Response to NRC Request for Additional Information
Holtec International
Docket No. 71-9374
HI-STAR 80 Transportation Package

Chapter 1 – General Information

NRC RAI 1-1

1-1 Clarify the tolerances specified in the licensing drawings of the Model No. HI-STAR 80 package.

The staff found that there were not any dimensional tolerances on any components within the HI-STAR 80 cask drawings, with the exception of the basket wall thicknesses in Drawings 9796 and 9797. The staff requests that the applicant provide tolerances for the cask and its components within the other drawings, such as Drawings 9800, 9795, 9798 and 9801, as well as the other basket dimensions in Drawings 9796 and 9797(such as the steel shims). Tolerances in material thickness and size can appreciably alter the performance of the package with respect to the drop tests cited in 10 CFR 71.71 and 71.73, as well as shielding. ISG-20 states that *"The reviewer should ensure that reasonable tolerances for dimensions and weights are specified, because packaging features may be subject to some variability in fabrication."* NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," provides guidance for preparing drawings of transportation packages submitted in an application for approval under 10 CFR Part 71. It states that engineering drawings should have tolerances that are consistent with the package evaluation.

While it is clear that tolerances allow for variability in fabrication, it is currently unclear what those tolerances are for this package. NUREG CR-5502, page 2, states: *"All dimensions indicated on drawings should include tolerances that are consistent with the package evaluation. Tolerances may be addressed by a drawing note that defines a general tolerance applicable to many features. If a design feature needs a more (or less) restrictive tolerance than indicated by the note, the appropriate tolerance should be specified explicitly in the dimensioning of that feature."*
The staff also notes that this is a recurring topic which has been repeatedly addressed in the HI-STAR 180 and 180 D applications, as well as in both the request for supplemental information and two rounds of RAIs for the Model No. HI-STAR 190 package.

Therefore, the applicant is requested to specify nominal tolerances on the engineering drawings. Calculations related to package performance shall incorporate tolerances indicated on the engineering drawings as necessary to show compliance with 10 CFR Part 71.

This information is required to demonstrate compliance with 10 CFR 71.33(a).

Holtec Response to RAI 1-1:

The drawings and the shielding analysis have been updated to include the tolerances for the components that affect the dose rates. The shielding analysis has been updated to utilize minimum thicknesses, as explained in the Chapter 5 responses. In most cases the shielding analysis governed the tolerances, however in some cases structural, thermal or criticality requirements drove the dimensional limits.

The component tolerances that have been added to licensing drawings are intended to support the safety analysis. Additional manufacturing controls are needed to ensure that dimensional stack-ups for each cask maintain package bounding dimensions and package bounding weights. Dimensions that remain without explicit tolerances may vary greatly without affecting safety margins.

Tolerances needed solely for interfaces and operations will be included on fabrication drawings. Weights are listed in Chapter 2, and remain unchanged.

The approach is intended to follow the same approach presented in the second round RAIs for the HI-STAR 190 package, Docket No. 71-9373.

In the basket drawings 9796, 9797 and 9798 tolerances have been added for the aluminum, steel and stainless steel components that are included in the shielding analysis. Tolerances on Metamic HT components remain unchanged.

Cask drawing 9800 has been updated to include tolerances on all components included in the shielding analysis. Each shell layer, cask bottom layer, and lid layer includes explicit tolerances. However, the Holtite neutron shielding material is an exception. Dimensions and tolerances on the drawing reflect the pocket size, rather than the Holtite dimensions. Holtite will be installed to meet mass requirements within the specified pockets, as explained in the shielding RAI responses.

Impact limiter drawing 9801 has been updated to include tolerances for structural components and structural components credited with dose reduction.

No changes have been made to the package assembly drawing 9795.

Chapter 2 – Structural and Materials Evaluation

NRC RAI 2-1

2-1 Clarify, in Chapter 7, how the top and bottom lifting trunnions will be rendered inoperable as tie-down devices for transportation or provide an analysis to verify that they meet the requirements of 10 CFR 71.45(b)(1).

Section 2.5.2 of the application states that there are no tie-down devices that are a structural part of the package for transport in the U.S., but that the bottom lifting trunnions may be utilized for transport outside the U.S. Figures 7.A.1 and 7.A.2 of the application depict the general arrangement of the HI-STAR 80 package on a transport vehicle with impact limiter and tie-downs attached, but do not distinguish between

transport configurations. Figure 7.A.1 indicates that the transport frame features saddles and bottom trunnion tie-downs. Figures 1.3.1 and 1.3.2 of the application are the same illustrations as Figures 7.A.1 and 7.A.2, but include notes that are not included in the figures in Chapter 7.

Note 1 of Figure 1.3.1 states that, in that illustration, the trunnions have been rendered inoperable by a custom tie-down device. Section 7.1.3 of the application makes no mention of rendering either set of trunnions (top or bottom) inoperable for tie-down, whether as part of a custom tie-down system or not. Because Chapter 7 is part of the Certificate of Compliance, the staff requires clarification, in Chapter 7, as to how the top and bottom trunnions will be rendered inoperable as tie-down devices for transport within the U.S., or an analysis, in Chapter 2, that shows that they meet the tie-down requirements of 10 CFR 71.45(b)(1)

This information is required to determine compliance with 10 CFR 71.45(b)(2).

Holtec Response to RAI 2-1:

The HI-STAR 190 cask trunnions will be rendered inoperable by attaching trunnion covers, by use of custom tie-down device, or other ancillary device that blocks access to the trunnions. Figure 7.A.1 has been revised to show trunnion covers over the top trunnions. Figure 1.3.2, which corresponds to Figure 7.A.1 contains notes that describe the general regulatory requirement. Note 1 has been added to Figure 7.A.1 to clearly state that the arrangement is only for transport outside the U.S. Similarly, Note 1 has been added to Figure 7.A.2 to clearly state that the arrangement is for both transport inside and outside the U.S. Other minor clarifications have been made to Figures 1.3.1, 1.3.2, 7.A.1 and 7.A.2. A step has been inserted in Subsection 7.1.3 to require trunnion covers, custom tie-down device or other blocking ancillary device be used to render accessible top and/or bottom trunnions inoperable (i.e. not accessible for lifting and handling) with reference to requirements in Figures 7.A.1 and 7.A.2.

NRC RAI 2-2

2-2 Justify the calculated radial lead slump value of 1.6 inches in diameter in the lower forging gamma shield due to the side drop (slap down) analysis.

Table 2.7.4 of the application reports a maximum radial lead slump of 1.6 inches in diameter for the lower forging lead shield as a result of the side drop (top down), slap down test. This value is echoed in Table 5.3.7 in which a radial lead slump of 0.8 inches in radius is reported for the lower forging.

The staff reviewed the LS-DYNA model output file (HS80ST2\d3plot) and measured a change in diameter for the lower forging lead shield of 3.55 inches as a result of the side drop (top down), slap-down case. As a second check, the staff measured a distance of 3.53 inches between the edge of the bottom forging lead shield and the bottom forging outer edge in the LS-DYNA model following the slap down scenario. This is much larger than the reported value of 1.6 inches.

This information is required to determine compliance with 10 CFR 71.51(a)(2)

Holtec Response to RAI 2-2:

Holtec has confirmed that the bounding radial lead slump in the lower forging occurs in the slapdown (top down) scenario with a maximum diameter change of 3.55 inches; the previously reported value was taken from the other slapdown (bottom down) scenario. Therefore, Tables 2.74 and 5.3.7 of the SAR have been updated accordingly. In addition, the bounding radial lead slump value has been corrected in Holtec report HI-2167023R1.

NRC RAI 2-3

2-3 Provide a discussion of the thermolysis and radiolysis data of Holtite-B as neutron shielding material.

Under normal conditions, the applicant claims that the depletion of the B-10 in the Metamic and Holtite neutron shield is negligible, less than 10^{-6} over 50 years. However, the applicant did not show testing data, calculation results, or relevant references to arrive at these conclusions for Holtite-B. Additionally, the applicant did not assess the micro-structural integrity of Holtite-B, as a result of thermolysis or radiolysis.

Given that the Model No. HI-STAR 80 is a new package, the applicant needs to confirm the applicability of previously NRC approved similar cases for the detailed information described above.

This information is required to determine compliance with 10 CFR 71.43(d) and 71.55(d).

Holtec Response to RAI 2-3:

<u>B-10 Depletion</u>

During normal conditions B-10 is subject to depletion as a result of neutron capture reaction under irradiation from the loaded fuel. Neutron depletion calculations under bounding neutron flux and 50-year exposure supports the 10^{-6} depletion factor cited in the SAR. The calculations are provided below:

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

<u>Thermolysis and Radiolysis Testing</u>

Holtite-B is qualified through an array of tests conducted at Holtec's Corporate Research Laboratory in Orvillon, PA to confirm it's stability characteristics under an overarching thermal and radiation environment. The test observations and evaluations are recorded in QA validated test reports cited in Reference [2-3] herein. The tests include Holtite-B testing in furnaces maintained under elevated temperatures that meet or exceed HI-STAR 80 operating temperatures, subjecting samples to Co-60 gamma radiation and neutron irradiation in test reactors. The test conditions are evaluated in the following table:

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

The above observations support the conclusion that Holtite-B is unaffected under exposure to design basis thermal and radiation environment in the HI-STAR 80 cask.

References:

[2-3] "Holtite-B Sourcebook", HI-2167314, Rev. 2.

Chapter 3 - Thermal Evaluation

NRC RAI 3-1

3-1 Provide specific and detailed information on the thermal properties of materials, e.g., cladding materials, basket materials and gasket materials, at higher temperatures, e.g., up to 800 ˚C.

Temperature data under HAC conditions are provided in Table 3.1.3 and includes maximum temperatures under fire, i.e., cladding materials, 502 ˚C, and fuel basket materials, 473 ˚C. The cladding temperature remains below the HAC limit of cladding temperature, 570 ˚C. The applicant also states that the containment gasket can be relaxed (softened) from fire, see (Section 2.1.2.1). Given that the Model No. HI-STAR 80 is a new package, detailed information on the thermal properties used in the analyses under HAC at higher temperatures, e.g., up to 800 ˚C, needs to be provided. The applicant also needs to update Tables 1.2.4, 3.2.2, 3.2.3, 3.2.8, and 3.2.9 with thermal properties up to 800 ˚C.

This information is required to determine compliance with 10 CFR 71.43(d) and 71.55(d).

Holtec Response to RAI 3-1:

The HI-STAR 80 SAR thermal property tables in Chapter 1 and Section 3.2 are suitably expanded in SAR Revision 2.B to address thermal analysis requirements as justified below.

Material	Properties	Justification
Metamic-HT	Provided in Table 1.2.4 up to 500°C	Bounds maximum computed 473°C fire accident basket temperature (Table 3.1.3)
Fuel	Table 3.2.3 expanded up to 600°C	Bounds maximum computed 502°C fire accident fuel temperature (Table 3.1.3)
Helium, Air	Table 3.2.2 expanded up to 800°C, Tables 3.2.8, 3.2.9 expanded up to 815°C	Bounds 800°C fire temperature
Structural Materials	Provided in Table 3.2.4 up to 815°C	Bounds 800°C fire temperature
Lead	Provided in Table 3.2.2 up to 700°C	Bounds maximum computed 608°C fire accident lead temperature

Material	Properties	Justification
Aluminum 6061-T6	Room temperature conductivity provided in Table 3.2.2	Elevated temperature conductivity of Aluminum alloys increase with temperature
Aluminum 2219 T8511	Room temperature conductivity provided in Table 3.2.2	Elevated temperature conductivity of Aluminum alloys increase with temperature
Holtite-B	Provided in Table 3.2.2	Note 1
Impact limiter crush material	Provided in Table 3.2.2	Note 2
Seals	Provided in Note 3	Note 3

Note 1: Reasonable lowerbound conductivity adopted under normal transport. Under fire accident conditions Holtite and its enclosing Aluminum structure assumed to be lost and fire flux conservatively applied to the intermediate shell. During post-fire accident air conductivity assumed to minimize heat dissipation.

Note 2: Under fire accident conditions parallel-to-layers direction honeycomb material conductivity is not affected by fire. However, normal-to-layers direction conductivity is severely degraded because the resin bonding the corrugated Aluminum layers is destroyed. To maximize heat input during fire the parallel-to-layers direction conductivity in Table 3.2.2 of the SAR is adopted and normal-to-layers direction is assumed to be unaffected and during post-fire cooldown theoretical lowerbound conductivity of air assumed to minimize post-fire cooldown heat dissipation.

Note 3: Seals are principally required to ensure containment functions of the HI-STAR 80 cask. As evaluated in Chapter 3 (Tables 3.1.1 and 3.1.3) the maximum seal temperatures reached under normal and accident conditions remain below Table 3.2.12 limits.

NRC RAI 3-2

3-2 Provide the values of the package heat load, thermal inertia of the loaded package, and rate of temperature rise (dT/dt) for both the F-12P and F-32 B baskets in Table 3.3.6 of the application to verify the maximum permissible time for completion of wet transfer operations.

The applicant stated in Section 3.3.4 that water inside the Model No. HI-STAR 80 package cavity is not permitted to boil during fuel loading operations, in accordance with NUREG-1536. The applicant used equations provided in Section 3.3.4 and performed an adiabatic heat up calculation to determine a bounding heat-up rate based on the package heat load and thermal inertia of the loaded package, and then obtained the maximum permissible time for completion of wet transfer operations, as shown in Table 3.3.6 of the application.

This information is required to determine compliance with 10 CFR 71.35 and 71.71.

Holtec Response to RAI 3-2:

The thermal inertia of the loaded package and the bounding rate of temperature rise for both F-12P and F-32B baskets have been added to Table 3.3.6 of SAR Rev 2.B. In order to avoid repetition of information, Table 7.D.1 of the SAR is referenced for the package heat load.

NRC RAI 3-3

3-3 Provide the heat load limit of non-fuel waste (NFW) loaded in the HI-STAR 80 package and specify a heat load limit required as a criterion to backfill the package cavity with helium or just air.

The applicant stated in Section 3.1.1 that it is not necessary to backfill the package cavity with helium to improve its thermal performance because of the significantly lower decay heat for a package loaded with non-fuel waste (NFW).

Given that the NFW loaded in HI-STAR 80 package can be low to high level waste, the applicant needs to provide the heat load limit of NFW allowable in the application or explain whether a heat load limit is required as a criterion to backfill an NFW-loaded package with helium or just air.

This information is required to determine compliance with 10 CFR 71.33 and 71.71.

Holtec Response to RAI 3-3:

The heat load limit of non-fuel waste (NFW) is 2kW and is provided in Table 7.D.7 of the SAR. The heat load limit of NFW is only about 3.7% of the design basis maximum heat load for BWR fuel (i.e. 54kW). Therefore it is not necessary to backfill the cask cavity with helium to improve the thermal performance.

NRC RAI 3-4

3-4 Clarify if the information provided in references for the fluorocarbon compound V1289-75 is applicable or appropriate to verify the performance of fluorocarbon compound V1285-75 used in the HI-STAR 80 package.

In its RSI response, the applicant stated that: (1) Note-5 of Table 3.2.12 is applied to denote seals that must withstand a temperature of 250°C (482°F) for at least 70 hours in the short term operational and accident conditions; and (2) Parker fluorocarbon compound V1285-75 has been identified as a suitable material for these seals. These two documents from the manufacturer are provided in the RSI response to serve as references for compliance of the Parker fluorocarbon compound V1289-75.

V1289-75 Parker Compound Data Sheet recommends a temperature range of (-50°F to 400°F) for the use of V1289-75, which is below 482°F as indicated in Note 5. The applicant should explain whether the use of the Parker fluorocarbon compound is appropriate for the HI-STAR 80 package.

It is also not evident that information for fluorocarbon compound V1289-75 is appropriate to serve as a reference for fluorocarbon compound V1285-75, particularly with respect to dry heat resistance. The applicant should clarify that information provided in the fluorocarbon compound V1289-75 references is applicable or appropriate for the fluorocarbon compound V1285-75.

This information is required to determine compliance with 10 CFR 71.71 and 71.73.

Holtec Response to RAI 3-4:

The reference to compound V1285-75 was a typographical error. We apologize for the confusion that may have caused. Parker Compound V1289-75 is the material selected and evaluated for seals designated by (Note-5) in SAR table 3.2.12.

The "recommended temperature range" in the V1289-75 Parker Compound Data Sheet provides guidance for the continuous service temperature of the seal. The "recommended temperature range" does coincide with the Normal Condition Temperature Limits in SAR table 3.2.12.

The Short Term Operations & Accident Temperature Limits specified in SAR table 3.2.12 bounds the worst case temperature rise occurring for the specified component throughout the duration of the postulated accident conditions for the transportation package. For components designated by Note 5, 482°F for 70 hours bounds the worst case analyzed transient temperature rise. Compound Data Sheet for compound V1289-75 includes results from the ASTM D573 Heat Age Test, which subjects the seal to 70 hours at 482°F. The results of the Heat Age Test indicate the seal is expected to remain fully functional throughout and after this transient temperature rise event.

NRC RAI 3-5

3-5 Clarify if the information provided in document "Parker Seal Test Report – FF400 Compression Set" is applicable to verify the performance of FF400 O-ring seal used in the Model No. HI-STAR 80.

Note-6 in Table 3.1.2 is applied to denote seals that must withstand a temperature of 320°C (608°F) for at least one hour followed by another temperature of 200°C (392°F) for at least 70 hours, and Parker perfluoro-elastomer compound FF400-80 has been identified as a suitable material for these seals.

However, the document "Parker Seal Test Report – FF400 Compression Set" provided by the applicant in response to staff's RSI indicates that: (1) a control specimen FF400 O-ring seal was tested under the common condition of 70 hours at 200°C (392°F), and (2) two specimens of FF400 O-ring seals were tested in separate ovens at 300°C and 320°C (608°F) for the first hour of the test duration in order to replicate the fire emergency conditions.

It appears the test described in the document "Parker Seal Test Report – FF400 Compression Set" is not identical to the test required by Note-6 in Table 3.1.2 (a FF400 O-ring seal is tested under two consecutive temperature conditions of 320°C for at

least one hour first, and then followed by 200°C for at least 70 hours. The applicant should clarify that the test conditions in the document "Parker Seal Test Report – FF400 Compression Set" is appropriate to verify the performance of FF-400 O-ring seal in the Model No. HI-STAR 80 package.

This information is required to determine compliance with 10 CFR 71.71 and 71.73.

Holtec Response to RAI 3-5:

The information provided in previously submitted document "Parker Seal Test Report – FF400 Compression Set" is applicable to verify the performance of FF400 O-ring seals used in HI-STAR 80.

The bounding Short Term Operations and Accident temperature limit for SAR table 3.2.12 components designated with Note-6 is 608°F (320°C) for one hour, followed by 70 hours at 392°F (200°C).

The "Parker Seal Test Report – FF400 Compression Set" includes five different test configurations, as summarized in Table 1 of the test report. Test configuration (3) matches the requirements of Note-6 in SAR Table 3.2.12: exposure to 608°F (320°C) temperatures for one hour, followed by exposure to 392°F (200°C) for 70 hours. The report concludes that the seal material maintained sealing ability during all test configurations, indicating that this seal material is suitable for use in HI-STAR 80.

NRC RAI 3-6

3-6 Justify whether a backfill pressure selected from a broad range of pressures (2.9 ~ 29 psia) is appropriate for shipment of non-fuel waste.

Compared to the pressure range of 25 ~ 29 psia required for the package cavity (Table 7.1.4) for backfill pressure requirements in shipment of spent nuclear fuel, there is a broad range of backfill pressure (2.9 ~ 29 psia) specified for the package cavity in shipment of non-fuel waste (Table 7.1.5). The applicant should provide the criteria which justifies whether a backfill pressure selected from a broad range of pressures (2.9~29 psia) is appropriate for shipment of NWF.

This information is required to determine compliance with 10 CFR 71.71.

Holtec Response to RAI 3-6:

Unlike SNF, NFW does not rely on elevated backfill pressure to enhance heat transfer due to its very low heat load. Therefore, a lower minimum backfill pressure requirement for NFW may be specified without consequence. With the flexibility provided in Table 7.1.5, the user may choose to backfill a cask containing NFW at sub-atmospheric backfill pressure to enhance containment of the package as defense-in-depth although it is not required by the safety analysis. On the other hand, the maximum backfill pressure requirement for spent nuclear fuel is adopted for NFW for practical purposes and ensures the cask cavity pressure for cask with NFW will be bounded by the evaluation for the cask with spent nuclear fuel evaluated in Chapter 3 of the

SAR. For these reasons a broad range of backfill pressure is specified for NFW in Table 7.1.5 of the SAR. Therefore no change is proposed to Table 7.1.5.

NRC RAI 3-7

3-7 Clarify whether the package accessible surface temperature measurement is required for the package loaded with moderate burnup fuel (MBF) and non-fuel waste (NFW) in preparation for transport.

The applicant stated, in Section 7.1.3 "Preparation for Transport", that the surface temperatures of the accessible areas of the package are measured if required. For packages containing HBF, the surface temperature measurements shall include the surface temperature measurements required by the post-shipment fuel integrity acceptance test specified in Chapter 8. The applicant should clarify whether the surface temperature survey is required for the package loaded with moderate burnup fuel (MBF) and non-fuel waste (NFW).

This information is required to determine compliance with 10 CFR 71.71.

Holtec Response to RAI 3-7:

We agree with the staff's position on the need to clarify the application. The temperature measurement of package accessible surfaces applies to all packages (all allowable contents). The surface temperature measurement required by post-shipment fuel integrity acceptance test specified in Chapter 8 only applies to packages containing HBF. In order to add clarify to the application, the temperature measurements (step 10) have been separated into to two separate steps in Subsection 7.1.3, namely steps 10 and 11. Furthermore, the temperature measurement that is applicable to all packages is only necessary to confirm that temperatures are within 10CFR71.43, if the user chooses not to use the personnel barrier, consistent with the step (now step 12). Step 12 indicates that the personnel barrier is optional as long as regulatory requirements are confirmed to be met.

NRC RAI 3-8

3-8 Perform the transient thermal analyses and provide the temperature history (as a function time) for the package loaded with HBF and MBF, respectively, to ensure the acceptability of the PCT acceptance criteria.

In response to staff's RSI, the applicant lowered the PCT acceptance criteria from 390°C (734°F) to 380ºC (716ºF) during drying to ensure adequate time is available for operations. Based on operational experience, the applicant stated that the acceptance criterion for the temperature limit of HBF is lowered to 380ºC (716ºF), which leaves sufficient time to perform operations like backfilling the cask cavity with helium. Additionally, a similar requirement for the MBF temperature limited to 550ºC (1022ºF) is added to the same section in the application.

The NRC staff still does not have reasonable assurance that adequate time will be available at temperatures of 380°C (716°F) for HBF and 550°C (1022°F) for MBF to

perform operations such as helium backfill. The applicant should perform transient thermal analyses and provide the temperature history (as a function time) for a package loaded with HBF and MBF, respectively. The temperature history will establish the appropriate time frames for PCTs rising from 380°C (716°F) to 400°C (752°F) for HBF and from 550°C (1022°F) to 570°C (1058°F) for MBF.

This information is required to determine compliance with 10 CFR 71.71.

Holtec Response to RAI 3-8:

To perform transient evaluation of vacuum drying condition, the HI-STAR 80 thermal model presented in Subsection 3.3.1 of the SAR has been modified to exclude the impact limiters and outer closure lid. Additionally, the cask is in vertical orientation during drying operations. These transient simulations of vacuum drying operations are performed and documented in Holtec report HI-2156468 Revision 2. For the reviewer's reference, the temperature rise with time is also provided in Figure 3-8.1 and 3-8.2 below. The results of these evaluations suggest that it takes at least 55 minutes for the PCT to rise from 380°C (716°F) to 400°C (752°F) for both PWR and BWR baskets containing high burnup fuel under their respective design basis maximum heat loads. The available time is calculated based on conservative thermal models which assume that the cold gap between the basket, basket shims and containment shell remains unchanged i.e. no credit for differential thermal expansion between components is included in the model. This modeling approach ensures available time for vacuum drying operations are conservative. It is noted that the PCT will be lowered by the cold helium once the backfill operation is started. In addition, the thermal conductivity of low pressure helium is much higher than that of low pressure water vapor. The heat dissipation inside the cask cavity is enhanced during the backfill operation. Thus, the PCT will remain below its limit of 400°C (752°F) if the helium backfill operation is started within 30 minutes after the vacuum drying operation time limit is reached.

During vacuum drying operation, the rate of increase in PCT is expected to either remain the same or decrease gradually with temperatures rise. This is also confirmed by transient simulations of vacuum drying operations documented in Holtec report HI-2156468 Revision 2. For the reviewer's reference, the temperature rise with time is also provided in Figure 3-8.3 and 3-8.4 below. The time duration for the PCT to rise from 550°C (1022°F) to 570°C (1058°F) is more than 2 hours for both PWR and BWR baskets containing only moderate burnup fuel under their respective design basis maximum heat loads. Thus, for baskets containing only moderate burnup fuel under design basis maximum heat loads, the PCT will remain below 570°C (1058°F) if the helium backfill operation is started within 2 hours after the vacuum drying operation time limit is reached.

Changing process operations from vacuum drying to helium backfill can be accomplished within the specified time limit using typical plant equipment, procedures, and quality assurance oversite. Before or during the vacuum drying cycle, helium backfill equipment and helium supply tanks are staged in the cask processing area and pre-connected to the closed cask orifice port. The vacuum drying operation is time limited depending on the cask heat load. Therefore, once vacuum drying begins, the latest possible end time is known, which will be several hours later. Technicians, supervisors and inspectors can be appropriately scheduled and deployed to the work area before the vacuum drying time limit is reached. Once the countdown for process change begins, the orifice port used for vacuum drying is closed in order to isolate the vacuum pump from the cask interior. Then the orifice port connected to the helium supply is opened, which initiates helium flow into the cask. This ends the time limited

operation. Users may implement variations on this approach to suit site specific conditions. Engineering evaluations may be used to justify an increased time limit for the operation changeover based on reduced fuel heat load or favorable site specific conditions.

Figures 3-8.1 through 3-8.4

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

Chapter 4 – Containment Evaluation

NRC RAI 4-1

4-1 Demonstrate the package containing the moderate burnup fuel (MBF) is leakage rate tested through a single barrier or redundant barriers, in accordance with ANSI N14.5.

The applicant noted in Table 8.1.2 that the pre-shipment and periodic leakage rate testing shall be performed on either the inner or the outer containment boundary closure seals (single barrier) for package containing MBF, but mentioned in Section 8.2.2 “Leakage Tests” that the maintenance leakage rate testing shall be performed on the redundant barriers along the leakage path through containment penetrations and lid closures for spent nuclear fuel (SNF).

Given that SNF also includes MBF, the applicant needs to explain why the package containing MBF is tested through a single barrier for the pre-shipment and periodic leakage rate testing, but may be tested through redundant barriers for the maintenance leakage rate testing.

This information is required to determine compliance with 10 CFR 71.51.

Holtec Response to RAI 4-1:

For MBF, the pre-shipment and periodic leakage rate testing shall be performed on either the inner or the outer containment boundary closure seals (single barrier, non-moderator exclusion function) as stated in Note 2 of Table 8.1.2. Maintenance leakage rate testing shall be performed on any containment boundary component or portion of the containment system affected by the maintenance, repair or component replacement as stated in the third paragraph of Subsection 8.2.2. The following statement in Subsection 8.1.2 has been deleted: “For packages containing SNF, leakage rate testing shall be performed on the redundant barriers along the leakage path through containment penetrations and lid closures. For packages containing NFW, leakage rate testing shall be performed on either the inner or outer barrier along the leakage path”.

NRC RAI 4-2

4-2 Demonstrate that the value of 4.41×10^6 cm^2 used for the calculation of the allowable leakage rates is greater than the surface area of the thirteen typical PWR fuel assemblies stored in the HI-STAR 80 package.

The applicant stated in Section 4.5.1 and Table 4.5.4 that a total surface area of 4.41×10^6 cm^2 of the contaminated solids stored in the HI-STAR 80 package is greater than the surface area of thirteen (13) typical PWR fuel assemblies, and therefore it is conservative to use the value of 4.41×10^6 cm^2 in calculation of the allowable leakage rates.

The applicant needs to show this derivation and demonstrate that the value of 4.41×10^6 cm^2 is greater than the surface area of the 13 typical PWR fuel assemblies.

This information is required to determine compliance with 10 CFR 71.35 and 71.51.

Holtec Response to RAI 4-2:

Contaminated solids transported in HI-STAR 80 are various components of the nuclear reactor or inserts to spent fuel assemblies such as: fuel channels, control rods, cut pieces of the reactor core grid or core spray. Holtec has reviewed surface areas of typical components and waste loadings expected to be transported in the HI-STAR 80 package. The amount of material transported is constrained by the weight limit and space available to store the material. Examples of a few possible loadings are shown below. The total surface area used in the calculations was set approximately double of the largest surface area expected to be transported in the cask.

Surface area estimated for typical components transported in the package:

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

The above review of the components shows that the surface area of the expected transported waste is lower than 441 m^2 (4.41E+06 cm^2), i.e. the activity concentration used in the SAR is bounding.

NRC RAI 4-3

4-3 Clarify the features of the pressure testing, as described below, for the HI-STAR 80 package.

The applicant stated in Section 8.1.3.2 that pressure testing may be performed independently for the inner and outer closures using a single temporary test seal on each closure as applicable. The applicant's statement seems to indicate that the pressure testing can be also performed by treating the inner and outer closures together as a group and using a single temporary test seal on each closure.

The applicant needs to clarify whether the above statement is true as one alternative for the pressure testing. If true, the applicant needs to provide the criteria, conditions, and requirements to (a) perform the pressure testing on the inner and outer closures

independently, and (b) perform the pressure testing on the inner and outer closures as a group.

This information is required to determine compliance with 10 CFR 71.35 and 71.51.

Holtec Response to RAI 4-3:

We agree with the staff's position on the need to clarify the application. The cask cavity, including the inter-lid space and space between port closures, must be pressure tested as stated in Paragraph 8.1.3.2 (first paragraph). Thus regardless of the configurations implemented to test all the spaces, all the spaces must be tested. The pressure test also applies to all containment boundary closures. In order to clarify the application, the following statement has been added to the first paragraph of Paragraph 8.1.3.2: "Furthermore, all containment boundary closures must be pressure tested." Finally to complete the clarification, the second paragraph of Paragraph 8.1.3.2 has been revised to read: "Pressure testing may be performed in various cask closure configurations as needed so that each containment boundary closure is pressure tested at least once. Containment boundary closures may be tested with single temporary test seal."

NRC RAI 4-4

4-4 Explain why ANSI N14.5 test types A.5.1 and A.5.2 do apply to the NFW packages for the pre-shipment leakage rate test unless the alternative pre-shipment leakage rate acceptance criterion In Note 3 of Table 8.1.1 applies.

According to Table 8.1.2, ANSI N14.5 test types A.5.1 Gas pressure Drop ((pressure decay, nominal test sensitivity = 10^{-1} to 10^{-5} ref·cm^3/s) and A.5.2 Gas Pressure Rise (vacuum retention, nominal test sensitivity = 10^{-1} to 10^{-5} ref·cm^3/s) do not apply to the NFW packages for the pre-shipment leakage rate test unless the alternative pre-shipment leakage rate acceptance criterion In Note 3 of Table 8.1.1 applies.

The applicant needs to explain why ANSI N14.5 test types A.5.1 and A.5.2 do apply to the NFW packages for the pre-shipment leakage rate test unless the alternative pre-shipment leakage rate acceptance criterion In Note 3 of SAR Table 8.1.1 applies.

This information is required to determine compliance with 10 CFR 71.43(f) and 71.51.

Holtec Response to RAI 4-4:

As stated in Holtec Response to RSI 4-5, ANSI N14.5 test types A.5.1 and A.5.2 <u>do not</u> apply to the NFW packages for the pre-shipment leakage rate test unless the alternative pre-shipment leakage rate acceptance criterion In Note 3 of Table 8.1.1 applies. Therefore, was as reflected in Revision 2.A of the SAR, ANSI N14.5 test types A.5.1 and A.5.2 were deleted from Table 8.1.2 and moved to Note 3 of Table 8.1.2 as example options when the alternative pre-shipment leakage rate acceptance criterion is applied by the user.

ANSI N14.5 test types A.5.1 and A.5.2 <u>do</u> apply to <u>all</u> packages (all allowable contents) for the alternative pre-shipment leakage rate test as currently stated in Note 3 of Table 8.1.2. No restriction of package type is made in Note 3. No change is proposed for revision 2.B of the SAR.

Chapter 5 – Shielding Evaluation

NRC RAI 5-1

5-1 Justify the approach for determining allowable fuel assemblies within the HI-STAR 80.

In discussing allowable fuel assemblies, Appendix 7.D states: *"… the fuel assembly may differ from any other dimension and specification, including specifications in the notes, listed in Table 7.D.2 for PWR fuel and Table 7.D.3 for BWR fuel."* It appears as though as long as the array size, e.g. 15x15, 10x10, etc., is the same that the allowable fuel assemblies can vary by any parameter.

The staff does not have enough information to determine that this approach is conservative with respect to meeting external dose rate limits for both NCT and HAC. For example, if the fuel mass is greater than assumed in the analysis, it will produce a higher source term. The applicant needs to provide additional information clarifying the approach and justify it ensures that all potential assemblies still meet regulatory dose rate limits.

This information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-1:

The goal of this proposed approach focuses on covering changes that the fuel vendors make to their design from time to time, with a main focus on criticality, where even small changes can a have a noticeable effect. A new Subsection 5.4.10 has been added to provide additional discussion and justification for this approach with respect to dose rates. Also, the specification of the corresponding requirements in Appendix 7.D is reorganized to make it clear that differences are only permitted for entries in certain tables.

NRC RAI 5-2

5-2 Discuss the effects on source spectra for assemblies irradiated with burnable poison rod assemblies present.

From the allowable loading tables in Appendix 7.D of the application, it does not appear that there are any restrictions or considerations for fuel burned with burnable poison rod assembly inserts. Based on NUREG/CR-6701, burnable poison rods insertion will affect the spectra and strength of source terms because of the production of extra actinides.

The applicant needs to discuss these effects on the source terms of spent fuel assemblies that have history of burnable poison rods for PWR fuel or control rods blade for BWR fuel during their depletion.

The information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-2:

The effect of reactivity control devices during reactor operation has been included in the revised evaluations of the source term uncertainty (see response to RAI 5-20), with results documented in the new Appendix 5.F of the revised Chapter 5. As a bounding approach, the exposure to such devices for the entire irradiation period was assumed and modeled. For PWR, a CRA was selected for this evaluation, which bounds any burnable poison rod assemblies since it has a larger overall effect on reactivity. Also, as stated above, it is assumed to be inserted for the entire irradiation period, while burnable poison rods are typically only present during the first cycle. For BWR fuel, a control blade insertion is assumed, also for the entire irradiation. The evaluations show that the effect of those devices on the dose rates around the cask is negligible.

As a side note, burnable poison rod assemblies in PWR fuel assemblies are currently not qualified for the HI-STAR 80. The only PWR non-fuel hardware qualified are control rod assemblies operated in the fully withdrawn position with respect to the active fuel zone during full power operation. This restriction has been added to the CRAs in Appendix 7.D.

NRC RAI 5-3

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

NRC RAI 5-4

5-4 Revise, as necessary, the tables within Chapter 5 of the application to clearly show the accumulated contributions to the external dose rates from all factors that increase dose rates within the application.

Within the application the applicant performs evaluations of the effects on external dose rates due to a different configuration, such as high burnup fuel, inclusion of stainless steel rods, and consideration of manufacturing tolerances. The applicant evaluated the effect of each of these factors individually and shows that, with each configuration, the HI-STAR 80 package still meets regulatory dose rate limits.

However it is not clear to the staff if the accumulated changes are being applied to all possible fuel configurations (burnup, enrichment cooling time, etc.), a limited subset, or the same fuel configurations that are being reported as the limiting configurations in Tables 5.1.1 through 5.1.8 of the application. It is not clear that these are the limiting configurations. It is also not clear if the applicant considered the increase in dose rates due to the combination of these effects (i.e. high burnup fuel, with steel rods at minimum tolerances).

The staff requests that the applicant discuss and justify that the configurations used to evaluate these studies are the limiting conditions. The applicant also needs to demonstrate that the combined combination of all effects still meet regulatory dose rate limits. For example, Table 5.1.7 of the application states that the maximum calculated hypothetical accident condition dose rate is 9.8 mSv/hr, and Tables 5.1.3 and 5.1.4 state that the maximum calculated normal conditions of transport dose rate at 2 meters from

the vehicle is 0.092 mSv/hr. It is not clear that once the increase in dose rates discussed above is included that the HI-STAR 80 still meets regulatory dose rate limits.

The information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-4:

The dose rates in Chapter 5 are updated to account for the accumulated contributions to the external dose rates from all factors that increase dose rates within the application, namely source term input uncertainties, irradiated stainless steel rods, manufacturing tolerances on the materials, and axial burnup and void distributions, for both NCT and HAC.

NRC RAI 5-5

5-5 Clarify if non-fuel hardware is allowed and provide associated shielding analyses as necessary.

Note 5 to Table 7.D.1 (F-12P) and Note 4 to Table 7.D.1 (F-32B) states: *"When complying with the maximum decay heat units in any basket cell location, decay heat from both the fuel assembly and any non-fuel hardware must be accounted, as applicable for the particular basket cell, to ensure the decay heat emitted by all contents in a basket cell does not exceed the limit."* However, the definition for allowable content in Appendix 7.D does not specify transport of non-fuel hardware within assemblies, nor is this considered within the external dose rate evaluations of the package.

The applicant needs to clarify what is included as "non-fuel hardware" and provide the necessary shielding analyses as well as allowable burnup and cooling times to demonstrate that inclusion of non-fuel hardware does not cause external dose rate under both NCT and HAC to exceed regulatory limits.

The information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-5:

The only non-fuel hardware qualified for transportation in the PWR assemblies are control rod assemblies, and for BWR assemblies are the channels (which provide an insignificant radiation source and decay heat load) (see Table 7.D.1). The revised NCT and HAC dose rates for the PWR assemblies now include the dose contribution for control rod assemblies based on the assumptions provided in Section 5.2.3 of the SAR. Also, Appendix 7.D has been updated to add the decay heat for the CRAs as calculated in Section 5.2 of the SAR.

NRC RAI 5-6

5-6 Revise Appendix 7.D to include decay heat from control rods or discuss how a user specifically includes this contribution to decay heat in determining the allowable contents.

The staff notes that the applicant calculated decay heat associated with activated control rod cladding and presented this information in Table 5.1.15 of the application (See RAI 5-5). However this table is not referenced in Appendix 7.D of the application. The applicant needs to discuss how the user calculates decay heat for non-fuel hardware and control rods in determining whether thermal limits are met.

The information is required to determine compliance with 10 CFR 71.43(g).

Holtec Response to RAI 5-6:

Appendix 7.D has been revised to include the decay heat of the control rod assembly transported within PWR assemblies. The control rod assembly decay heat is calculated based on the assumptions stated in Section 5.2.3 of the SAR. These assumptions will be added as restrictions for the CRAs in Appendix 7.D.

NRC RAI 5-7

5-7 State if solid steel rods were included in the modeling of the fuel assembly 17x17 and update the analysis if necessary.

Table 7.D.2 of the application states for fuel assembly 17x17S1 that it can have 25 guide tubes but Note 2 says that it *"may include solid steel rods instead of guide tubes."* The applicant needs to discuss how it determined that regulatory dose rate limits for both NCT and HAC are met considering the additional Co-60 source from the steel rods.

The information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-7:

Section 5.4.6.2 has been added to the SAR to address the dose rates from 25 irradiated stainless steel rods. The dose rates are based on two assemblies out of the four assemblies in the inner four basket cells. The dose rates are based on the same methodology as use to calculate the dose rates with four irradiated stainless steel rods (see Section 5.4.6 and response to RAI 5-3 for details), with the exception that only two assemblies may contain up to 25 irradiated stainless steel rods per basket and these two assemblies must be within the inner four basket cells. This restriction has been added to Appendix 7.D. The dose rates with two assemblies containing 25 irradiated stainless steel rods are within the dose limits as listed in Tables 5.4.29 through 5.4.31.

NRC RAI 5-8

5-8 Clarify the assumptions used in determining control rod activation.

Pertaining to the analysis for control rod activation, provide the following information:

a) Clarify the assumption on control rod activation with respect to insertion. Page 5.2-3 in Section 5.2.3 of the application states that: *"At full power, the control rods are assumed to operate in a fully withdrawn position with respect to the active fuel region."* It is not clear to the staff if this is an analysis assumption. Because assuming control rods fully withdrawn is not a conservative assumption, if this is the assumption used, the applicant needs to justify this assumption or include loading restrictions in Note 1 of Table 7.D.1 of the application to reflect that control rods inserted by any amount for any duration are not allowed for shipment.

b) The control rod source described in Table 5.2.15 of the application only shows the source and decay heat in terms of Co-60 from the stainless steel cladding. Note 1 of Table 7.D.1 of the application does not restrict control rod material; therefore, the applicant also needs to address non Ag-In-Cd control rods, such as the source from Hafnium rods, that could potentially be shipped, or restrict the allowable control rod materials in Note 1 of Table 7.D.1 of the application accordingly.

The information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-8:

Note 1 of Table 7.D.1 has been updated to include all assumption used in determining the control rod assembly source terms and decay heats. These assumptions include operation in the fully withdrawn position during full power operations as well as cladding material (stainless steel) and absorber material (AgInCd).

NRC RAI 5-9

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

NRC RAI 5-10

5-10 Clarify the contents of the Non-Fuel Waste Basket (NFWB).

Table 7.D.7 of the application lists the allowable contents for the Core Component Cassette for the NFWB. The staff finds the description of allowable contents ambiguous and requests that the applicant add specific details of what, specifically, can be transported in this basket. Currently the description states: *"Core components such as fuel channels, transition pieces, spacer grids, control rods, neutron monitors, and*

burnable absorbers and meeting the following specifications." Use of the language "such as" makes this description ambiguous and can lead to misinterpretation of what is allowed.

The staff requests that the applicant be specific on what components can be shipped and/or provide a loading procedure that informs package users of the allowable properties of contents including material, density, geometry, source and source distribution. For items such as "neutron monitors," the applicant should be specific and discuss what type of monitor/detector; for example: fission chamber, BF_3 tubes, or He-3 detector. The applicant should discuss whether these detectors are intact or damaged and whether there is any potential release of gaseous fission products or helium.

The information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-10:

The specification of the allowable NFW in (now in Table 7.D.8 and Table 7.D.9) has been revised. Clarifications and specifications are added with respect to material, density, geometry, source and source distribution of this content. These are directly based on the discussion and presentation of the revised calculational approach to qualify the content, included in Subsection 5.4.8 of the revised SAR. In this subsection, all characteristics are discussed in detail, and justifications are given for the selected limitations or why no limitation is needed for a specific parameter.

For the neutron monitors, only LPRMs with fission chambers are to be qualified. The LPRMs are segmented to fit the cask. The fission chambers are very small and contain only minute amounts of fissile material, hence there is no release of any significant amounts of gases.

NRC RAI 5-11

5-11 Discuss how activity that is not from Co-60 is accounted for in the non-fuel waste basket content (NFWB-1).

Section 5.2.4 of the application contains a list of nuclides from core components for the NFWB-1. However this list is not reflected in Appendix 7.D as allowable nuclides. Based on the list of core components, the applicant should also discuss the presence of crud and surface contamination.

The staff requests that the applicant states which nuclides are allowed for transport and how each one is limited. Table 7.D.7 of the application only contains limits for Co-60. The applicant needs to discuss how activity from other nuclides are accounted for in a mixed source.

The staff also notes that Table 7.D.7 Section I.A.1.f of the application contains a limit for Type A quantity. The applicant needs to discuss the purpose of this limit as the staff does not know what this limit is for, but the staff notes if it is intended to cover other non-Co-60 nuclides that the staff does not accept multiples of A2 quantities as an appropriate means to limit allowable quantities as discussed in NRC RIS-2013-04.

This information is required to determine compliance with 10 CFR 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-11:

The NFW specification (now in Table 7.D.8 and Table 7.D.9) has been revised. Since the content can contain a large number of isotopes, it is impractical to specify source strength limits for all isotopes. Instead, source strengths are specified by energy group, in addition to the source strength for Co-60. The specification includes limits for activation and surface contamination. The limits are established based on the calculation in revised Subsection 5.4.8 of the SAR. The reference to the A2 value has been removed.

NRC RAI 5-12

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

NRC RAI 5-13

5-13 Demonstrate that the burned impact limiter steel shell can support the weight of the package without being crushed under hypothetical accident conditions (HAC) and recalculate the dose rates as necessary.

Section 5.1.2.2 of the application states: "*Conservatively, the shielding analysis of the hypothetical accident condition for the design basis fire assumes the neutron shield is completely lost and the impact limiters steel backbone is only present.*" The staff did not locate the appropriate structural analyses to support this conclusion. As assumed in the applicant's analyses, a thermal test, as prescribed in 10 CFR 71.73(b)(4), will burn all the wooden material inside the impact limiters, leaving only the stainless skin of the impact limiter. The applicant needs to demonstrate that the impact limiter steel shell (without the crush material) can support the weight of the package without being crushed under HAC or recalculate the dose rates from the location of the crushed surface.

The information is required to determine compliance with 10 CFR 71.51(a)(2).

Holtec Response to RAI 5-13:

The impact limiter steel shell is not modeled for HAC, it is neglected together with the impact limiter crush material. Only the impact limiter steel strongbacks, shear rings, and shield rings are modeled. This approach is considered conservative, since the structural evaluations of the drop conditions show that the impact limiters remain attached to the cask, and that only local crushing of the impact limiters occurs.

NRC RAI 5-14

5-14 Discuss the effects of the puncture HAC event on external dose rates.

The staff did not locate any discussion in Chapter 5 of the application on radiation attenuation effects due to the puncture event. The staff is aware that this produced local effects to the cask surface that are likely to be averaged out at the 1 meter dose rate location. The staff has accepted this explanation for the HI-STAR 190 package where there is a larger margin to HAC external dose rate limits; however, for the HI-STAR 80 package, the staff notes that there is very little margin to the limit for HAC. Table 5.1.7 of the application states that the calculated HAC dose rate is as high as 9.8 mSv/hr. This is only a 2% margin to the limit. The applicant needs to demonstrate that consideration of the puncture event at a 1 meter location directly across from the puncture does not cause external dose rates to increase beyond regulatory limits.

The information is required to determine compliance with 10 CFR 71.51(a)(2).

Holtec Response to RAI 5-14:

The HAC dose rate calculations have been updated to include the puncture event in the MCNP models. Structural analyses concluded that there is a localized reduction in lead thickness as a result of the puncture event. The puncture is modeled as a cylindrical reduction in lead thickness. The axis of the cylinder is modeled at approximately the midplane of the fuel and the location of the puncture is modeled between the aluminum ribs to maximize dose rates. The puncture event parameters now listed in Table 5.3.7a include the localized reduced lead thickness and the radius of the cylinder used to model the puncture.

NRC RAI 5-15

5-15 Justify the assumptions with respect to lead slump within the HAC external dose rate evaluations.

The maximum axial and radial lead slump values in Table 2.7.4 of the application are not consistent with those reported in Table 5.3.7 of the application. The applicant needs to discuss the lead slump assumptions within the evaluation of external dose rates under HAC and justify that the amount of slump assumed is appropriate or conservative.

The information is required to determine compliance with 10 CFR 71.51(a)(2).

Holtec Response to RAI 5-15:

The maximum axial and radial gaps as listed in Table 2.7.4 are 0.81 and 3.55 inches, respectively (the maximum radial gap was previously 1.6 inches, but is now revised as a result of RAI 2-2). The maximum axial gap occurs in the radial lead shield and is modeled with a gap of 0.81 inches at the top and bottom of the radial lead shield. The maximum radial gap occurs in the lead in the lower forgings and is modeled as a radius reduction of 0.8 inches (i.e., diameter of 1.6 inches / 2). The maximum radial gap in the lead in the lower forgings has been revised to 3.55 inches as a result of RAI 2-2 and is modeled in the shielding analyses as a radius reduction

of 1.775 inches, i.e., 3.55 inches / 2. The calculation is conservative since in reality no lead would be removed from the cask.

NRC RAI 5-16 through NRC RAI 5-25

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

Chapter 6 – Criticality Evaluation

NRC RAI 6-1

6-1 Clarify the intended contents of the package, i.e., spent fuel or fresh fuel or both.

The applicant states on page 6.1-1: "The following fuel basket designs are available for use in the HI-STAR 80, as specified in Table 7.D.1:

- a 12-cell basket (F-12P), designed for fresh or spent undamaged PWR UO2 fuel assemblies with a specified maximum enrichment. Fresh or spent fuel assemblies are stored in all 12 cells of the basket, or 10 cells with locations 4 and 9 empty, or 10 cells with locations 5 and 8 empty;

- a 32-cell basket (F-32B), designed for fresh or spent undamaged BWR UO2 fuel assemblies with a specified maximum enrichment and fresh or spent undamaged BWR MOX fuel assemblies with a specified composition. Fresh or spent fuel assemblies are stored in all 32 cells of the basket, or 28 cells with locations 13, 14, 19, and 20 empty, or 24 cells with locations 12, 13, 14, 15, 18, 19, 20 and 21 empty, or 24 cells with locations 7, 8, 13, 14, 19, 20, 25 and 26 empty."

However, the proposed CoC appears to indicate that only spent fuel and non-fuel waste are the intended contents; fresh fuel is not the intended content. The applicant needs to clarify the intended content for this packaging design.

This information is required to determine compliance with 10 CFR 71.55(b), 71.55(d), 71.55(e), and 71.59(a).

Holtec's Response to RAI 6-1:

We apologize for the confusion. The HI-STAR 80 is not intended for the transportation of fresh fuel. The statements in Chapter 6 were only supposed to refer to the modeling approach since no burnup credit is taken for criticality control in the HI-STAR 80, and hence criticality evaluation for both basket designs are performed with fresh fuel. For clarification, we have updated the statements in Section 6.1.1.

NRC RAI 6-2

6-2 Clarify the significance of the fuel name extension in the allowable spent fuel types.

In Tables 6.1.2, 6.1.3, 7.D.2, and 7.D.3 of the SAR, the applicant identifies the allowable fuel assemblies with extension “S”, e.g., 15x15S, 17x17S, 8x8S, etc. However, it is not clear whether the extension “S” has any special meaning because this is not a typical definition of fuel assembly types. The applicant needs to clarify the meaning of the fuel type extension “S”.

This information is required to determine compliance with 10 CFR 71.55(b), 71.55(d), 71.55(e), and 71.59(a).

Holtec’s Response to RAI 6-2:

There is no specific meaning to the extension “S” other than to distinguish those types from similar but not quite identical fuel types in other Holtec transport and storage cask.

The typical Holtec naming convention for the generic PWR/BWR assembly classes is {N}x{N}{S}, where {N} is the number of the fuel rods in a row/column, and {S} is the single letter in the alphabet order for each added class to the set of {N]x{N} assemblies. Further, those fuel type identifications are used across multiple documents. For the HI-STAR 80 SAR, the naming convention has been slightly extended using a letter followed by a number. The fuel assembly classes in this application are somewhat different from the ones in other applications previously submitted by Holtec, hence the new extension is used.

NRC RAI 6-3

6-3 Provide criticality safety analyses for the fuel assembly designs as defined in Figure 6.B.7.

Figure 6.B.7 of the application shows that some BWR fuel assembly designs include a mixed load of MOX and UO_2 rods in the same fuel assembly. However, the application does not seem to include information on criticality safety analyses for a package with this type of content. The applicant needs to provide criticality safety analyses for the HI-STAR 80 package containing this type of fuel assemblies or provide a justification for how this type of mixed load assembly is bounded in an existing criticality safety analyses.

This information is required to determine compliance with 10 CFR 71.55(b), 71.55(d), 71.55(e), and 71.59(a).

Holtec’s Response to RAI 6-3:

We apologize for the confusion. The criticality safety analysis presented in Section 6.2.3 is in fact the analysis for the mixed load of MOX and UO_2 rods in the same fuel assembly as defined in Figure 6.B.7. The section has been revised, an expanded to clarify this.

NRC RAI 6-4

6-4 Provide code benchmarking analyses to determine the bias and bias uncertainty of the MCNP code for performing criticality analyses for package containing fuel assembly designs as defined in Figure 6.B.7.

Figure 6.B.7 of the application shows that some BWR fuel assembly designs include a mixed load of MOX and UO_2 rods in a single fuel assembly. However, it is not clear if the code used for criticality safety analyses has been benchmarked for this type of fuel assembly designs.

The applicant needs to perform appropriate code benchmarking analyses to determine the bias and uncertainty of the MCNP code for performing criticality analyses for package with this type of payload. The applicant also needs to include information on the selected critical experiments for the code benchmark analyses and justify that the selected experiments are appropriate for this fuel configurations.

This information is required to determine compliance with 10 CFR 71.55(b), 71.55(d), 71.55(e), and 71.59(a).

Holtec's Response to RAI 6-4:

From our perspective, the criticality benchmarking presented in the application is appropriate and sufficient for the proposed combination of Uranium Oxide and Plutonium Oxide in the fuel, and no additional bias or bias uncertainty is required to demonstrate criticality safety. This is based on the following considerations, and the large margin that is shown for the MOX assemblies compared to UO_2.

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

NRC RAI 6-5 and NRC RAI 6-6

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]