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

Chapter 1 – General Information

NRC RSI 1-1

1-1 Provide the material compositions for all MOX fuel types to be shipped in the Model No. HI-STAR 80 package and provide safety analyses, as necessary, consistent with the MOX fuel type(s) to be shipped.

The application does not include information on the material composition of the pre-irradiated MOX fuel. Based on NUREG-1617, including Supplement 1 "Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel," significant differences in material composition exist for different types of MOX fuel: larger gamma source and decay heat impact the safety evaluations of the package. Also, with an A_2 value of 1.0×10^{-3} TBq for most of the common plutonium isotopes, a relatively small increase in the plutonium-containing fines could have a significant influence on the overall containment criteria. The applicant needs to include the material compositions for all MOX fuel types to be shipped in the package and provide safety analyses, as necessary, consistent with the MOX fuel type(s) to be shipped.

This information is required to determine compliance with 10 CFR 71.43(g), 71.47(b), 71.51(a), 71.51(b), 71.51(c), 71.51(d), 71.55(b), 71.55(d), 71.55(e), 71.59(a), 71.59(b), and 71.59(c).

Holtec Response to RSI 1-1:

Transportation of MOX SNF in HI-STAR 80
The following is a description of analyses pertaining only to MOX fuel assemblies transported in HI-STAR 80. The purpose of this summary is to give the reader a concise guide to the information that is currently provided in various sections of the SAR.

MOX Fuel Assembly Characteristics:
Overall description of contents of the package is provided in Section 1.2.2 [1-1.1]. MOX fuel assembly specification is provided in Tables 7.D.1 and 7.D.3 of Appendix D. More specifically, composition (Pu vector) of evaluated MOX assemblies is provided in paragraph (j) of Table 7.D.1. Dimensions of fuel are provided in Table 7.D.3.

Thermal Evaluation
The thermal conductivity of fuel pellets is an input to determine the effective planar thermal conductivity of a fuel assembly placed inside a basket storage cell. These calculations are documented in Holtec report HI-2146291 and results presented in Tables 3.3.1.A and 3.3.1.B of the SAR. These calculations conservatively adopt a lowerbound value of 2.0 W/m-K for the pellet thermal conductivity. The thermal

conductivity of MOX fuel rod is slightly lower than that of UO_2 in the temperature range experienced under dry storage or transport. Therefore, this can potentially affect the effective fuel properties presented in the SAR. The following discussion addresses this concern:

According to Section 3.1.2.2 of Reference [1-1.3], the ratio of the MOX to UO_2 thermal conductivity is 0.92. Thus, the thermal conductivity of MOX is calculated to be larger than 2.0 W/m-K based on the lowerbound value reported in Table 3.2.3 of the SAR. Additionally, Reference [1-1.4] also shows that the thermal conductivity of UO_2 and MOX is much higher than 2.0 W/m-K for temperatures lower than 600^oC. Therefore, the pellet thermal conductivity of 2.0 W/m-K adopted in the effective properties evaluations documented in Holtec report HI-2146291 is conservatively understated for both UO_2 and MOX. The effective fuel planar thermal conductivities reported in SAR Section 3.3 remain bounding.

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390] It should also be noted that the decay heat limits provided in Table 7.D.1 of the SAR are applicable irrespective of whether the fuel is MOX or not.

Based on the above considerations, the thermal evaluations presented in the SAR remain bounding and are applicable to both UO_2 and MOX fuel assemblies.

Containment Evaluation
The containment criteria are discussed in Subsections 4.2.1 and 4.3.1 [1-1.1]. When HI-STAR 80 is loaded with SNF (including MOX) the leaktight criterion, as defined in ANSI N14.5, is applied for system leakage test. Therefore, no further evaluation of MOX source term is required in the containment chapter.

Shielding Evaluation
As stated in Chapter 5, only the F-32B basket can be loaded with MOX fuel. The description of design basis MOX assembly (including composition) used in shielding analyses is provided in Table 5.2.2 [1-1.1].

<u>Source Terms</u>
The relevant gamma, cobalt and neutron source terms for MOX assemblies are provided in Tables 5.2.6, 5.2.10 and 5.2.13 of [1-1.1], respectively. More details on source terms calculations are provided in the calculation package in reference [1-1.2] in Appendices A and D.

Note that there are two MOX fuel types listed in Table 7.D.1, 10x10 and 8x8. The shielding calculations are performed for the bounding 10x10 MOX fuel assembly. This is conservative since the heavy metal mass of 10x10 MOX fuel bounds the mass of 8x8 assembly, as shown in Table 7.D.1(h).

<u>Shielding Model</u>
The shielding model is discussed in section 5.3 [1-1.1]. The modeling of MOX assemblies in MCNP5 is similar to UO_2 assemblies. However, the difference in composition of MOX fuel is accounted for in the shielding model. The compositions and densities for the homogenized regions of the BWR MOX fuel assembly are listed in Table 5.3.6 [1-1.1]. The isotopic masses for the MOX assemblies are conservatively based on MOX assembly with a burnup of 50,000 MWD/MTU, an initial Pu-fiss

enrichment of 4.5 wt. %, and a cooling time of 15 months for the 50,000 to 70,000 MWD/MTU burnup dose rate cases.

Shielding Evaluation

Methodology of shielding evaluation is discussed in Section 5.4 [1-1.1]. The methodology of evaluation of MOX assemblies is similar to UO_2 assemblies except, that the neutron source strength of MOX assemblies varies with the burnup level raised to the power of 1.7. The peak relative burnup for BWR MOX fuel is also 1.195. Using the power of 1.7 relationship, results in a 13% ($1.195^{1.7}/1.195$) increase in the neutron source strength in the peak node for the BWR MOX assemblies. As stated in Appendix 7.D [1-1.1], only 4 MOX assemblies are permitted per cask with the locations and cooling times listed in Table 7.D.6 [1-1.1].

Criticality Evaluation

Criticality evaluation of MOX assemblies is discussed in Subsection 6.2.3 [1-1.1]. The bounding analysis is performed for fresh UO_2 fuel. Confirmation analyses show that the design basis UO_2 fuel remains bounding for MOX fuel transported in HI-STAR 80. It is shown that, any basket location that is qualified for fresh UO_2 fuel with 5.0 wt% enrichment is qualified to contain a MOX assembly that meets compositions provided in Table 6.2.6 [1-1.1]. Note that fuel composition in Table 6.2.6 is based on fuel assembly limits in Table 7.D.1 [1-1.1].

References to RSI 1-1:

[1-1.1] HI-STAR 80 SAR, HI-2146261, latest revision.

[1-1.2] “Radiation Source Terms Calculations for HI-STAR 80”, HI-2167210, revision 1.

[1-1.3] “FRAPCON-3 Updates, Including Mixed-Oxide Fuel Properties”, NUREG/CR-6534, Vol. 4, PNNL–11513, 2005.

[1-1.4] J.R. Topliss, I.D. Palmer, S. Abeta, Y.Irisa and K. Yamate, “Measurement and analysis of MOX physical properties”, In Technical Committee Mtg. on Recycling of Plutonium and Uranium in Water Reactor Fuel Windermere, UK Jul. 1995

[1-1.5] “Shielding Analysis for HI-STAR 80”, Holtec Report HI-2167211 Revision 1

Chapter 2 – Structural and Materials Evaluation

NRC Observations:

NRC RSI 2-1

2-1 Provide information on the geometric shape and size of solid “chunks” of metals with activated elements, fissile metals, and plutonium. Provide information on whether the increased end-of-life rod internal pressures from Integral Fuel Burnable Absorber (IFBA) PWR fuels were considered for their potential to result in increased radial hydride reorientation and their effect on the mechanical properties of the cladding.

The application includes various fuel/material classifications and describes contents with metals of activated elements, fissile metals and plutonium, all in solid form generally described as “chunks”. However, rubbles may be of a small “chunk” size and, for these small “chunks”, the applicant should address their potential pyrophoricity under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

The applicant explains that the drying process of high burnup fuel follows discussions made in ISG-11 Rev. 3 and the ASTM C1533 Guide. The staff has been analyzing potential hydrogen effects on high burnup cladding integrity, including the effects of radial hydride reorientation. The extent of reorientation is dependent on the peak cladding hoop stresses during drying operations, which will be higher in IFBA rods with poison materials that lead to helium generation. This application considers IFBA in the criticality analysis, but is unclear if these rods were considered for their potential to affect the mechanical properties of the cladding due to radial hydride reorientation.

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

Holtec Response to RSI 2-1:

Holtec splits the response to this RSI into two sections a) and b), as outlined below:

RSI 2-1(a) The application includes various fuel/material classifications and describes contents with metals of activated elements, fissile metals and plutonium, all in solid form generally described as “chunks”. However, rubbles may be of a small “chunk” size and, for these small “chunks”, the applicant should address their potential pyrophoricity under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

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

Holtec Response 2-1(a):

The licensing approach for high burnup fuel (HBF) reconfiguration is discussed in Section 1.4. The structural analyses of fuel rods in Section 2.11 show that the fuel is

expected to remain essentially undamaged during the hypothetical accident conditions. In addition it is noted that a Post-Shipment Fuel Integrity Acceptance Test in Section 8.1.8 is performed to identify a potential reconfiguration of the fuel prior removal of contents.

However, if the fuel reconfiguration would occur and metal chips or powder would form within the package, the oxygen is removed from the cask environment at all times, thus preventing ignition of (pyrophoric) metal powders.

During NCT and HAC, ingress of oxygen is prevented by sealed containment qualified as “leaktight”, as defined in ANSI N14.5. In addition, as shown in Table 7.1.4, the cask cavity has a pressure greater than the atmospheric pressure thus preventing ingress of oxygen.

During loading and unloading of the fuel in the spent fuel pool, as discussed in sections 7.1.2 (Loading of Contents) and 7.2.2 (Removal of Contents), the fuel is covered with water therefore ignition is prevented.

If the fuel is loaded or unloaded outside of spent fuel pool, as discussed in sections 7.1.2 (Loading of Contents) and 7.2.2 (Removal of Contents), the package is backfilled with inert gas whenever fuel is not covered with water preventing potential ignition of pyrophoric materials.

Therefore, it can be concluded that ignition of pyrophoric materials within the HI-STAR 80 is not feasible at any cask operating conditions.

RSI 2-1(b) The applicant explains that the drying process of high burnup fuel follows discussions made in ISG-11 Rev. 3 and the ASTM C1533 Guide. The staff has been analyzing potential hydrogen effects on high burnup cladding integrity, including the effects of radial hydride reorientation. The extent of reorientation is dependent on the peak cladding hoop stresses during drying operations, which will be higher in IFBA rods with poison materials that lead to helium generation. This application considers IFBA in the criticality analysis, but is unclear if these rods were considered for their potential to affect the mechanical properties of the cladding due to radial hydride reorientation.

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

Holtec Response 2-1(b):

The cask drying criteria are outlined in Section 7.1, Package Loading. The Forced Helium Dehydration (FHD) System or optionally Vacuum Drying System (VDS) is connected to the cask and used to remove moisture from the cask cavity. As the water is drained from the cask, an inert gas is introduced into the cask to prevent oxidation of the fuel cladding. The cask drying criteria are outlined in Tables 7.1.2 and 7.1.3.

To prevent change in mechanical properties due to hydride reorientation, the cladding temperature is limited to 400°C if HBF is present in the basket. Note that this is the most

likely scenario for HI-STAR 80, since the cask is specifically designed to transport recently discharged fuel from spent fuel pools to a central storage in Sweden. If the cask is loaded solely with medium burned fuel (MBF), the cladding temperature is allowed to increase 570°C during drying process.

It is recognized that hydrides present in irradiated fuel rods (predominantly circumferentially oriented) dissolve at cladding temperatures above 400°C [2-1.1]. Upon cooling below a threshold temperature (Tp), the hydrides precipitate and reorient to an undesirable (radial) direction if cladding stresses at the hydride precipitation temperature Tp are excessive. For moderate burnup fuel, Tp is conservatively estimated as 350°C [2-1.1]. In a recent study, PNNL has evaluated IFBA fuel rods for reorientation under hydride precipitation temperatures for MBF [2-1.1]. The study concludes that hydride reorientation is not credible during short-term operations involving low to moderate burnup fuel (up to 45 GWD/MTU).

Accordingly, the higher ISG-11 temperature limit is justified for moderate burnup fuel and is adopted in the HI-STAR 80 SAR for short-term operations for MBF fueled MPCs (see Table 7.1.3).

In this regard it is relevant to cite the same justification as accepted by the NRC for short term operations in HI-STORM 100 FSAR subsection 4.3.1 [2-1.2], docketed in 72-1014 prior to 2006 (shown in Exhibit 2-1.1 attached to this response) and approved for MBF storage.

To avoid confusion, please note that the IFBA rods are not considered in criticality analyses of this application. As stated in subsection 6.1.2: “*No credit is taken into account for poison material in the fuel assembly, such as the Gadolinia (Gd_2O_3) normally present in BWR fuel and IFBA normally used in PWR fuel.*”

Exhibit 2-1.1 Excerpt from HI-STORM 100 FSAR [2-1.2]

4.3.1 Evaluation of Moderate Burnup Fuel

It is recognized that hydrides present in irradiated fuel rods (predominantly circumferentially oriented) dissolve at cladding temperatures above 400°C [4.3.8]. Upon cooling below a threshold temperature (T_p), the hydrides precipitate and reorient to an undesirable (radial) direction if cladding stresses at the hydride precipitation temperature T_p are excessive. For moderate burnup fuel, T_p is conservatively estimated as 350°C [4.3.8]. In a recent study, PNNL has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperatures for MBF [4.3.8]. The study concludes that hydride reorientation is not credible during short-term operations involving low to moderate burnup fuel (up to 45 GWD/MTU). Accordingly, the higher ISG-11 temperature limit is justified for moderate burnup fuel and is adopted in the HI-STORM FSAR for short-term operations with MBF fueled MPCs (see Table 4.3.1).

† B4C is a refractory material that is unaffected by high temperature (on the order of 1000°F) and aluminum is solid at temperatures in excess of 1000°F.

References to RSI 2-1:

[2-1.1] D.D. Lanning and C.E. Bayer, Estimated maximum cladding stresses for bounding PWR fuel rods during short term operations for dry cask storage, PNNL, January 2004.

[2-1.2] HI-STORM 100 FSAR, HI-2002444, Revision 3.

NRC RSI 2-2

2-2 Provide a plan, as part of the operating procedures, to ensure that channeled BWR spent nuclear fuel (SNF), selected for loading, is undamaged in accordance with the CoC conditions.

Crud-Induced Localized Corrosion (CILC) channeled BWR SNF has the potential for corrosion-induced damage to the cladding and, therefore, cladding integrity may be uncertain. If these assemblies are not dechanneled, visual inspection or ultrasonic testing of the cladding will not be viable. The user will likely need to rely on reactor operating records and/or SNF sipping methods to reasonably demonstrate that the cladding condition is within the bounds of the CoC conditions and not grossly-breached.

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

Holtec Response to RSI 2-2:

A typical approach for qualifying assemblies as undamaged in accordance with the CoC definition has been added to Chapter 7. Since this activity will be performed in advance of the loading, and is therefore not directly part of the loading process, it is included in Section 7.0. Chapter 7, Page 7.0-3:

Fuel assembly selection and verification shall be performed by the user in accordance with written, approved procedures that ensure that only SNF assemblies authorized in the CoC are loaded into the HI-STAR 80 cask. Fuel assembly selection, and some aspects of assembly verification, are typically performed well in advance of the actual loading date, specifically with respect to the selection and verification of the assemblies to meet the definition of undamaged fuel in the CoC. A typical approach to show compliance with the CoC definition of undamaged fuel may include the following steps:

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

Fuel handling shall be performed in accordance with written site-specific procedures.

Chapter 3 – Thermal Evaluation

NRC RSI 3-1

3-1 Explain the inconsistency of the temperatures between Table 3.1.1.B and Table 3.1.3 for the vent port seal and the spray cooling seal.

The applicant provided the maximum temperature of the vent port, drain port and spray cooling seals as a co-value of 139^{o}C in SAR Table 3.1.1.B, and the initial temperatures of the vent port, drain port and spray cooling seals as 126^{o}C, 139^{o}C and 125^{o}C, separately, in Table 3.1.3 of the application.

Given that the NCT temperatures are used as the initial conditions of the HAC fire analysis, the seal temperatures shown in Table 3.1.1.B and Table 3.1.3 should be consistent. Otherwise, the applicant needs to provide and clarify, in Table 3.1.1.B, the maximum component temperatures, separately, for the vent port, drain port, and spray cooling seals.

This information is required to determine compliance with 10 CFR 71.35 and 71.71 and 71.73.

Holtec Response to RSI 3-1:

The vent port, drain port and spray cooling seals are three different containment boundary seals in HI-STAR 80 cask design. The maximum temperature of the most limiting seal component among vent port seals, drain port seals and spray cooling seals was presented in Table 3.1.1.B, which corresponds to drain port seal under NCT. The temperature rise of these seal components is different under fire accident condition since they are located at different locations in the cask. Therefore, to demonstrate containment seal temperatures remain within their prescribed limits, predicted temperature of all these seal components are reported separately in Table 3.1.3. To avoid confusion, Tables 3.1.1.A and 3.1.1.B are revised to report the maximum component temperatures, separately, for the vent port, drain port and spray cooling seals.

NRC RSI 3-2

3-2 Clarify the statements in Note (g) of Table 3.2 of the application.

The applicant stated in Note (g) of Table 3.2 that the aluminum shells and the ribs outside the cask cavity are assumed to melt under HAC. The applicant needs to clarify that:

(a) Both aluminum shells and the ribs are either not modeled in the HAC fire analysis or are modeled in the HAC fire analysis with different material properties before and after melting (as lead in Table 3.2.2),

(b) The assumption in Note (g) is either applicable only to a 30-minute fire or to both the 30-minute fire and the post-fire cooldown.

(c) The assumption for the thermal evaluation in the post-fire cooldown is conservative.

The applicant needs to explain and clarify the phenomena, described by Note (g), in the application.

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

Holtec Response to RSI 3-2:

Holtec believes that NRC staff is referring to Table 3.2.10 of the SAR. The melting temperature of aluminum (Al-6061) is 582-652°C (1080-120°F) [3-2.1], which is much lower than the flame temperature of 802°C (1475°F). Thus, the components made from aluminum are expected to melt when they are directly exposed to fire. It is noted that the lead components can also melt during fire accident. However, the lead components are enclosed within steel components, and therefore will not be lost during fire accident. It is for the same reason that the thermal conductivity of liquid lead is applied for areas above its melting point, as specified in Table 3.2.2 of SAR. On the other hand, the cask aluminum components (i.e. enclosure shell, conductive ribs and outer intermediate shells) are not enclosed by steel components. Therefore, the aluminum components are expected to be lost during fire accident. This explanation is added to Section 3.4 of the SAR.

(a) The modeling of aluminum shells and ribs in the HAC fire analysis is described in Section 3.4 of the SAR. The cask aluminum components (i.e. enclosure shell, conductive ribs and outer intermediate shells) together with the neutron shield enclosed by these aluminum components are not explicitly included in the thermal model for both the 30-minute fire and the post-fire cooldown. It is assumed that all the aluminum is lost at the beginning of the fire. The inner intermediate shells made of steel are directly exposed to fire for 30 minutes, which conservatively minimizes the thermal resistance from the fire to the cask cavity. During post-fire cooldown, this also minimizes the cask external surface area for heat dissipation.

(b) The cask aluminum components (i.e. enclosure shell, conductive ribs and outer intermediate shells) are assumed to melt when exposed to high temperature from fire. Therefore, they are not included explicitly in the thermal model under both the 30-minute

fire and the post-fire cooldown. Note (g) is revised to explicitly include the post-fire cooldown also.

(c) **[**

PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390

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Therefore, the thermal evaluation of the post-fire cooldown is conservative.

References to RSI 3-2:

[3-2.1] Aluminum Alloy 6061-T6 Material Data Sheet, ASM Aerospace Specification Metals, Inc., Pompano Beach, FL.

NRC RSI 3-3

3-3 Provide references for the statements in Note-5 and Note-6 of Table 3.2.12 of the application.

The applicant stated that the cask seal shall withstand a temperature at 250^{o}C (482^{o}F) for at least 70 hours (Note-5) and that the seal shall withstand a temperature at 320^{o}C (608^{o}F) for at least 1 hour, followed by a temperature at 200^{o}C (392^{o}F) for at least 70 hours.

The applicant needs to provide references to be able to verify such statements in Note-5 and Note-6.

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

Holtec Response to RSI 3-3:

Table 3.2.12 in the application defines the bounding temperatures for various components in the package.

Note-5 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. Parker fluorocarbon compound V1285-75 has been identified as a suitable material for these seals. Two documents from the manufacturer are provided with this response to serve as references for compliance: “V1289-75 Parker Information Sheet” and “V1289-75 Parker Compound Data Sheet.”

Note-6 is applied to denote seals that must withstand a temperature of 320°C (608°F) for at least one hour followed by 200°C (392°F) for at least 70 hours. Parker perfluoroelastomer compound FF400-80 has been identified as a suitable material for these seals. Two documents from the manufacturer are provided with this response to serve as references for compliance: “FF400-80 Parker Information Sheet” and “Parker Seal Test Report – FF400 Compression Set.”

Alternate seal compounds and alternate seal suppliers may be used, as long as the seal critical characteristics are met.

NRC RSI 3-4

3-4 Provide the NCT maximum component temperatures, as shown in Table 3.1.1.B, for the F-32B fuel basket loaded with 32 FAs, 28 FAs, and 24 FAs.

The applicant described the loading patterns for the F-32B fuel basket in Holtec Report No. HI-2156468. The loading patterns are categorized as 32 FAs (1.687 kW/FA), 28 FAs (1.928 kW/FA, and 24 FAs (2.35 kW/FA), all with the maximum permitted total heat load of 54 kW. The applicant provided the maximum fuel cladding temperatures in Table 3.3.4 and the temperature contours of F-32B basket loaded with 32 FAs, 28 FAs, and 24 FAs, respectively, in Figures 6.3, 6.4 and 6.5 of the Holtec Report HI-2156468 and determined the heat load pattern of 32 FAs as the bounding case for the F-32B fuel basket.

To justify the bounding scenario of F-32B loaded with 32 FAs for the thermal and containment review, the applicant needs to provide the maximum component temperatures (with all containment seals included), as shown in Table 3.1.1.B, for the F-32B basket loaded with 32 FAs, 28 FAs, and 24 FAs, respectively.

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

Holtec Response to RSI 3-4:

HI-STAR 80 SAR being a safety summary document, only the limiting predicted temperatures and cask cavity pressures are presented. All loading patterns and basket types are evaluated under normal conditions of transport to determine the limiting scenario by comparing the peak cladding temperature (PCT), as shown in Table 3.3.4 of the SAR. Typically, the scenario with bounding PCT also results in bounding component temperatures. Therefore, the SAR only contains the results for the most limiting scenario. Regardless, all the component temperatures for all loading patterns and basket type have been verified against their limits to ensure they satisfy the acceptance criteria.

In order to better assist the NRC staff, Holtec Report HI-2156468 is revised to document the temperature and pressure for all loading patterns and basket types under normal transport condition. The maximum temperature and pressure for F-12P basket loaded with 12FAs and 10FAs are documented in Tables 6.3.A and 6.4.A of this report. The maximum temperature and pressure for F-32B basket loaded with 32FAs, 28FAs and 24FAs are documented in Tables 6.3.B and 6.4.B of this report. For F-32B basket, the total heat load in all the three loading patterns is the same, and the cask component temperatures are essentially the same. For F-12P, the total heat load with 12FAs is slightly higher than that with 10FAs, and the cask component temperatures with 12FAs bound those with 10FAs.

NRC Observations

NRC RSI 3-5

3-5 Clarify the temperatures of package components during the fire.

In Table 3.1.3 of the application and Table 6.8 of the Holtec Report No: HI-2156468, the temperatures of the fuel cladding, fuel basket, inner closure lid, and inner seal and test plug seal at the outer closure lid are identical, at the end of a 30-minute fire, to their initial temperatures (with no increase). The applicant needs to explain such phenomena that show no increase of the temperatures on these components at the end of an HAC fire.

The applicant needs to explain these phenomena in the thermal chapter of the application or correct the typos in Table 3.1.3 and Table 6.8 of the Holtec Report No: HI-2156468.

This information is required by the staff to determine compliance with 10 CFR 71.73.

Holtec Response to RSI 3-5:

Due to the large thermal inertia of HI-STAR 80 Package, temperatures of components which are not directly exposed to fire (e.g. the fuel cladding, fuel basket, inner closure lid, and inner seal and test plug seal at the outer closure lid) are not affected by the fire immediately. Additionally, these components are not exposed directly to the high fire flame temperature and will therefore not reach local extreme temperatures at the end of the 30-minutes fire. A similar phenomenon is observed in previously approved similar cask designs in HI-STAR 180 SAR (USNRC Docket No. 71-9325) and HI-STAR 180D (USNRC Docket No. 71-9367). This explanation is also added to Holtec Report HI-2156468.

NRC RSI 3-6

3-6 Perform NCT thermal evaluations with 12-hour insolation values.

The applicant adopted the 24-hour insolation for NCT thermal evaluations due to the large mass of metal and the size of the package. However, to determine the “maximum” temperatures of the package with insolation, the applicant needs to directly apply the

constant 12-hour insolation values of 800, 200, and 400 W/m^2 for the NCT thermal evaluations, with the personnel barrier and the principal features described in Section 3.3.1.4 of the application.

The applicant needs to use the NCT results, based on a 12-hour insolation, as the initial conditions of the HAC thermal evaluations.

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

Holtec Response to RSI 3-6:

During normal transport conditions, the HI-STAR 80 Package is subjected to cyclic solar heating during the 12-hour daytime period followed by cooling during the 12-hour nighttime. However, the HI-STAR 80 Package thermal model presented in the SAR includes insolation at exposed surfaces averaged over a 24-hour time period. The HI-STAR 80 thermal analysis adopted this approach, following the methodology in the approved HI-STAR 180D SAR (USNRC Docket No. 71-9367), HI-STAR 180 SAR (USNRC Docket No. 71-9325), and HI-STORM FW FSAR (USNRC Docket No. 72-1032).

As explained in Section 3.3.1.3 of the SAR, the HI-STAR Package is cyclically subjected to solar heating during the 12-hour daytime period followed by cooling during the 12-hour nighttime. Due to the large mass of metal and the size of the Package, the dynamic time lag exceeds the 12-hour heating period. During the 12-hour daytime, the Package will not reach its steady state condition. A cyclic event of 12 hour solar insolation followed by a 12 hour no insolation will have an insignificant effect on the fuel cladding temperatures.

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390] However, in the HI-STAR 80 thermal analysis, a solar absorptivity of 1.0 is conservatively applied on the external surfaces of the cask.

Therefore, it is conservative to apply 24-hour averaged insolation with assumed solar absorptivity of 1.0.

NRC RSI 3-7

3-7 Revise the heat-up temperature (390°C) to a lower value to ensure that the cask cavity can be backfilled with helium in time for cooldown.

The applicant stated, in Section 3.3.6.1 “Vacuum Drying,” that the time required for the fuel to heat-up from an initial temperature of 100°C (212°F) to 390°C (734°F) is determined. If drying completion criteria are not met, then the cask cavity must be backfilled with helium for cooldown before it reaches the temperature limit of 400°C (752°F) for high burnup fuel, as specified in ISG-11 Rev. 3.

To reduce the risk in determining the time required for the fuel to heat-up from 100^{o}C beyond 390^{o}C, the applicant needs to revise the heat-up temperature from 390^{o}C to a lower heat-up temperature to both increase the safety margin and ensure that the cask cavity can be backfilled with helium in time for the cooldown.

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

Holtec Response to RSI 3-7:

Holtec agrees with NRC staff's observation to lower the PCT acceptance criterion during drying to ensure adequate time is available for operations. Based on operational experience, the acceptance criterion for temperature limit of HBF is lowered to 380^{o}C (716^{o}F), which leaves sufficient time to perform operations like backfilling the cask cavity with helium. Additionally, a similar requirement for MBF as 550^{o}C (1022^{o}F) is added to the same section in the SAR.

Chapter 4 – Containment Evaluation

NRC RSI 4-1

4-1 Clarify the inconsistency in the components of the containment system.

The containment system components listed in Section 4.1 of the application, e.g., cover plate bots, leak test port, test port plug/seal, spray cooling cap, orifice helical, spray cooling thread insert, etc., are not consistent with the containment system components described in Section 4.1.1. The applicant needs to clarify this inconsistency to ensure that the entire containment boundary is exactly defined.

This information is required to determine compliance with 10 CFR 71.33 and 71.51.

Holtec Response to RSI 4-1:

We apologize for the confusion. We have revised both Section 4.0 and 4.1 to provide a clearer description of the containment system and its components.

NRC RSI 4-2

4-2 Clarify whether the entire containment boundary will be leakage-rate tested during the fabrication process and the maintenance of the package.

The applicant listed, in Table 8.1.2, all components needed for fabrication leakage rate test and maintenance leakage rate test. To ensure that none of the containment components will be missed in the test, the applicant needs to add statements in the containment chapter that the entire containment boundary, including base material, welds, seals, closures, valves, or other boundary elements, will be leakage-rate tested during the fabrication process, in accordance with ANSI N14.5.

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

Holtec Response to RSI 4-2:

The entire containment boundary will be leakage-rate tested during the fabrication process, but only the portions of the containment boundary affected by maintenance activities will be leakage-rate tested after any maintenance performed on the cask. SAR Sections 4.4.1 and 4.4.4 are updated to clarify this.

NRC RSI 4-3

4-3 Provide calculations of the surface area of the typical thirteen PWR fuel assemblies.

The applicant stated, in Section 4.5.1 and Table 4.5.4 of the application, that a total surface area of 4.41 x 10^6 cm^2 of the contaminated solids is assumed and used in the calculations of the surface activity of the expected waste to determine the allowable leakage rates when loaded with NWFB-1 and this “assumed” area is greater than the surface area of thirteen typical PWR fuel assemblies. The applicant needs to provide the calculations of the surface area for review.

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

Holtec Response to RSI 4-3:

The number of 13 PWR assemblies was only intended as a comparative example, to highlight the large level of conservatism in the assumed surface area of the activated core components. Such core components would have a much smaller surface area for a given weight than a fuel assembly. Hence assuming a surface area equivalent to 13 PWR assemblies is very conservative. The discussion in SAR Chapter 4 has been revised to clarify this.

NRC Observations

NRC RSI 4-4

4-4 Clarify the test procedures of the leakage rate tests written and approved by an ASNT certification Level III specialist/examiner.

The applicant stated in Section 8.1.4 that the leakage rate test on the package containment system shall be performed per written and approved procedures in accordance with the requirements of Chapter 7 and the requirements of ANSI N14.5, 1997. The applicant needs to clarify, and add the statements in the application, that an American Society for Nondestructive Testing (ASNT) certification Level III specialist/examiner is required for approval of the leak testing procedures.

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

Holtec Response to RSI 4-4:

We confirm that an American Society for Nondestructive Testing (ASNT) certification Level III specialist/examiner is required to approve leak testing procedures. This requirement has been incorporated into SAR Revision 2.A (Chapter 8, Subsection 8.1.4 and Subsection 8.2.2) provided with this response.

NRC RSI 4-5

4-5 Clarify the criteria used for the pre-shipment leakage rate test for the containment system, particularly for the vent/drain port brushing/plug seals.

The applicant described the test methods for the pre-shipment leakage rate test, as shown in Table 8.1.2 and ANSI N14.5 Appendix A:

(1) the gas filled envelope method (A.5.4, nominal test sensitivity of 10^{-8} ref-cm^3/sec) is used for the HI-STAR package loaded with the fuel assemblies.
(2) the gas pressure drop method (A.5.1, nominal test sensitivity of 10^{-5} ref-cm^3/sec) or the gas pressure rise method (A.5.2, normal test sensitivity of 10^{-5} ref-cm^3/sec) is used for the package loaded with the non-fuel waste (NFW) contents,
(3) the alternative test method (no detected leakage, test sensitivity of at least 10^{-3} ref-cm^3/sec) for the package loaded with the fuel assemblies or the NFW contents, under the conditions that the gasket is prequalified and reusable.

The applicant needs to explain why the alternative test method, #3 above, is not applicable to the vent/drain port seals (Table 8.1.2) and if this indicates that the gaskets used at the vent port and the drain port are not allowed to be pre-qualified or reusable.

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

Holtec Response to RSI 4-5:

We apologize for the confusion and confirm that the alternative method in item (3) above is also applicable to the vent/drain port Bushing/Plug seals. Table 8.1.2 has been revised to make the editorial correction. A review of items (1) and (2) above has also led to other enhancements to SAR Tables 8.1.1, 8.1.2 and 4.5.9. The following proposed changes have been incorporated into SAR Revision 2.A provided with this response.

- SAR Table 8.1.1: The alternative pre-shipment leakage rate test acceptance criterion described in Note 3 also applies NFW packages as contents; therefore, Note 3 is now explicitly referenced.
- SAR Table 8.1.1 and SAR Table 4.5.9: The leakage rate test sensitivity is 1/2 of the leakage rate acceptance criterion per ANSI N14.5; therefore, the leakage rate test sensitivity criterion applicable to NFW packages has been revised accordingly.
- SAR Table 8.1.2: ANSI N14.5 test types A.5.1 and A.5.2 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. Therefore ANSI N14.5 test types A.5.1 and A.5.2 have been 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.
- SAR Table 8.1.2: ANSI N14.5 test types A.5.1 and A.5.2 do not apply to the NFW packages for the periodic leakage rate test; therefore, the option has been removed from Table 8.1.2.
- SAR Section 7.1, Section 7.2 and Table 7.1.5: Various steps have been updated to distinguish operations between Fuel Packages and Non-Fuel Waste Packages including the addition of specific requirements for the backfill gas of Non-Fuel Waste

Packages with appropriate consideration to leakage test types, leakage test acceptance criteria and other leakage test requirements specified in Chapter 8.

- SAR Paragraph 1.2.1.8 and Subsection 1.2.4: Editorial changes related to the backfill gas for Non-Fuel Waste Packages for consistency with the requirements in Chapter 7.
- SAR Subsection 3.1.1: Editorial changes related to the backfill gas for Non-Fuel Waste Packages for consistency with the requirements in Chapter 7.

Chapter 5 – Shielding Evaluation

NRC RSI 5-1

5-1 Provide validation analyses for SAS2H and ORIGEN-S code for the purpose of source term calculations for UO_2 and MOX fuel with burnup up to 70 GWd/MTU.

The design basis fuel burnup for the HI-STAR 80 package is 70 GWd/MTU, for both UO_2 fuel and MOX fuel. The applicant calculated the source terms for the design basis fuels with the SAS2H computer code. On page 5.0-1 of the application, the applicant states: “The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 5.1 systems.” However, from the references provided in the application for these two codes, the NRC staff was unable to find the validation information for these two code versions for burnup up to 70 GWd/MTU for UO_2 and MOX fuels.

This information is required to determine compliance with 10 CFR 71.47(b), 71.51(a)(1) and 71.51(a)(2).

Holtec Response to RSI 5-1:

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

NRC RSI 5-2

5-2 Provide a detailed description of the method used to model the non-fuel waste basket (NFWB) contents.

The staff did not find that there was enough information to determine that the NFWB model was sufficient for demonstrating regulatory dose rate limits are met. For the staff to be able to evaluate this content, the applicant needs to provide the following:

a) The applicant needs to include an explicit description of the allowable source. The description should describe either a list of the maximum activity of the allowable nuclides or break this down into allowable energy and particle per second per volume.

b) Since the applicant did not use a bounding point or line source model, justification for the assumed amount of self-shielding should be explicitly addressed. This needs to include justification for the material, density and geometry used in the evaluation. The credited minimum amount of self-shielding material must be required by the CoC either directly or by reference.

c) The applicant needs to include a discussion of source distribution within the waste. The staff did not find this in the current application. The application has a specific activity limit, however it is not clear how the applicant prevents highly non-uniform sources from being shipped.

d) The applicant needs to discuss the modeling of the cartridge container and any other components associated with the NFWB content. The applicant needs to discuss which components are included in the shielding model and discuss if they are credited for either a shielding or shoring function, or both.

This information is required to determine compliance with 10 CFR 71.47(b), 71.51(a)(1) and 71.51(a)(2).

Holtec Response to RSI 5-2:

Details of the methodology used to determine maximum specific activities and subsequent dose rates for the HI-STAR80 containing the non-fuel waste basket (NFWB-1) are provided in Appendix I of Holtec Report HI-2167211. The following discussion is based on Appendix I with more details added to address the specific questions of this RSI.

As discussed below in the response, the approach used to calculate the dose rates from the core components within the NFWB-1 is based on a well-defined bounding configuration by using:

- Fixed maximum mass of the core components;
- Co-60 source is specified as a specific activity;
- Density is fixed as a high value;
- Three source region configurations within the HI-STAR 80 are used.

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

In conclusion, the four questions in this RSI are addressed as follows:

- Co-60 is used as the source since it has the highest decay gamma energies and will therefore produce the lowest specific activity.
- By varying the mass of the core components (with a fixed density of stainless steel) and subsequently the height and inner thickness of the source region and using the limiting Co-60 specific activity based on the source region mass, the amount of self-shielding provided by the core components is accounted for within the NFWB-1.
- By varying the location of the source region within the NFWB-1 and using the limiting Co-60 specific activity based on the bounding source region location within the NFWB-1, highly non-uniform source locations are accounted for within the NFWB-1.
- The NFWB wall is credited for shielding for normal conditions, however, it is not credited for shielding during accident conditions.

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

Chapter 6 – Criticality Evaluation

NRC RSI 6-1

6-1 Provide information to demonstrate that fresh UO_2 fuel with an enrichment up to 5.0 wt% bounds MOX fuel with a large margin or provide criticality safety analyses for the MOX fuel package as necessary.

On page 6.2-4 of the application, the applicant states: "*It is well understood that fresh UO_2 fuel with an enrichment up to 5.0 wt% bounds MOX fuel with a large margin*." The applicant further references the "*Holtec International Report HI-951251, Safety Analysis Report HI-STAR 100 Cask System, USNRC Docket 71-9261, latest revision.*" However, the staff was unable to find this information in Revision 16 of this report. It is not clear either whether this assertion is valid for all types of MOX fuel or not. Based on NUREG-1617, there are significant differences in material composition for different types of MOX fuel and that will impact the criticality safety of the package. The applicant needs to provide information to demonstrate that fresh UO_2 fuel with an enrichment up to 5.0 wt% bounds MOX fuel with a large margin or provide criticality safety analyses for the MOX fuel package, as necessary.

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

Holtec Response to RSI 6-1:

We apologize for the confusion. Since we have in fact performed criticality safety analyses for MOX fuel in comparison with UO_2 fuel, as presented in Section 6.2.3 of the SAR, we do not rely on any general statement about MOX fuel for our safety evaluations. Hence we have removed the first sentence in Subsection 6.2.3 and the reference provided there.