specific heat value at 100ºC.

- Providing impact limiter crush material reference density ranges.

- Evaluating the effect of a burnup profile on the computed temperatures during transport. Performing a transient evaluation to evaluate the sensitivity of applying solar insolation as 24-hour averaged value to package external surfaces.

- Adding alternative fuel specification sensitivity analyses to determine dose rates from all alternative loading patterns for locations where high dose rates are expected.

- Evaluating fuel cladding integrity under cask reflooding conditions.

- Replacing the top and bottom trunnion threads with an anti-rotation locking system to ensure that the trunnions would not rotate during package down-ending or up-ending.

- Evaluating the local bird caging scenario to evaluate potential high burnup fuel reconfiguration.

- Providing operational step(s) to the cask loading/closure procedures for the removal of standing water from closure lid bolts holes.

- Using the latest edition of ANSI N14.5 (2014) for leak testing.

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

(a) HI-STAR 180D Cask Drawing No. 8545, Sheets 1-13, Rev. 8

(b) F-37 and F-32 Fuel Baskets Drawing No. 8553, Sheets 1-6, Rev. 7

(c) HI-STAR 180D Impact Limiter Drawing No. 8552, Sheets 1-5, Rev. 3

The staff concludes that the information presented in the application provides an adequate basis for the evaluation of the Model No. HI-STAR 180D package against 10 CFR Part 71 requirements for each technical discipline.

2.0 STRUCTURAL AND MATERIALS EVALUATION

The staff has reviewed the proposed changes to the HI-STAR 180D transportation package to verify that the licensee has adequately evaluated the structural performance of the package and demonstrated that the system meets the regulations of 10 CFR Part 71.

While numerous changes affecting the previously approved design were made, the staff's structural review focused primarily on those changes which impacted the structural performance of the package. These include changes affecting the structural integrity of the neutron shielding components, changes to the inner closure lid design, changes to the design of the trunnions and the lifting analysis, changes to the specifications of the impact limiter materials, changes to the containment boundary and bolting analyses, changes to evaluations of certain normal conditions of transport (NCT), and changes to fuel basket welds.

2.1 Changes to Shielding Components

Holtec has proposed changes to the shielding of the package including the use of a new composition of the Holtite-B neutron shielding material. In support of the new shielding analysis, calculations of the differential thermal expansion of both the lead and Holtite shielding components were updated in the structural calculation package for HI-STAR 180D, Holtec Report No. HI-2125252. The applicant has included gaps surrounding the shielding components to ensure any stresses induced in the surrounding members from differential thermal expansion do not cause the members to exceed applicable stress limits. Based on the updated calculations of the differential thermal expansion and the inclusion of adequate gaps, the staff concludes that the use of a new composition of the Holtite-B neutron shielding material is acceptable.

The supporting calculations of the pressure limits for the neutron shielding cavities and the subsequent pressures at which pressure relief devices in the cavities are set to open were also updated. All Holtite-B component enclosures in the HI-STAR 180D package are equipped with pressure relief devices. The applicant has performed structural analyses to determine the pressure limits for each Holtite-B enclosure that would prevent the plates and welds forming the cavities from undergoing plastic deformation. These analyses are reported in Calculation 25 of Holtec Report No. HI-2125252. The resulting pressure limits are listed in Table 2.1.1 of the application for the different Holtite-B enclosures with the set pressures limited to 35 psig.

Holtec has included a discussion of the Holtite-B enclosure cavities and the pressure relief devices in Section 2.6.1.4.1 of the application. The applicant has also included the pressure relief devices in the HI-STAR 180D licensing drawing, Holtec Drawing No. 8545. Note 16 of that drawing describes the pressure relief devices and lists the maximum pressure at which the relief devices are to be set for the different Holtite-B enclosures; Note 16 limits the set pressures to 35 psig. Based on the structural analyses of the neutron shielding cavities and the requirements of the pressure relief devices in the licensing drawings, the staff finds the changes to neutron shielding cavities and neutron shielding cavity pressure relief devices to be acceptable and sufficient to meet the requirements of 10 CFR 71.51(a).

The evaluation of the welds connecting the monolithic shield cylinder to the containment boundary base plate was updated to consider new loads associated with Hypothetical Accident Conditions (HAC). The applicant has reanalyzed the welds in Revision 9 to Holtec Report No. HI-2125252 and updated Section 2.7 of the application discussing the results. Based on the results of the structural analysis, the staff finds that the welds connecting the monolithic shield cylinder to the containment boundary base plate have adequate structural integrity to satisfy the requirements of 10 CFR 71.73.

2.2 Changes to Inner Closure Lid Design

Holtec has proposed altering the design of the inner closure lid of the package by removing the lead gamma shielding and replacing a portion of the Holtite-B neutron shielding with steel gamma shielding. The licensee has updated Holtec Drawing No. 8545 to reflect the new inner closure lid design. The effects of this change under hypothetical accident conditions (HAC) were evaluated with finite element analysis modeling. The licensee presents the results of the finite element analyses in the finite element analysis report, Holtec Report No. HI-2125251, and the structural calculation package for HI-STAR 180D, Holtec Report No. HI-2125252, and discusses the conclusions of these evaluations in Chapter 2 of the SAR. The licensee has revised the lid lifting calculations in Holtec Report No. HI-2125252 to support the new lid design. These analyses show that the proposed lid design resulted in negligible differences in the structural performance and continued compliance with Part 71 requirements.

2.3 Changes to Trunnion Design

The applicant has proposed a new design of the HI-STAR 180D top and bottom lifting trunnions. The proposed change replaces the previously approved threaded trunnions with trunnions incorporating an anti-rotating locking system. The lifting trunnion stress analysis, Calculation 1, in Holtec Report No. HI-2125252 and Section 2.5.1, “Lifting Devices,” was updated and calculation 1 demonstrates that the trunnion design satisfies the acceptance criteria of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.” Calculation 26 of Holtec Report No. HI-2125252 demonstrates the adequate structural performance of the anti-rotation system. Holtec Drawing No. 8545 was revised to show the latest trunnion design. The analyses provided in Holtec Report No. HI-2125252 show that the proposed trunnion design is sufficient to meet the structural standards of 10 CFR 71.45(a).

2.4 Changes to Impact Limiters Material Specifications

The maximum operating temperature of the impact limiter crush material has been lowered to below the normal transport maximum temperature for the crush material listed in Table 3.1.1 of the application. The new temperature range is presented in Table 2.2.10, “Critical Characteristics of the AL-STAR Impact Limiter Crush Material and Fastener Strain Limiters,” of the application. The applicant has also revised the discussion of the temperature dependence of impact limiter crush material properties and the description of the material testing to demonstrate that the crush strength limits listed in Table 2.2.10 will be satisfied over the range of operating temperatures. As the crush strength limits will continue to be satisfied, the basis for the staff’s previous approval the HI-STAR 180D impact limiters and crush material remains valid, and the staff finds the lowered maximum operating temperature of the impact limiter crush material acceptable.

Holtec has analyzed effects of variability in the density of the crush material on the performance of the HI-STAR 180D impact limiters. This density variability is due to material and fabrication tolerances and has an effect on the structural properties of the crush material. The applicant conducted a sensitivity study on the performance of the impact limiters with varying crush material density described in the finite element analyses, Holtec Report No. HI-2178010. The sensitivity analysis shows that minor variation in the density of the crush material have a negligible effect on the performance of the structural impact limiters. The applicant has reanalyzed the governing HAC drop events considering the different crush material densities using finite element analysis software as discussed in Holtec Report No. HI-2125251.

Holtec has updated Table 2.2.10 of the application to include the range of densities for the impact limiter crush material and Section 2.7, "Hypothetical Accident Conditions," of the application with the results of the reanalyzed, finite element drop tests. Based on the results of the finite element analysis drop tests supported by the sensitivity test, the staff finds that the HI-STAR 180D package with impact limiters composed of crush material with the specified density range sufficiently meets the requirements of 10 CFR 71.73.

2.5 Changes to the Containment Boundary and Bolting Analyses

Holtec has revised the preload and torque values for the inner closure lid cover plate bolts and the outer closure lid access port plug to be consistent with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III, Division I, Subsection NB requirements. The licensee had previously made the safety case for these preloads, but the licensee is proposing this change to be compliant with the ASME code. The licensee has performed calculations in accordance with ASME B&PV Code Section III, Division I, Subsection NB to determine the proposed preload values as documented in Holtec Report No. HI-2125252, Revision 9 and updated Chapter 7 of the application to reflect the new preload values. Based on the structural calculations, the staff finds the revised the preload and torque values for the inner closure lid cover plate bolts and the outer closure lid access port plug acceptable.

Holtec has performed structural calculations and addressed potential stresses induced by interference fit on the containment shell. As the monolithic shield cylinder cools, the cylinder constricts the containment shell and produces secondary stresses. Holtec has assessed the stresses caused by temperature change and interference fit in Holtec Report No. HI-2125252, Revision 9 and shown that these stresses do not lead to structural instability of the components. A discussion of these stresses is added in Section 2.3.1 of the application and the results of the structural analysis of the stresses are provided in Section 2.6. Based on the results of the structural analysis, the staff finds that the applicant has adequately considered the effects of the interference fit stresses on the HI-STAR 180D package.

The structural analysis of the inner and outer closure lid bolts was updated. The new analysis uses finite element analysis software to analyze the bolts, as discussed in Holtec Report No. HI-2125251, Revision 7. The results from the finite element models are then used in the evaluations of the bolts, as presented in Holtec Report No. HI-2125252, Revision 9. The updated bolting analysis replaces a semi-empirical analysis and considers the effects of differential thermal expansion due to normal operation, extreme cold, and fire accident conditions on the closure lid bolts, the bolt threads, and the gaskets. The analyses show that the containment boundary is not breached. Sections 2.6 and 2.7 of the application were updated to include the results of the bolting analysis. The staff finds that the updated analysis of the closure lid bolts is sufficient to demonstrate that the Model No. HI-STAR 180D package meets the requirements of 10 CFR Part 71.

The applicant has proposed adding the option to repair cask bolt holes by installing threaded inserts in place of the damaged bolt holes, as shown in Note 58 to Holtec Drawing No. 8545. The drawing note requires that the repaired configuration with the installed threaded insert be evaluated and required to meet applicable stress limits prior to repair.

The licensee has included discussion of the inspection, evaluation, and repair of the cask closure fasteners in Section 8.2.3.4 of the SAR which requires the repaired joint to meet the ASME Code stress limits required of the initial joint and the material and manufacturing process testing required of the repair to be in accordance with ASME B&PV Code Section III, Division 1, Subsection NB.

Based on the description and commitments of the repair process, the staff finds that the inclusion of the option to use threaded inserts for repairing cask bolt holes is acceptable and meets the requirements of 10 CFR 71.43(c) for positive closure.

2.6 Changes to Evaluations of Certain Normal Conditions of Transport

The evaluations of the NCT test requirements for water spray, free drop, and penetration, presented in Appendix I of Holtec Report No. HI-2125251, Revision 7, were modified. The applicant determined the water spray test poses no significant risk to the containment boundary or shielding systems of the package due to the large thermal inertia, previously analyzed leak tightness, and corrosion resistance of the package.

The applicant performed an evaluation of an additional NCT free drop scenario. The added evaluation considers a one-foot, NCT drop of the package in a top down end drop orientation. The finite element modeling analysis results show that stress levels in the containment boundary are within the design basis stress limits required for normal conditions, and the drop does not result in any loss of shielding material.

The updated puncture analysis considers the penetration test of 10 CFR 71.71(c)(10) to be bounded by the analysis performed for the HAC puncture test required by 10 CFR 71.73(c)(3), which shows that the stress levels for the HAC puncture test are acceptable for the design basis stress limits required for normal conditions.

The applicant has updated Section 2.6 of the application discussing these evaluations and presenting certain results from the analyses. The staff finds that the NCT evaluations for water spray, free drop, and penetration sufficiently demonstrate that the HI-STAR 180D package will maintain the structural integrity of the containment and shielding systems under NCT and meet the requirements of 10 CFR 71.71.

2.7 Changes to Fuel Basket Welds

The applicant has revised the dimensions of the friction stir welds along the length of the exterior of the F-37 and F-32 fuel baskets. Note 16 of the basket drawings, Holtec Drawing No. 8553, has been updated to more accurately describe the effective throat size of the weld which is smaller than the thickness of the basket panel. Holtec has evaluated the structural integrity of the welds considering the accurate weld size and the weld quality factor for single groove weld joints listed in Table NG-3352-1 of the ASME B&PV Code Section III, Division I, Subsection NG in Appendix 17B of Holtec Report No. HI-2125252, Revision 10. The evaluation uses finite element analysis results to qualify the basket welds to withstand the HAC 30-foot drop. The staff finds the basket drawings adequately describe the basket corner friction stir welds and the weld evaluation sufficiently demonstrates the structural integrity of the welds.

2.8 Holtite-B Neutron Shielding Material

The applicant discussed design revisions to incorporate spatially-distributed particles of copper to enhance the thermal conductivity of the Holtite-B material, as defined in Table 8.1.11 of the application. The staff confirmed that the properties in the application (and used in the supporting safety analyses) are consistent with those identified in the Holtite-B Sourcebook (Report HI-2167314, Revision 5).

The applicant clarified that the Holtite-B Sourcebook reports Holtite-B properties (bulk density, hydrogen density) without any addition of copper and are, therefore, different from the values reported in Table 8.1.11 of the application. However, the property values used in the application's shielding evaluations are conservative when compared to property values accounting for the copper addition.

The applicant also confirmed that, per Section 8.1.5.3 of the application, each manufactured lot of Holtite-B neutron shield material is tested to verify that the boron carbide content, hydrogen density and Holtite-B material density meet the requirements in Table 8.1.11 of the application.

The applicant also confirms that Holtite- B does not serve a structural function in the HI-STAR 180D package. The staff considers the justification to be acceptable.

2.9 Metamic-HT Basket Structural Material and Neutron Absorber

The applicant revised the discussion regarding the use of different ASME Code editions for friction stir welding (FSW) of Metamic-HT. Section 8.1.5.4 of the application delineates the requirements for FSW Procedure Qualification, Welder Operator Qualification and Welded Coupon Test. The applicant cited ASME Section III, NCA-1140, which allows for either all items of a nuclear power plant be constructed to a single Code Edition and Addenda, or each item be constructed to individually specified Code Editions and Addenda. F.

The applicant clarified that the friction stir weld procedure qualification was originally performed to the 2007 Edition of the ASME Code with certain essential variables that do not change. The applicant further clarified that there is no requirement to requalify FSW weld procedures to a later edition of the Code.

All welding by FSW process shall meet the applicable requirements of ASME Section IX (2013 edition, first edition to incorporate FSW requirements). Further, per ASME Section IX, QG-108, joining procedures, procedure qualifications, and performance qualifications that were made in accordance with Editions and Addenda of Section IX, as far back as the 1962 Edition may be used in any construction for which the current Edition has been specified. Therefore, the staff agrees with that approach.

2.10 Impact Limiters

The applicant revised the discussion on acceptable crush strength ranges (for Type 1 and Type 2 impact limiter materials) per the temperature range pertinent to package operation (Table 2.2.10 of the application). The critical characteristic of the impact limiter crush material is its crush strength over the operating temperature range, which are identified in the drawing package (per Section 8.1.5.3 of the application). The applicant identified the temperature limit range of applicability of these properties in Table 3.2.10 of the application.

The applicant further provided results from sample crush material tests and clarified that the properties are obtained in accordance with ASTM D7336 (“Standard Test Method for Static Energy Absorption Properties of Honeycomb Sandwich Core Materials”). As discussed in Section 8.1.5.2. of the application, similar standardized test methods are to be used for the procured crush materials for HI-STAR 180D impact limiters.

The applicant further clarified that the ranges for Type 1 and Type 2 crush materials (as defined in the impact limiter design drawing 8552) are sufficiently large to account for uncertainties and variations in critical properties with manufacturing, characterization methods and temperature. The supporting analyses performed in the finite element analysis for the 9-m (30-ft) drop test (Report HI-2125251, Revision 8, and Section 2.7 of the application) consider the maximum crush strength values to predict maximum decelerations and minimum crush strength values to predict maximum crush depth.

Further, as discussed in Section 8.1.5 of the application, the vendor supplying the crush material is to ensure that the specified ranges are satisfied while accounting for uncertainties and variations in critical properties with manufacturing, characterization methods and temperature. The staff considers the revisions to the mechanical properties and property tolerance for the impact limiter materials to be acceptable.

The applicant also revised the thermal conductivities assumed in the thermal evaluation of the package for the test conditions per 10 CFR 71.71(c) and 71.73(c) (Table 3.2.2 of the application). The applicant identified reasonably bounding values for the thermal conductivities in both the axial and radial directions, and clarified that these conservative values were assumed for the analysis of the post-fire cooldown (per the thermal test in 10 CFR 71.73(c)(4)).

The applicant justified these properties by discussing the behavior of the material during the thermal test scenario (Table 3.4.1 of the application). Further, the applicant made assumptions to maximize heat input during the thermal test and minimize post-fire cooldown heat dissipation. Therefore, the staff considers these revisions to be acceptable.

2.11 Lead Slumping

The applicant clarified that a small quantity of lead, used for shielding in the packaging, is included in the structural evaluation model. The applicant defined the material properties assumed in the structural evaluation in Table 2.2.11 of the application, which references Report HI-2125251. The staff noted that the properties in the application and the cited reference are different. The applicant acknowledged and justified the difference.

The application cited limited data in the temperature range of interest, which may increase uncertainty when interpolating the available data. However, the applicant stated that the mechanical properties listed in Report HI-2125251 lead to conservative results for lead slump during the thermal test condition per 10 CFR 71.71(c)(4).

The applicant provided a quantitative comparison as justification that the mechanical properties in Report HI-2125251 yield conservative result leading to greater deformations of the lead material. Therefore, the staff considers the revision to be acceptable.

2.12 Design Code Alternatives

The applicant proposed alternatives to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code requirements for package construction (as defined in

Table 2.1.14 in the application). These alternatives are reference in Section 8.1.2 of the application, which is a CoC condition.

- Regarding the cask containment system, the applicant proposed an alternative to the requirement of ASME B&PV Code, Section III, Division 1, Subsection NB, paragraph 2330. The requirement states that the user is to establish a nil-ductility-transition temperature (Tndt) for impact testing of the material. The applicant proposed, instead, to follow the guidance in Regulatory Guide (RG) 7.11 (for ferritic steels with maximum wall thickness less than 4 inches) and RG 7.12 (for ferritic steels exceeding a maximum wall thickness of 4 inches). The RGs define a maximum Tndt. The applicant stated that the Tndt defined in the application (Table 8.1.9) for the steel material is lower than that defined in the RG. Therefore, the staff considers the alternative to be conservative and acceptable. The staff notes that the alternative is also stated in updated discussion for Metamic-HT crack propagation in Section 2.1.2.2 of the application.

- Regarding the cask containment system and other non-structural (shielding) components, the applicant proposed an alternative to the requirement of ASME B&PV Code, Section III, Division 1, Subsection NB, paragraph 4622. The requirement states that the user is to post-weld heat treat all welds, including repair welds. The applicant proposed to not conduct post-weld heat treatment in specific welds that are less than the containment boundary, and justified the alternative by stating that these welds are exempt from post-weld heat treatment per ASME B&PV Code, Section III, Subsection NB, Table NB-4622.7(b)-1. The staff reviewed Table NB-4622.7(b)-1 to confirm that the nominal thickness of the nickel-alloy and low-alloy steel components identified in Table 2.1.14 allows for exemption to post-weld heat treatment. Therefore, the staff considers the alternative to be acceptable.

- Regarding the cask containment system, the applicant proposed an alternative to the requirement of ASME B&PV Code, Section III, Division 1, Subsection NB-5120. The requirement states that the user is to perform radiographic examination after post-weld heat treatment of the welds. The applicant proposed to not conduct radiographic examination of the pressure-retention boundary welds after post-weld heat treatment. The applicant stated that all welds (including repairs) will have passed radiographic examination prior to post-weld heat treatment of the entire containment boundary. The applicant justified the exemption by stating that confirmatory radiographic examination after post-weld heat treatment is not necessary because this process is not known to introduce new weld defects in nickel steels. The staff considers the justification to be reasonable and, therefore, to be acceptable.

- Regarding the monolithic shield cylinder, the applicant proposed an alternative to the requirement for standard impact test temperature per ASME B&PV Code, Section II for the materials specified in Table 2.1.14 of the application. The applicant proposed an alternative for testing at a higher impact test temperature of −40°F [−40°C] per Section III, Division 1, Subsection NF.

    - Section II, Part A, "Specification for Carbon and Low-Alloy Steel Forgings, Requiring Notch Toughness Testing for Piping Components, SA-350/SA-350-M" defines a standard impact test temperature of −50°F [−46°C] for the LF2, Class 1 grade (Table 4). The standard also identifies a requirement for minimum average impact energy of 15 ft-lbf [20 J] and a minimum impact energy permitted for one of the tested specimens of 12 ft-lbf [16 J] (Table 3).

- Section II, Part A, “Specification for Steel Castings, Ferritic and Martensitic, for Pressure-Containing Parts, Suitable for Low-Temperature Service, A-352/SA-352M”, defines a standard impact test temperature of −50°F [−46°C] for the LCC grade (Table 1). The standard also identifies a requirement for minimum average impact energy of 15 ft-lbf [20 J] and a minimum impact energy permitted for one of the tested specimens of 12 ft-lbf [16 J] (Table 1).

- Per paragraph NF-2331, “Charpy V-Notch Testing for Absorbed Energy Values”, the test results of three specimens, collectively and singly, shall meet the respective requirements of Table NF-2331(a)-4 for Class 1 supports. For SA-350 LF2, Class 1 material (minimum yield strength of 36 ksi [250 MPa]), Table NF-2331(a)-4 defines a minimum average absorbed energy value (C_v) of 25 ft-lbf [34 J] and a mimimum Cv for one of the tested specimens of 20 ft-lbf [27 J]. Therefore, the staff considers that the Cv criteria in Subsection NF is more conservative than that of Section II. Further, the lowest service temperature in Subsection NF is consistent (or conservative) relative to the temperature criteria for the tests in 10 CFR 71.71 and 71.73. Therefore, the staff considers the alternative proposed by the applicant to be acceptable.

2.13 Cask Reflooding Operations

The applicant revised the discussion regarding the reflood analysis in Section 2.6.1.3.5 of the application. The applicant confirmed that, per Table 1.4.1 of the application, cask reflooding is an unloading operation and is not considered to occur prior to cask transport. The applicant stated that cask reflood analysis is also not expected for certifications of transportation packages per NUREG-1617, Regulatory Guide 7.9, ISG-11, ISG-19 and ISG-8.

The applicant further stated that the analysis is performed as a defense-in-depth analysis to support the multi-layered approach for safe transport of high burnup fuel, as described in Chapter 1 of and summarized in Table 1.4.1 of the application.

The staff reviewed the spent fuel assumptions (end-of-life condition, material properties) in the analysis and considers them reasonable for demonstrating that the maximum total strain in the fuel cladding due to the reflood event is well below the failure strain limit of the material. Therefore, the staff considers the revision to be acceptable.

2.14 Evaluation Findings

Based on the review of the statements and representations in the application, the NRC staff concludes that the changes to the structural design have been adequately described and evaluated and that the package has adequate structural integrity to meet the requirements of 10 CFR Part 71. The staff reviewed the structural codes and standards used in the package design and finds that they satisfy the requirements of 10 CFR 71.31(c). The staff reviewed the lifting system for the package and concludes that it satisfies the standards of 10 CFR 71.45(a) for lifting attachments. The staff reviewed the structural performance of the packaging under the normal conditions of transport prescribed in 10 CFR 71.71 and concludes that the package satisfies the requirements of 10 CFR 71.51(a)(1) and 10 CFR 71.55(d)(2). The staff reviewed the structural performance of the packaging under the hypothetical accident conditions prescribed in 10 CFR 71.73 and concludes that the packaging has adequate structural integrity to satisfy the subcriticality, containment, and shielding requirements of 10 CFR 71.51(a)(2) and 10 CFR 71.55(e). The staff reviewed the analysis of the package closure system for normal and accident pressure conditions and concludes that the system satisfies the requirements of 10 CFR 71.43(c) for positive closure.

The staff has reviewed the materials sections of the application and concludes that the applicant has met the requirements of 10 CFR 71.33. The applicant described the materials used in the transportation package in sufficient detail to support the staff's evaluation. The staff also concludes that the applicant has met the requirements of 10 CFR 71.31(c). The applicant identified the applicable codes and standards for the design, fabrication, testing, and maintenance of the package and, in the absence of codes and standards, has adequately described controls for material qualification and fabrication. The staff also concludes that the applicant has met the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a). The applicant demonstrated effective materials performance of packaging components under normal conditions of transport and hypothetical accident conditions. Based on the review of the statements and representations in the application, the NRC staff concludes that the materials used in the package design have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The objective of the review was to verify that the HI-STAR 180D package thermal design was adequately described and evaluated under normal conditions of transport and hypothetical accident conditions, as required per 10 CFR Part 71. The thermal-related changes associated with this certificate of compliance revision included replacing lead with steel in the inner closure lid and analyzing the changes in Time-to-Boil results, changes in engineered gaps, changes in properties (thermal) to Holtite-B, analyzing the effect of burnup profile and decay heat profile, analyzing the effect of performing a transient thermal analysis due to time-varying insolation, analyzing the effect of impact limiter thermal conductivity, analyzing the effect of asymmetric geometry, and analyzing thermal effects of the monolithic shield cylinder surface enhancement features. Regulations applicable to the thermal review include 10 CFR 71.31, 71.33, 71.35, 71.43, and 71.51.

3.1 Description of Thermal Design

There were no significant changes with the HI-STAR 180D package's thermal design or the content (i.e., decay heat). However, the inner closure lid was modified by replacing the lid's interior lead material with steel. According to the Document 2178010-NRC "Summary of Proposed Changes" (page 33 of 36), one impact of changing the lid material and its corresponding thermal inertia was the need to update the Time-to-Boil related calculations. According to Section 3.3.3 of the application, and the response to RAI 3-6 (Enclosure 1 to Holtec Letter 2178011-NRC), either an adiabatic heat-up calculation or the described FLUENT thermal model can be used to determine appropriate Time-to-Boil limits. Table 3-6.1 of RAI 3-6 response provided a benchmarked example calculation result which showed a 22-hour Time-to-Boil period based on the adiabatic calculation and a 22.4-hour Time-to-Boil period based on the 3D FLUENT model.

Other proposed modifications in the revision dealt with changes in certain engineered gaps, including those components below the containment baseplate (item DI06 in Document 2178010-NRC "Summary of Proposed Changes"), changes in the cold axial gap between the basket and cask cavity (item DI15 in Summary of Proposed Changes), and changes in the gaps associated with the shielding dose blocker parts (item PC-9 in Summary of Proposed Changes). According to the Summary of Proposed Changes, DI06 was related to engineered gaps for the lead and Holtite-B components and reflected an updated coefficient of thermal expansion for

Holtite-B. The evaluation of the gaps was discussed in the report “Holtite-B Applications Report for the HI-STAR 180D” and, according to the Summary of Proposed Changes, the engineered gaps did not change the thermal evaluations. According to the Summary of Proposed Changes, DI15 was related to a revised reference temperature; this affected the reported tolerance for the cold gap between the cask cavity and basket in Cask Licensing Drawing 8545 Rev. 7; it was noted that the cold axial gap between the basket and cask cavity was revised to align the cask drawing (8545 Rev. 7) and the thermal calculation package (document HI-2125241, Rev 6). In addition, the Summary of Proposed Changes reported that the changes in item PC-9 dealt with Bottom Lead Gamma Shield, Bottom Neutron Shield, and the Bottom Lead Gamma Shield Cap Plate; the impact of the changes (as specified in drawing 8545 Rev. 7) were evaluated in the HI-2125241 document.

Finally, regarding the significance of these gaps on thermal performance, the response to RAI 3-1 noted that heat transfer is primarily through the metal regions in and around the Holtite regions and indicated the gaps would have only a second-order effect on calculated temperatures and pressures. The impact of these gaps may need to be considered in future amendments where second-order effects become important, such as changes (e.g., increased decay heat) that result in increased temperatures of components that are near allowable limits (e.g., Holtite-B).

3.2 Material Properties and Component Specifications

The applicant provided thermal conductivities (axial and radial) for Type 1 and Type 2 impact limiter crush material in Table 3.2.2 of the application. The results from using Type 1 conductivity values, reported in Table 3.1.3, indicated the fuel cladding has NCT and HAC (cooldown) temperatures of 363°C and 393°C, respectively. Likewise, according to Table 3.1.1 and Table 3.1.3, the inner closure lid inner seal had a maximum NCT temperature of 169°C and the outer closure lid inner seal had a maximum HAC (cooldown) temperature of 219°C. In addition, Table 3.4 and Table 7.20 of the HI-2125241 thermal calculation presented updated axial and radial thermal conductivity values using Type 2 impact limiter material and corresponding thermal results. The fuel cladding temperatures at NCT and HAC were reported as 363°C and 395°C, respectively. In addition, the inner closure lid inner seal maximum temperatures at NCT and HAC (cooldown) were reported as 170°C and 214°C, respectively. It is observed that the cladding temperatures at NCT and HAC when using either impact limiter material are below the allowable values reported in Table 3.2.11. The containment boundary seal temperatures at NCT, when using either impact limiter material, are below the allowable values reported in Table 3.2.12 of the application. Likewise, Table 3.2.12 and Figure 3.4.1 indicated that metallic seal temperatures were less than the short-term allowable temperatures for the thermal hypothetical accident condition.

According to the Summary of Proposed Changes (item PC-4), the package includes an enhanced Holtite-B neutron absorber with an updated composition. The response to RAI 3-5 showed that the Holtite-B thermal properties (including thermal conductivity, density, specific heat) used in the thermal evaluations were conservative relative to actual properties defined in the Holtite-B sourcebook (HI-2167314); thus, the thermal evaluations would tend to result in bounding temperatures. In particular, Table 3.5-1 of the RAI response indicated a 0.4 W/m-K thermal conductivity value used in the application evaluations whereas the property value in the Holtite-B sourcebook was 0.46 W/m-K. In addition, the response to RAI 3-2 summarized calculations which indicated the percentage of decomposition Holtite-B gases in the air immediately surrounding the package surface as approximately 0.002% and the energy input from combustion of the Holtite-B decomposition products being negligible compared to the thermal input from the 30-minute hypothetical accident condition.

3.3 General Considerations

According to the Summary of Proposed Changes (item DI13), the thermal analysis assumed the decay heat profile was equivalent to the burnup profile, which tended to underpredict decay heat near the center of the fuel assemblies and overpredict decay heat at the ends of the fuel assemblies. The applicant ran a sensitivity calculation to compare component temperatures assuming either the burnup profile stated in the application and an actual decay heat profile. Table 7.19 of the HI-2125241 thermal calculation showed that, for the actual heat load profile, the fuel basket, fuel cladding, and containment shell temperatures would be approximately 2°C greater. It is noted that the applicant did not update the application to reflect the more representative decay heat profile. However, the appropriate choice of profile when modeling decay heat may have to be considered if future amendments result in package temperatures near allowable values (e.g., Holtite-B).

The application described the effect of the insolation modeling methodology on Holtite-B temperatures because the Holtite-B neutron shield was near its allowable temperature. According to Figure 3.3.9 and Table 3.3.13, a steady-state insolation analysis resulted in a Holtite-B temperature of 198.65°C, which is approximately 5°C less than the 204°C allowable value reported in Table 1.2.12.

In addition, the thermal section titled "Determination of Solar Heat Input" discussed a transient analysis, whereby the solar insolation was applied in a 12-hour on/off cycle within a 24-hour period, in order to determine the transient's effect on the Holtite-B temperature. The result of the transient analysis, provided in Table 3.3.13, indicated that Holtite-B could increase to 201.56°C, which is approximately 2°C less than the allowable value. The application mentioned that the steady-state insolation analysis would continue to be used since the difference between the two analyses was small.

However, the choice of insolation boundary condition methodology (steady-state versus transient) may have to be reviewed in any future amendment (e.g., requesting higher decay heat) to ensure package temperatures, including Holtite-B, do not exceed allowable values; it is noted that the shielding chapter of the SER discusses the impact of the Holtite-B condition.

Section 7.8.1 of HI-2125241 mentioned that a 200 W/m^2 package surface insolation value is more realistic than the calculation's assumed 340 W/m^2 value in the application and noted, in Table 7.18, the 200 W/m^2 insolation value reduced many package component temperatures (including Holtite-B neutron shield) between 3°C and 5°C. However, per page 6 of HI-2125241, the application's assumed 340 W/m^2 value was computed and used to account for the greater surface area of the fin (which is not explicitly modeled). The discussion in Section 7.8 did not address that the 200 W/m^2 insolation value does not consider the additional solar thermal input received by the fin's greater surface area.

Evaluations were also performed to determine the impact of fuel assemblies being asymmetric (not-centered) within the fuel cell and the basket not being centered within the canister when the package is transported in the horizontal orientation. A comparison of Table 3.3.14 (asymmetry boundary condition) and Table 3.3.1 (centered boundary condition) showed that the effective fuel planar thermal conductivities were lower when it was assumed the fuel assemblies were centered within the fuel cell (i.e., levitating). Likewise, Table 3.3.15 indicated higher temperatures when it was assumed the fuel basket was centered within the canister. For example, it was reported that the licensing basis (i.e., centered) PCT was 345°C whereas an asymmetric condition resulted in a PCT of 342°C. In addition, a 336°C PCT was reported for the asymmetric condition when including credit for modeling convection (laminar flow) of the helium (per Section 1.2.1.8 of the application) within the free volume of the cask cavity except the fuel assemblies (per Section 3.3.1.5 of the application).

In Appendix F of thermal calculation package HI-2125241, the applicant briefly mentioned a comparison of package surface temperatures for the case of a monolithic shield cylinder surface with surface enhancement and without surface enhancement; the FLUENT analysis used the k-ω model with transitional flow option enabled for low Reynolds number correction. The thermal energy from the decay heat ultimately is transferred from the monolithic shield cylinder, which is constructed of carbon steel having a relatively high thermal conductivity (Section 1.2.1.8 and drawing 8545, Rev. 7), to the ambient (air) via radiant and natural convection heat transfer. Section F.1 stated that the surface enhancement is not required for the package to meet transportation design objectives in the horizontal orientation, but rather the surface enhancement is present to aid heat transfer in the vertical storage orientation. Based on the results presented in Appendix F, it is noted that package surface temperatures could be reduced by a few degrees when the surface enhancement is considered in the thermal analysis.

Finally, Section 8.2.4 described an acceptance thermal test, to be performed ever five years, in which surface temperatures are measured and recorded at four equally spaced circumferential locations at the mid height of the active fuel after the package is loaded with spent nuclear fuel with the goal of verifying adequate heat dissipation to ensure interior temperature limits are not exceeded. Although the surface measurements may provide an indication that the package meets the regulatory limit for package surface temperature (10 CFR 71.43(g)), there was no discussion to indicate that the thermal test could be used to demonstrate the package's overall thermal performance capabilities (e.g., interior temperatures).

3.4 Evaluation Findings

Based on a review of the thermal chapter and discussion of the application, the staff concludes that the HI-STAR 180D thermal design, relative to the proposed thermal-related changes in Summary of Proposed Changes (HI-2178010-NRC), has been adequately described and evaluated and has reasonable assurance that the package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

The objective of the review is to verify that the Model No. HI-STAR 180D package containment design is adequately described and evaluated under normal conditions of transport (NCT) and hypothetical accident conditions (HAC), as required per 10 CFR Part 71.

4.1 Description of the Containment System

The staff verified the containment boundary for the HI-STAR 180D package has changed from the previous Certificate of Compliance (CoC) approval (ML14255A491); changes were made to the design of the closure lid region and seal options, specifically: 1) the revised inner closure lid design removes the lead shielding, and 2) seal part drawing numbers, seal / groove dimensions, seal seating load, and seal jacket material have all been changed as described in Appendix 8.A for the Technetics seal, "Option 2."

4.1.1 Inner Closure Lid Design Change

The applicant provided an evaluation of the change to the inner closure lid in Appendix H of Holtec Report No. HI-2125251 and concluded that the change did not have a significant effect on the stress intensity results and the decelerations predicted for the HAC 9-meter top end drop as presented in the applicant's LS-DYNA® structural analysis model submitted as part of the application; therefore, the staff finds the design change to the inner closure lid will not affect the performance of the containment boundary of the package and is acceptable as the structural model stress intensity results remain below the allowable stresses (see Section 2.2, "Changes to Inner Closure Lid Design," of this SER).

The leakage rate testing described in Section 8.1.4 of the application is to be performed on the revised inner closure lid design and is evaluated in Section 4.4.1, "Fabrication Leakage Rate Test," of this SER.

4.1.2 Technetics Seal, "Option 2"

In Appendix 8.A of the application, the applicant introduced information on the seal manufacturer, seal part/drawing number and described critical seal design parameters including seal and groove dimensions with tolerances, seal seating load with tolerances, surface finishes for sealing surfaces, and specific seal spring and material combinations for the Technetics seal, "Option 2". The Technetics seal, "Option 2," includes: the inner closure lid seals, the outer closure lid seals, the inner port cover seals, and the outer lid access port plug seal. The seals that are part of the containment boundary include: the inner closure lid inner seal, the outer closure lid inner seal, the vent and drain port cover inner seals, and the outer lid access port plug seal.

The staff reviewed the seal and groove dimensions presented in Appendix 8.A to verify the seals described would properly fit within the seal grooves as designed. The applicant provided updated calculations as part of the application, in Calculation 8 of Holtec Report No. HI-2125252, that the staff then verified used the revised seal seating loads for the inner port cover seal and outer lid access port plug seal. The applicant presented revised seal temperature limits in Table 3.2.12 of the application in a response to the staff's request for additional information (RAI) 4-1 (ADAMS Accession No. ML19337A805). Based on the staff's review of the Technetics Group and applicant's response to the staff's RAI 4-1, the staff finds the seal temperature limits to be acceptable. The staff concluded that there are no chemical, galvanic, or other reactions when using a silver jacketed seal material; therefore, based on the staff's review of the applicant's response to RAI 4-1, the staff finds the use of the silver jacketed seal material to be acceptable.

The applicant described, in Table 8.1.1 of the application, the seal acceptance criterion as leaktight, which is defined as a leakage rate of no greater than $1x10^{-7}$ reference cubic centimeter per second (ref-cm^3/s) of air, in accordance with American National Standards Institute (ANSI) N14.5, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment." The application also addressed in Section 2.2.1.1.6 of the application, the minimum useful springback of the seal to maintain the leaktight acceptance criterion, and the seal seating load for the inner and outer lid closure seals as critical seal parameters to provide leaktight containment.

Regarding the seal assembly springback parameter, the applicant described how this parameter provides for maintaining the leaktight acceptance criterion and protection from degradation, which the staff verified had not changed from the HI-STAR 180D safety analysis report Revision 3, (ADAMS Accession No. ML14203A276).

Similarly, the seal seating load for the Technetics seal, “Option 2,” inner and outer closure lid seals, remains bounded by the seating load for seal, “Option 1,” which was referenced in Section 4.3 of Holtec Report No. HI-2125251, and, therefore, the staff finds this acceptable.

The seal seating loads for the Technetics seal, “Option 2,” inner port cover seal and outer lid access port plug seal were revised, as described in Tables 8.A-3 and 8.A-4 of the application, and the staff verified the bounding value was used in Calculation 8 of Holtec Report No. HI-2125252, to demonstrate that the minimum total bolt preload in Table 7.1.1 of the application is adequate.

The staff verified these parameters are provided in Table 2.2.12 of the application and referenced in Appendix 8.A of the application. The staff also verified that Appendix 8.A is incorporated by reference on the licensing drawings. Therefore, based on the staff’s review of the seal design changes, the staff finds the seal design changes acceptable.

There are no changes to the American Seal & Engineering metallic seal, “Option 1,” critical parameters, for each of the containment boundary seals. These parameters have also been included in Appendix 8.A and have been previously approved. The two seal designs described in Appendix 8.A of the application are unique designs and other seal designs cannot be used, neither can seal field changes be made for the HI-STAR 180D package without prior approval from the NRC because containment boundary seals are important to safety components and modifying an important to safety component would result in an unanalyzed condition.

4.1.3 Containment Boundary and Containment System Closure

The containment boundary includes the: containment shell, containment baseplate, the containment closure flange, inner closure lid, inner closure lid inner metallic seal, outer closure lid, outer closure lid inner metallic seal, inner closure lid port covers (vent and drain), inner closure lid port covers (vent and drain) inner metallic seals, outer closure lid access port plug, outer closure lid access port plug seal, and associated welds and closure bolts.

Regarding the containment system and its components, the staff verified the following:

- All containment system components are shown in the licensing drawings.
- Containment system component information presented in the drawings is consistent with the information presented in the Structural and Thermal Evaluation sections of the application.
- Bolt torque patterns, lubrication requirements, and torque values are provided in Figure 7.1.1 and Table 7.1.1 of the application.
- Torque requirements in Table 7.1.1 of the application were referenced in fuel loading operations, cask closure operational procedures, and preparation for transport operational procedures described in Chapter 7 of the application.
- The torque values were calculated as a function of bolt preload, as noted in Table 7.1.1 of the application, and the applicant concluded in Calculation 8 of Holtec Report No. HI-2125252 the torque values are sufficient to maintain compression of the seals.

Therefore, the staff finds the torque values based on the revised bolt preload to be acceptable. The staff finds that the applicant has adequately demonstrated, in Section 4.1.4 of the application, that the containment system for the HI-STAR 180D package cannot be opened unintentionally or by an internal pressure within the package and therefore, the requirement in 10 CFR 71.43(c) is met.

4.2 Containment Under Normal Conditions of Transport (NCT)

Under NCT, the containment system of the package is designed to be leaktight as defined in ANSI N14.5-2014, i.e., there is no leakage greater than 1×10^{-7} ref-cm^3/s of air with a test sensitivity of 5×10^{-8} ref-cm^3/s of air, as described in Table 8.1.1 of the application. The staff verified the applicant provided a definition of leaktight in the glossary of the application and it was consistent with the definition in ANSI N14.5-2014.

The staff verified that the thermal and structural evaluations, presented in the application, demonstrate that there is no release of radioactive material under NCT. The applicant stated, in Section 3.3.2 and Table 3.1.2 of the application, that the maximum normal operating pressure (MNOP) of the HI-STAR 180D is 402.5 kPa with 3 percent rod rupture for the F-37 basket. This is lower than the design internal pressure for the cavity space of 552 kPa, which is provided in Table 2.1.1 of the application; therefore, the staff finds the MNOP to be acceptable.

In Table 3.1.2 of the application, the applicant demonstrated that the pressure in the inter-lid space is 155.8 kPa for the F-37 basket, which is also lower than the cask inter-lid space maximum operating pressure of 689.5 kPa, which is also provided in Table 2.1.1 of the application; therefore, the staff finds the cask inter-lid space pressure to be acceptable.

The applicant reported the maximum NCT temperatures for the containment shell, inner closure lid, outer closure lid, containment baseplate, and inner and outer lid seals in Table 3.1.1 of the application. The staff confirmed that the NCT containment boundary temperatures from the HI-STAR 180D SAR Revision 3 (ADAMS Accession No. ML14203A276) have not changed in the current application and do not exceed the temperature limits presented in Tables 3.2.10 and 3.2.12 of the application.

The applicant provided lid seal temperatures for a defense-in-depth, steady-state fuel reconfiguration scenario in Table 3.3.11 of the application. While the lid seal temperature exceeded the normal conditions temperature limit of 200°C (392°F) the lid seal temperature did not exceed the normal conditions temperature limit of 250°C (482°F), that indicates the lid seal will remain leaktight for between one year and 50 years. The applicant also described that the containment boundary will remain intact for the non-mechanistic fuel reconfiguration event analyzed in Section 3.3.5 of the application. Therefore, based on the staff’s review of the containment boundary temperatures and the associated containment boundary component temperature limits as presented in the application, the staff finds the containment boundary temperatures to be acceptable.

In Table 2.6.5 of the application, the applicant provided revised containment boundary stress intensities and safety factors for load combination N1, normal load condition, associated with NCT and demonstrated that all safety factors exceed 1 at key locations for each component of the containment boundary, as detailed in Section 2.6 of the application); therefore, the applicant concluded and the staff finds the containment boundary stress intensities and safety factors to be acceptable.

In Section 2.6.1.4.2 of the application, the applicant summarized that the containment boundary seals, which includes the closure lid seals and the vent and drain port cover seals, do not unload beyond the minimum force required to maintain leaktight conditions during NCT. The staff concludes that the results of the structural and thermal analyses, as well as the proposed leakage rate testing, conducted during fabrication to the ANSI N14.5 containment leaktight acceptance criterion and before every shipment and before every shipment to the ANSI N14.5 containment leaktight acceptance criterion, demonstrates compliance with 10 CFR 71.51(a)(1).

4.3 Containment Under Hypothetical Accident Conditions (HAC) of Transport

As described in Table 8.1.1 of the application, the containment system of the package is designed to be leaktight as defined in ANSI N14.5-2014 under HAC. The staff verified that the thermal and structural evaluations demonstrate no expected release of radioactive material under HAC.

The applicant reported, in Section 3.4.3.2 and Table 3.1.4 of the application, that the maximum cavity accident pressure, with assumed 100 percent fuel rod rupture, is 1154.6 kPa, which bounds the inter-lid pressure and is lower than the accident condition internal pressure (design pressure limit) of 1277.6 kPa, provided in Table 2.1.1 of the application; therefore, the staff finds the maximum cavity accident pressure to be acceptable.

Table 3.1.3 of the application lists the maximum HAC temperatures calculated by the applicant for the containment shell, inner closure lid, outer closure lid, containment baseplate, and inner and outer lid seals and the staff confirmed that the reported HAC temperatures have not changed from the temperatures reported in the HI-STAR 180D SAR Revision 3, (ADAMS Accession No. ML14203A276). The reported temperatures do not exceed the temperature limits presented in Tables 3.2.10 and 3.2.12 of the application; therefore, the staff finds the containment boundary temperatures for HAC acceptable.

In Section 2.7.4 of the application, the applicant summarized that the fire event, which occurs after the 9-meter drop accident and puncture event, does not lead to loss of seal integrity in either lid. Also, in Table 2.7.8 of the application, the applicant demonstrated that the inner and outer closure lid bolts average service stress during the fire event for the inner and outer closure lids remain below the allowable stresses, and therefore, the staff finds this acceptable.

In Section 2.7.8 of the application, the applicant concluded that both lids will maintain a positive contact load at their interface after each hypothetical accident event, which indicates that both the primary and secondary lid seals will remain functional to contain radioactive material. The applicant summarized, in the same section, that the sealing function is maintained at the end of each accident event and at the end of the HAC sequence. Further, the applicant stated that the inner closure lid port cover bolt torque requirement was sufficient to maintain closure under HAC.

The staff concludes that the results of the structural and thermal analyses, as well as the proposed leakage rate testing to the ANSI N14.5 containment leaktight acceptance criterion, demonstrates compliance with 10 CFR 71.51(a)(2).

4.4 Leakage Rate Tests for Type B Packages

4.4.1 Fabrication Leakage Rate Test

The purpose of the ANSI N14.5 fabrication leakage rate test is to demonstrate that the containment system, as fabricated, provides containment to the leaktight acceptance criterion.

In Table 8.1.2 of the application the applicant describes fabrication leakage rate tests performed on the containment shell, baseplate, closure flange, inner closure lid, outer closure lid, vent and drain port covers, and containment welds using a gas filled envelope leakage rate test (test A.5.3 from ANSI N14.5-2014), while the containment seals undergo an evacuated envelope leakage rate test (test A.5.4 from ANSI N14.5-2014).

Based on the information provided in the application, the staff verified the following:

- that the fabrication leakage rate test is performed on all containment boundary components,
- that the allowable leakage rate and test sensitivity is shown in Table 8.1.1 of the application,
- that all containment components are tested to the ANSI N14.5 leaktight acceptance criterion,
- that both the gas filled envelope and evacuated envelope leakage rate tests can achieve the test sensitivity specified in Table 8.1.1 of the application by confirming this in ANSI N14.5-2014,
- and finally, that multiplying the leakage rate acceptance criterion and leakage rate test sensitivity when using air as a tracer gas by a factor of 1.85 when using helium as a tracer gas, as specified in Table 8.1.1 of the application, is acceptable.

4.4.2 Pre-shipment Leakage Rate Test

The ANSI N14.5 pre-shipment leakage rate test was described in Sections 7.1.2.1, 7.1.2.2, 8.1.4, and 8.2.2 of the application and is performed before each shipment after the contents are loaded and the containment system is assembled. The staff verified that pre-shipment leakage rate tests are to be performed on all containment seals, as stated in Table 8.1.2 of the application.

All seals are tested to the ANSI N14.5 leaktight acceptance criterion because all seals are replaced prior to each shipment.

4.4.3 Periodic Leakage Rate Test

The purpose of the ANSI N14.5 periodic leakage rate test is to demonstrate that the containment capabilities of packagings built to an approved design have not deteriorated over an extended period of use. The staff verified that periodic leakage rate tests (an evacuated envelope leakage rate test per test A.5.4 from ANSI N14.5-2014) are performed on all containment seals, as stated in Table 8.1.2 of the application.

The staff verified the allowable leakage rates and test sensitivities are shown in Table 8.1.1 of the application and all seals are tested to the ANSI N14.5 leaktight acceptance criterion. The staff verified that the periodic leakage rate tests are described in Section 8.2.2 of the application to be valid for one year, and the staff verified that Section 7.1.3 of the application describes that the periodic leakage rate tests will be performed if more than 12 months have elapsed since the performance of the last periodic leakage rate tests.

4.4.4 Maintenance Leakage Rate Test

The purpose of the ANSI N14.5 maintenance leakage rate test is to confirm that any maintenance, repair, or replacement of components has not degraded the containment system and a test is performed prior to returning the package to service. The staff verified that maintenance leakage rate tests (a gas filled envelope leakage rate test per test A.5.3 from ANSI N14.5-2014) are performed on the containment shell, baseplate, closure flange, inner closure lid, outer closure lid, vent and drain port covers, containment welds, and containment seals, as stated in Table 8.1.2 of the application. The staff verified the allowable leakage rates and test sensitivities are shown in Table 8.1.1 of the application and all containment components are tested to the ANSI N14.5 leaktight acceptance criterion.

4.4.5 Leakage Testing Procedures and Leakage Rate Testing Personnel Qualifications

In Sections 8.1.4 and 8.2.2 of the application, the applicant described that the leakage rate tests shall be performed per written and approved procedures in accordance with ANSI N14.5-2014, and approved by an individual qualified and certified to American Society for Nondestructive Testing (ASNT) nondestructive testing (NDT) Level III in leak testing, the staff finds this to be acceptable based on its review of ANSI N14.5-2014.

The applicant described, in Sections 8.1.4 and 8.2.2 of the application, that leakage rate testing shall be performed by personnel who are qualified and certified in accordance with ASNT Recommended Practice No. SNT-TC-1A, “Personnel Qualification and Certification in Nondestructive Testing,” the 2006 edition as cited in reference 8.1.2 of the application, the staff finds this acceptable based on its review of ANSI N14.5-2014.

4.4.6 Leakage Testing Conclusions

The staff verified that Condition 6(a) of the CoC states, “The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.” The staff also verified that Condition 6(b) of the CoC states, “The package shall meet the acceptance tests and be maintained in accordance with Chapter 8 of the application.” These two conditions of the CoC are necessary to ensure that all portions of Chapters 7 and 8 of the application are complied with. The staff ensured that the language in Chapters 7 and 8 of the application was consistent with these two conditions of the CoC; therefore, any changes to Chapters 7 and 8 of the application do necessitate NRC’s approval.

The staff concludes the fabrication, pre-shipment, periodic, and maintenance leakage rate tests verify the integrity of the containment boundary, and that the containment components will maintain their leaktight containment function during transportation operations. The staff concludes the leakage rate tests are consistent with the guidelines of ANSI N14.5-2014.

4.5 Evaluation Findings

4.5.1 Description of Containment System

The staff has reviewed the description and evaluation of the containment system and concludes that:

(i) the application identifies established codes and standards for the containment system;

(ii) the package includes a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package during transport;

(iii) the package is made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction.

4.5.2 Containment under Normal Conditions of Transport

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

4.5.3 Containment under Hypothetical Accident Conditions

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

In summary, the staff has reviewed the Containment Evaluation section of the SAR and concludes that the package has been described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71, and that the package meets the containment criteria of ANSI N14.5-2014.

5.0 SHIELDING EVALUATION

5.1 Review Objective

The objective of this review is to verify that the shielding design of the Model No. HI-STAR 180D package, with the proposed changes specified in this amendment request, continues to provide adequate protection against direct radiation from its contents and that the package design meets the external radiation limits of 10 CFR Part 71 under NCT and HAC.

The package is designed as an exclusive use package in accordance with the regulatory requirements of Part 71 of Title 10 of Code of Federal Regulations (10 CFR), Subparagraph 71.47(b). The staff's evaluation of the proposed changes and their impacts on radiation shielding of the package is documented in following sections of this SER.

5.2 Description of the Shielding Design

5.2.1 Design Features

The applicant submitted an application for revision of the Certificate of Compliance, along with a revised Safety Analysis Report and supporting documents for the proposed changes. Based on Holtec's report, HI-2125255, "Shielding Analyses for the HI-STAR 180D," the proposed changes related to shielding design of the package include:

1. Updated lid design.
2. Updated thickness of materials of the cask baseplate.
3. Updated Holtite composition and density based on the latest data.
4. Updated modeling approach for Holtite and lead.
5. Considering lead slump for design basis accident conditions.

6. Updated neutron axial profile and scaling factor, to be more realistic.
7. Including Fuel Reconfiguration Scenario 3, with a source strength reduction by a factor of 2.
8. Including a determination of dose rate from all alternative fuel loading patterns.

The staff reviewed the application related to the shielding design of the package with the proposed changes. The following section of this Safety Evaluation Report (SER) documents the staff's review and conclusions on these proposed changes to the package. The staff's review focuses on the technical aspects of the revised shielding designs and the proposed changes in contents that are important to safety. The staff performed its review following the guidance provided in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

The HI-STAR 180D packaging system consists of a fuel basket and an overpack qualified to transport commercial spent nuclear fuel by rail, road, and seagoing vessel. The overpack is a cylindrical metal cask protected with impact limiters. The fuel basket is a stainless steel cylinder partitioned by Metamic plates into cells that hold spent fuel. The overpack includes shielding radially and axially, at the top and bottom areas of the overpack, by its stainless steel components, lead, and a Holtite neutron shield.

The packaging system employs two closure lids at the top of the cask: (1) a containment closure flange assembly and (2) an inner fuel basket closure lid. The top containment lid includes a built-in neutron shielding (Holtite) disc. The inner closure lid has an embedded supplemental neutron shielding (Holtite) disc and a lead disc embedded in a stainless steel shell. At the bottom of the cask, the bottom shield cylinder assembly features built-in supplemental neutron shielding (Holtite) and gamma shielding (lead) both layered around the containment shell and over the containment baseplate. One of the proposed changes is to replace the lead disc in the inner lid with stainless steel. With the proposed design change, the gamma shield of the inner lid becomes one solid disk made up of stainless steel.

5.2.2 Summary of Maximum Radiation Levels

The applicant provided a summary of the maximum dose rates in Tables 5.1.1 to 5.1.8 of the application for the package under NCT and HAC. The maximum dose rate on the surface of the package loaded with the F-32 Basket is 140.2 mrem/hr under NCT. The maximum dose rate on the surface of the package loaded with the F-37 Basket is 137.2 mrem/hr under NCT. These calculated dose rates demonstrate that the package shielding design meets the regulatory requirements of 10 CFR 71.47(b) and 71.51(a)(1) which is 200 mrem/hr under NCT.

The maximum dose rate at 2 meters from the package with the F-32 Basket is 8.05 mrem/hr under NCT. The maximum dose rate at 2 meters from the package with the F-37 Basket is 8.01 mrem/hr under NCT. These calculated dose rates demonstrated that the package shielding meets the regulatory requirements of 10 CFR 71.47(b)(3) which is 10 mrem/hr under NCT.

The maximum dose rate in the vehicle cab for the package with the F-32 Basket is 1.93 mrem/hr under NCT. The maximum dose rate in the vehicle cab for the package with the F-37 Basket is 1.92 mrem/hr under NCT. These calculated dose rates demonstrated that the package shielding design meets the regulatory requirement of 10 CFR 71.47(b)(4) which is 2 mrem/hr under NCT.

For the package under HAC, the maximum dose rate at 1 meter from the surface of the damaged package loaded with the F-32 Basket is 406.7 mrem/hr, and at 1 meter from the package loaded with the F-37 Basket is 425.1 mrem/hr. These calculated dose rates demonstrated that the package shielding design meets the regulatory requirements of 10 CFR 71.51(a)(2) which is 1000 mrem/hr under HAC.

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

5.3 Source Specifications

This application does not propose any changes in the contents to be transported in the package with respect to fuel designs, including fuel types and configurations, e.g., PWR 15x15, 17x17. However, the applicant requested to use a new burnup profile which may potentially affect the source distribution and should be considered a change to the source terms.

The applicant presented the updated profile in the table shown in page C-2 of Appendix C of the shielding calculation package for this application, Holtec report, HI-2125255, “Shielding Analyses for the HI-STAR 180D.” The burnup profile is broken into 52 axial segments and modeled by assigning a source probability to each of the 52 axial sections of the active region, based on a representative axial burnup profile. The representative burnup profile is derived from the actual nuclear power plant reload records as described in Holtec Report, 2125157, “Data Definition Document for HI-STAR 180D Cask.” For fuel gamma radiation, the probability is proportional to the burnup, since the gamma source strength changes linearly with burnup. For neutrons, the probability is proportional to the burnup raised to the power of 4.2, since the applicant found that this correlation was appropriate.

Although the staff notes that this burnup profile is derived from a specific nuclear power plant outside the United States, the staff finds that the burnup profile developed in this report is very close to the bounding burnup profile presented in Table 2 of NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses.”

Additionally, as presented in the HI-2073681, “Safety Analysis Report for HI-STAR 180,” this approximation has been justified by comparing the probability distributions used in the shielding calculations with a probability distribution directly derived from the burnup distribution and the corresponding depletion calculations.

5.4 Shielding Model

5.4.1 Methods

Section 1.3 of the application provides the drawings that describe the HI-STAR 180D Packaging. These drawings were used to create the MCNP models used in the radiation transport calculations. The drawing package also illustrates the HI-STAR 180D on a typical transport vehicle with a personnel barrier installed. Figures 5.3.1 and 5.3.2 of the application show the cross-sectional views of the HI-STAR 180D cask loaded with F-37 and F-32 baskets respectively, as they were represented in the MCNP models.

For this amendment, one of the proposed changes is to include consideration of a potential loss of material in the Holtite-B neutron shield. Under normal conditions of transport, the applicant states that the minimum Holtite mass with maximum weight loss was considered. The hot-condition circumferential gap corresponding to the nominal Holtite pocket dimensions were explicitly modeled. The axial gaps were not modeled but were considered in the Holtite-B density calculations. The other Holtite-B components were modeled with no gaps, but with reduced densities. The composition of the Holtite-B used in the models is presented in Appendix A of Holtec Report No: HI-2125255. The staff reviewed this assumption and finds that this approach for simplifying gaps in the models is acceptable because reducing the density of the shielding material can, in general, compensate for small gaps.

Each basket allows several loading patterns as defined in Chapter 7 of the application. In addition, the applicant provided a list of alternative burnup, cooling time, and enrichment combinations for each basket region, as shown in Table 5 and Table 6 of Appendix A to Holtec Report No: 2125255.

The applicant evaluated the dose rates for the package loaded with the alternative burnup, cooling time, and enrichment combinations and demonstrated that the package resulted in a dose rate similar to the one from the reference loading. On this basis, the staff finds that fuel with burnup, enrichment, and cooling time combinations, as presented in Table 5 and Table 6, is also acceptable for transport in the HI-STAR 180D package.

The applicant also analyzed fuel reconfiguration for packages containing high burnup fuel. The applicant presented its fuel reconfiguration analyses in Appendix F of Holtec Report, HI-2125255, “Shielding Analyses for the HI-STAR 180D.” The applicant states that it performed three scenarios of fuel reconfigurations:

1. In Scenario 1, the active regions of all fuel assemblies are modeled as collapsed to their half height, with a corresponding increase in density (two times the nominal density value). The collapse could lead to a situation where axial sections of higher and lower burnups are collapsed into each other. In this scenario, a flat axial burnup profile was utilized for the fuel. This would maximize dose rates in the axial direction at the bottom of the cask (Dose Location 5, see Figure D-2).

2. Scenario 2 uses the same physical model as Scenario 1, but a compressed axial profile is used instead of a flat profile. This maintains the source term peak at the center of the fuel height, and therefore maximizes the dose rate in radial direction.

3. Scenario 3 evaluates the potential effect of large areas with a reduced fuel amount by retaining the full fuel height, while reducing the fuel density and the source strength by a factor of 2.

The applicant presented the results of the fuel reconfiguration study in Table F-2 of the report. The results show that the maximum dose rate is 760.14 mrem/hr at 1 meter from the package. This result demonstrated that, in the event of fuel reconfigurations, the dose rates would remain below the regulatory limits of 1000 mrem/hr at 1 meter from the package. On this basis, the staff finds that the package meets the regulatory requirement of 10 CFR 71.51(a)(2).

In Section 5.1.1 of the application, the applicant states that the main neutron shielding is provided by the Holtite-B neutron shield embedded in the cask body and inner lid. One of the proposed changes in this application is to replace the lead embedded in the inner closure lid with stainless steel. With this change, the inner closure becomes one solid piece of stainless steel with the shielding at the top end of the package.

The applicant provided a revised drawing for the new design of the top inner lid, as shown in Sheet 5 of licensing drawing 8584, Revision 8. Consequently, the shielding capacity is reduced because stainless steel has a gamma attenuation coefficient much lower than lead. However, the staff finds that the applicant’s shielding analyses demonstrated that the dose rate in the axial direction remains in compliance with regulatory requirements of 10 CFR 71.47 and 71.51, even with the reduced shielding capacity of the revised design of the inner closure lid.

The applicant also considered the potential loss of material and evaluated the shielding performance of the package with the following assumptions:

1. minimize the Holtite density by assuming the minimum amount of copper and boron carbide (B4C) in Holtite-B;

2. maximizing Nylon-66 weight loss to reduce the density of Holtite-B by a factor of "1-weight loss," the composition of Holtite is also updated to consider the weight loss to the Nylon-66 portion of Holtite-B;

3. minimize Holtite-B mass in the modeled MSC Holtite-B cavity baseplate and lid.

In Section 3.4 of the application, the applicant demonstrated that all materials used in the HI-STAR 180D remain at or below their design temperatures during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition, as a result of temperature changes.

The staff reviewed the material composition of the Holtite-B neutron shield materials described in the HI-2167314 Report and found them acceptable since the improved Holtite-B composition provides better shielding capabilities for high and low burnup fuel contents.

For the package under HAC, the applicant states that the base plate, which is made of lead, is replaced with void in the shielding model. Also, the lead in the bottom lead shield is reduced in the radial direction by 6.35 cm, and in the axial direction by 2 mm to account for slumping. The discussion of the lead slump modeling is presented in Section 5.3 of the application.

The discussion of the sensitivity study is provided in Subsection 5.4.7.4 of the application, "Sensitivity Study of the Accident Condition Model – Lead Slump Analysis." In this sensitivity study, the applicant modeled the HI-STAR 180D package loaded with a F-32 basket under HAC. The MCNP model for this sensitivity analysis includes the following assumptions: 1) to model the lead slump in the base plate (Bottom Lead Gamma Shield), it is assumed that a circular segment with the height of 12.7 cm is replaced with a void, and 2) Holtite is also replaced with void.

In the design basis calculations, to model the slump of the lead in the base plate (Bottom Lead Gamma Shield), 6.35 cm of lead is replaced with a void. To show that the design basis model of the hypothetical accident conditions is conservative, the applicant presented a sensitivity study for the lead slump with a different geometry.

The results show that the design basis lead slump analyses bound the other potential lead slump scenarios. On this basis, the staff finds that the applicant's design basis shielding analyses for a package under HAC are conservative and acceptable.

5.4.2 Material Properties

The applicant used material properties identical to those in the previous revisions. The composition of the Holtite-B used in the modeling is presented in Appendix A of Holtec Report No: 2125255. Also, in this Amendment, a lower density of lead is used to model the presence of gaps and lead/Holtite pockets. The density of the lead is reduced to an amount that bounds the effect of engineered gaps. A lead density of 11.0 g/cc is assumed. The lead composition and density are shown in Table A-2 of the Holtec Report No: 2125255.

The staff reviewed the properties of the materials used for shielding components and finds that they are consistent with specifications from the ASTM standard and Holtec specification for the neutron shield material Holtite-B. The use of a reduced lead density to account for the gap between the lead shield and the steel is based on a calculation of the actual volume of the lead shield space and the actual thickness of the lead layer.

The staff reviewed the material composition of the Holtite-B neutron shield materials described in the HI-2167314 Report and found them acceptable since the improved Holtite-B composition provides excellent shielding capabilities for high and low burnup fuel contents. Also, the Holtite positions within the monolithic cylinders are offset to minimize any streaming through the side of the cask.

5.4.3 Flux-to-Dose-Rate Conversion

The applicant used MCNP to calculate dose rates at the various desired locations. MCNP calculates neutron or photon fluxes and these values can be converted into dose using dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file. The response functions used in these calculations are listed in Table 5.4.1 (b) of the SAR and were taken from ANSI/ANS 6.1.1-1977.

5.5 Evaluation

The staff reviewed the methods the applicant used to calculate the source terms and dose rates. In Section J-2 of Appendix J to the Holtec Report No: 2125255, Rev.4, the applicant described the process that was used to determine alternative fuel specifications and bounding loading patterns.

Regarding the source terms, the applicant states that a shielding analysis of the HI-STAR 180D Package demonstrated that dose rates meet the limits of 10CFR 71.47 while considering the combined effect of the nonlinear relationship between burnup and source terms. The approximation related to the new axial burnup profile is discussed in SAR Section 5.4.1.

The staff reviewed the new burnup profile and the applicant's justifications and finds that the new burnup profile is equivalent to the profile that is directly derived from the fuel burnup records from reactor operation data. It is a renormalized form of the burnup profile derived from reactor records. On this basis, the staff finds this burnup profile to be acceptable.

Also, the staff concludes that in this approach, the burnup profile is dependent on fuel specific properties and therefore the approximation of probability proportional to the burnup raised to the power of 4.2 is valid for both normal condition of transport and accident conditions.

However, it is imperative to point out that the burnup profile introduced here may not be applicable to spent fuel from all US reactors because this burnup profile is derived from the operation records of reactors from outside the United States, based on the review of the data presented in Holtec Report, 2125157. Reviewers for the future applications may need to assess the applicability of this burnup profile to any new spent fuel contents.

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

In terms of fuel reconfiguration for packages containing high burnup fuel analyzed by the applicant, the staff reviewed this approach and finds that the applicant has adequately analyzed the hypothetical fuel reconfigurations under HAC. The staff considers fuel reconfigurations Scenarios 1 and 2 more plausible, if there was fuel reconfiguration.

The staff does not believe that Scenario 3 is plausible because it is physically impossible to have a reconfiguration that would cause the fuel to reduce its density and strength by a factor of 2. This assumption is not conservative because it assumes a reduced source terms by a factor of 2 with no physical basis. Nevertheless, the first two scenarios would bound most of the plausible fuel reconfigurations. On this basis, the staff determined the applicant's fuel reconfiguration analyses to be acceptable and the results demonstrate that the dose rate meets the regulatory limit as prescribed in 10 CFR 71.51(b)(2).

The staff reviewed the sensitive study and found that the applicant's HAC model acceptable since the radiation steaming through the lead slump circular segment does not result in an increase of the maximum dose rates for the design basis fuel in transport during an accident condition.

In addition, the applicant developed an approach to determine the dose rate from alternative fuel loading patterns. Dose rates from all alternative patterns for all tally locations, where high dose rates are expected, at the surface and 2 m from the package, show that the regulatory dose rate limits are met. The applicant provided a detailed explanation for this method in Appendix J of Holtec report, HI-2125255, "Shielding Analyses for the HI-STAR 180D."

The staff also found the proposed new burnup profile to be acceptable because it properly represents the gamma source and neutron source distributions as a function of fuel burnup. The normalized probability is calculated by multiplying the relative burnup by the node height and dividing by the active fuel height. The relative neutron source is equal to the relative burnup raised to the power of 4.2 for UO_2 fuel.

The applicant used the 1977 edition of the ANSI/ANS 6.1.1 standard for flux-to-dose rate conversion factors in its shielding analyses. This is consistent with the recommendation of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel." On this basis, the staff finds that the applicant's flux-to-dose rate conversion factors are adequate and acceptable.

5.6 Conclusion

Based on its review of the statements and representations provided in the application, the staff has reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and NRC guidance. On this basis, the staff finds that the revised HI-STAR 180D package design meets dose rate limits of 10 CFR Part 71.

Based on the statements and representations in the application, as supplemented, the staff concludes that the package meets the requirements of 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

The applicant revised the application to include an additional evaluation scenario of potential fuel reconfiguration under HAC. The Model No. HI-STAR 180D package is designed to prohibit water in-leakage and, therefore, is analyzed to exclude moderator impacts on reactivity under HAC, per the recommendations of Interim Staff Guidance 19 (SFST-ISG-19), "Moderator Exclusion under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel under the Requirements of 10 CFR 71.55(e)."

The applicant previously demonstrated subcriticality of several potential fuel reconfiguration scenarios for the Model No. HI-STAR 180D package with water in-leakage, as defense-in-depth to support approval of the package design.

For the added scenario, the applicant evaluated an increased pitch in the lower 25% of the fuel assemblies in both the F-32 and F-37 baskets. The F-32 basket evaluation considered fresh UO_2 fuel enriched to 4.55 weight percent, while the F-37 basket evaluation considered spent UO_2 fuel assemblies in Configuration 1 (see Table 7.D.6 of the application), with an initial enrichment of 4.55 weight percent and a burnup of 15 Gigawatt days per metric ton uranium (GWd/MTU). The results of these analyses are summarized in Tables 6.3.20 and 6.3.21 for the F-32 and F-37 baskets, respectively, and demonstrate that the package remains subcritical under the conditions considered in the additional reconfiguration scenario.

The staff reviewed the applicant's analyses. The staff determined that, while the additional scenarios present conditions that are more reactive than the scenarios considered in the previously approved package application, the package remains subcritical under the additional conditions considered in the revised criticality analysis.

The staff based this determination upon a review of the changes to the configuration of the fissile contents for F-32 and F-37 baskets and their associated results. The applicant is not requesting any change to the allowable contents in the certificate of compliance associated with this new analysis. Therefore, the staff finds with reasonable assurance that the package continues to meet the criticality safety requirements of 10 CFR Part 71.

7.0 OPERATING PROCEDURES

The staff verified that bolt torque patterns, lubrication requirements, and torque values are provided in Figure 7.1.1 and Table 7.1.1 of the application, and that torque requirements in Table 7.1.1 of the application were referenced in fuel loading operations, cask closure operational procedures, and the preparation for transport operational procedures described in Chapter 7 of the application.

The torque values were calculated as a function of bolt preload, as noted in Table 7.1.1 of the application, and the staff verified that the torque values are sufficient to maintain compression of the seals.

The ANSI N14.5 pre-shipment leakage rate test, described in Sections 7.1.2.1, 7.1.2.2 of the application, is performed before each shipment after the contents are loaded and the containment system is assembled. The staff verified that pre-shipment leakage rate tests are to be performed on all containment seals. All seals are tested to the ANSI N14.5 leaktight acceptance criterion and all seals are replaced prior to each shipment.

Explicit operational steps to the cask loading/closure procedures for the removal of standing water from closure lid bolts holes have been provided.

8.0 ACCEPTANCE TESTS AND MAINTENANCE

Section 8.2.4 describes an acceptance thermal test, to be performed ever five years, in which surface temperatures are measured and recorded at four equally spaced circumferential locations at the mid height of the active fuel after the package is loaded with spent nuclear fuel with the goal of verifying adequate heat dissipation to ensure interior temperature limits are not exceeded. Although the surface measurements may provide indication that the package meets the regulatory limit for package surface temperature (10 CFR 71.43(g)), the staff noted there was no discussion to indicate that the thermal test could be used to demonstrate the package's overall thermal performance capabilities (e.g., interior temperatures). Thus, the staff disagrees with the applicant's statements.

The applicant has added the option to repair cask bolt holes by installing threaded inserts in place of the damaged bolt holes. Per Holtec Drawing No. 8545, Note 58 requires that the repaired configuration, with the installed threaded insert, be evaluated and be required to meet applicable stress limits prior to repair.

The inspection, evaluation, and repair of the cask closure fasteners is discussed in Section 8.2.3.4 of the application, which requires the repaired joint to meet the ASME Code stress limits required of the initial joint and the material and manufacturing process testing required of the repair to be in accordance with ASME B&PV Code Section III, Division 1, Subsection NB.

The applicant also confirmed that, per Section 8.1.5.3 of the application, each manufactured lot of Holtite-B neutron shield material is tested to verify that the boron carbide content, hydrogen density and Holtite-B material density meet the requirements in Table 8.1.11 of the application.

The applicant proposed an alternative to the requirement of ASME B&PV Code, Section III, Division 1, Subsection NB, paragraph 2330, by following the guidance in Regulatory Guide (RG) 7.11 and RG 7.12 which define a maximum Tndt. The applicant stated that the Tndt defined in the application (Table 8.1.9) for the steel material is lower than that defined in the RG. Therefore, the staff considers the alternative to be conservative and acceptable. The staff notes that the alternative is also stated in the updated discussion for Metamic-HT crack propagation in Section 2.1.2.2 of the application.

The applicant proposed to not conduct post-weld heat treatment in specific welds that are less than the containment boundary, and justified the alternative by stating that these welds are exempt from post-weld heat treatment per ASME B&PV Code, Section III, Subsection NB, Table NB-4622.7(b)-1. The staff reviewed Table NB-4622.7(b)-1 to confirm that the nominal thickness of the nickel-alloy and low-alloy steel components identified in Table 2.1.14 allows for exemption to post-weld heat treatment. Therefore, the staff considers the alternative to be acceptable.

The applicant proposed to not conduct radiographic examination of the pressure-retention boundary welds after post-weld heat treatment. The applicant stated that all welds (including repairs) will have passed radiographic examination prior to post-weld heat treatment of the entire containment boundary. The applicant justified the exemption by stating that confirmatory radiographic examination after post-weld heat treatment is not necessary because this process is not known to introduce new weld defects in nickel steels. The staff considers the justification to be reasonable and, therefore, to be acceptable.

In Table 8.1.2 of the application, the applicant describes fabrication leakage rate tests performed on the containment shell, baseplate, closure flange, inner closure lid, outer closure lid, vent and drain port covers, and containment welds using a gas filled envelope leakage rate test (test A.5.3 from ANSI N14.5-2014), while the containment seals undergo an evacuated envelope leakage rate test (test A.5.4 from ANSI N14.5-2014).

Based on the information provided in the application, the staff verified that the fabrication leakage rate test is performed on all containment boundary components, that the allowable leakage rate and test sensitivity is shown in Table 8.1.1 of the application, that all containment components are tested to the ANSI N14.5 leaktight acceptance criterion, that both the gas filled envelope and evacuated envelope leakage rate tests can achieve the test sensitivity specified in Table 8.1.1 of the application by confirming this in ANSI N14.5-2014, and finally, that multiplying the leakage rate acceptance criterion and leakage rate test sensitivity when using air as a tracer gas by a factor of 1.85 when using helium as a tracer gas, as specified in Table 8.1.1 of the application, is acceptable.

In Sections 8.1.4 and 8.2.2 of the application, the applicant described that the leakage rate tests shall be performed per written and approved procedures, in accordance with ANSI N14.5-2014, and approved by an individual qualified and certified to American Society for Nondestructive Testing (ASNT) nondestructive testing (NDT) Level III in leak testing. ASNT Recommended Practice No. SNT-TC-1A, “Personnel Qualification and Certification in Nondestructive Testing,” 2006 edition, is cited in reference 8.1.2 of the application: the staff finds this acceptable based on its review of ANSI N14.5-2014.

The staff verified that pre-shipment and periodic leakage rate tests are to be performed on all containment seals. All seals are tested to the ANSI N14.5 leaktight acceptance criterion because all seals are replaced prior to each shipment. The staff verified that the periodic leakage rate tests are described in Section 8.2.2 of the application to be valid for one year, and the staff verified that Section 7.1.3 of the application describes that the periodic leakage rate tests will be performed if more than 12 months have elapsed since the performance of the last periodic leakage rate tests.

The staff verified that maintenance leakage rate tests are performed on the containment shell, baseplate, closure flange, inner closure lid, outer closure lid, vent and drain port covers, containment welds, and containment seals, as stated in Table 8.1.2 of the application. The staff verified the allowable leakage rates and test sensitivities are shown in Table 8.1.1 of the application and all containment components are tested to the ANSI N14.5 leaktight acceptance criterion.

CONDITIONS

The following changes were made to the CoC:

Item No. 3(a) identifies the new address of the applicant while Item No. 3(b) identifies the revised application, as supplemented.

Condition No. 5(a)(3) has been modified to include the new revisions of the licensing drawings.

Condition No. 6(b) has been modified to remove the requirements for the bend test qualification of a representative friction stir weld sample since such requirements are now included in the revised Metamic HT Sourcebook.

Condition No. 10 has been modified, after renewal of the CoC, to extend the CoC expiration date by 5 years.

The References section of the certificate was updated to reference Revision No. 5 of the application, dated April 16, 2020.

CONCLUSION

Based on the statements and representations in the application, the staff finds that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with CoC No. 9367, Revision No. 2.