Response to NRC's 2nd Request for Additional Information
Holtec International
Docket No. 71-9374
HI-STAR 80 Transportation Package

Chapter 5 – Shielding Evaluation

NRC RAI 5-1

5-1 As already requested by RAI 1-1, dated February 7, 2017, provide the minimum thickness or nominal thickness and tolerances of the Holtite neutron shield (typically provided within the package drawings). Similarly, as previously requested by RAI 5-19, dated February 7, 2017, consider and include the dimensional tolerances in all calculations of the external dose rates. In addition, clarify the discrepancy in the minimum required bulk density of Holtite provided within Table 8.1.10 of the application.

Section 71.33 of 10 CFR requires that the package description in the application include specific dimensions for materials used as neutron absorbers or moderators. Additionally, NUREG-1617, Section 5.5.1, indicates that reviews must verify that applications include design features important to shielding such as dimensions, tolerances, and densities of materials for neutron or gamma shielding. In reviewing the revised drawings provided in the RAI response, the staff found there was no thickness or dimensional tolerance for the Holtite neutron shield. Also the minimum bulk density (prior to aging and thermal expansion) in Table 8.1.10 of the application is less than that from Table A6 from HI-2177580, "Holtite B Application Report for HI-STAR 80." The staff uses both the minimum thickness and minimum density of the neutron shield to verify that the as modeled package is representative of the package as built and operated.

Thus, the staff cannot verify the calculated external dose rates without specific information on the dimensions and density of the neutron shield. The applicant shall include this information as part of the package drawings and modify the minimum bulk density in Table 8.1.10 of the application. The minimum dimensions for the Holtite neutron shield should be equal to those listed in Table 5.3.9 of the application and used within HI-2177580.

This information is required to determine compliance with 10 CFR 71.33(a) and 71.47(b).

Holtec Response to RAI 5-1:

The revised drawings in Section 1.3 of the SAR now contain minimum or nominal and tolerance values for all Holtite neutron shielding components. The revised calculations take those into account as follows:

- For the Holtite in the side wall of the cask, which is the largest amount of Holtite in the cask, the model uses the minimum gaps (i.e. minimum dimensions of the Holtite AND

the surrounding structure), the minimum cross sections of the Holtite parts, and the minimum density, for hot conditions of the Holtite. See Section 5.3.1 for some further discussions.

- For the Holtite in the top and bottom flange, and in the lid, the design basis models still uses the simplified modeling of the Holtite by filling the entire corresponding cavity, with an appropriately adjusted (reduced) density. However, a study is now provided in Section 5.4.11 where the minimum Holtite dimensions in those cavities are modeled according to the drawing specifications for a selected number of cases, and dose rates are compared with the design basis calculations. The comparison shows that the results are essentially identical, and in most locations the design basis calculations are bounding. The simplified modeling in the design basis is therefore considered appropriate.

With respect to density information on Holtite, the information in Table 8.1.10 and the information used and reported in Chapter 5 have been revised, and are now specified consistently with the Holtite Application report. For details on what densities are used in the shielding analyses see Sections 5.3.1.1, 5.3.2 and 5.4.10.

NRC RAI 5-2

[PROPRIETARY INFORMATION WITHHELD PER 10 CFR 2.390]

NRC RAI 5-3

5-3 Justify the Co-59 impurity assumed in the source term calculation for the hardware associated with the fuel assemblies.

Table 5.2.3 of the application provides the assumed Co-59 impurity in the spent fuel hardware activation calculations. The staff found that the assumed Co-59 impurity is significantly lower than what has previously been approved by the staff and inconsistent with the acceptance criterion provided in the Standard Review Plan for spent fuel transportation, NUREG-1617, which was derived from the PNL-6906 report, “Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal,” June 1989. Specifically, the PNL report states that a Cobalt impurity of 0.1% is considered an upper limit.

The largest contributor to external dose rates from activated fuel assembly hardware is from Co-60 due to the activation of Co-59. Co-59 is an impurity within the hardware components. Assuming a lower impurity than what is actually present will significantly underestimate the source from these components. The applicant needs to justify the Co-59 impurity data, as shown in Table 5.2.3 of the application for the source term calculations for the fuel hardware for the HI-STAR 80 package.

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

Holtec Response to RAI 5-3:

To alleviate any concerns with the selected values, they are for now specified in Section 7.D as limits for the specified content.

NRC RAI 5-4

5-4 Section 7.D of the SAR contains the specific contents that are allowed to be shipped in the HI-STAR 80 package and is referenced in the CoC. Incorporate the following changes:

a) The top and bottom of the fuel assembly does not experience as much burnup due to leakage and therefore, in some cases, fuel designers use natural uranium in these areas to improve fuel economy: these are called blankets. The axial burnup profile for a blanketed fuel assembly experiences relatively more burnup at the center than a non-blanketed fuel due to the reduced fissile material at the ends. The higher burnup in the center creates a larger source term and higher external dose rates relative to a non-blanketed assembly burnup profile. However, some of this effect is due to the way assembly average burnup is evaluated. Burnup is expressed as MWd or GWd per MTU and, therefore, even though the natural uranium does not contribute to the power of the assembly, it is considered when evaluating assembly average burnup. The applicant showed that, if the MTU within the blankets is neglected when performing the assembly average burnup calculation, then the assigned assembly average burnup value increases, and the burnup profile of this higher burnup assembly bounds that of the non-blanketed assembly. To account for the burnup profile due to blanketed spent fuel, the applicant states in Section 5.E.1.1 of the application that the average burnup of blanketed spent fuel will be calculated without the axial blankets, therefore increasing the average burnup to the extent that it is bounded by the design basis axial burnup profile. This requirement is reflected in Table 7.D.10 of the application as an adjustment to the burnup calculation for blanketed fuel; however, this table states that it only applies to the calculation of assembly decay heat. The applicant needs to include the adjustment to the burnup calculation for blanketed PWR fuel assemblies when evaluating minimum required cooling times as well (Table 7.D.4 of the application).

b) The proposed footnote to Table 7.D.2 of the application states that assemblies other than those approved in Table 7.D.2 of the application can be shipped as long as criticality analyses are performed, meet certain requirements, and be submitted to the staff within three years of loading. This statement does not meet the regulation in 10 CFR 71.35(a), which requires that a package application include a demonstration that the package satisfies the standards specified in subparts E and F. Therefore any analyses supporting the safety standards in subparts E and F must be approved prior to shipment. Provide a justification for how the proposed footnote meets 10 CFR 71.35(a) or withdraw the proposed change from the application.

c) Tables 7.D.4, 7.D.5 and 7.D.6 of the application state they are “based on” the number of 1-year irradiation cycles listed in Table 7.D.7 of the application. The source term generated by the depletion analysis is affected by the power density, and therefore, if a spent fuel assembly is burned with a larger power density than what was assumed

within the analysis, it would have a larger source term and the analysis would be non-conservative. The power density is directly related to the number of cycles because if a fuel assembly achieves a certain burnup level in fewer cycles, then that means it was burned at a higher power density. Therefore, the minimum number of irradiation cycles in Table 7.D.7 needs to be a requirement and the staff finds the language referring to this table needs to be modified to reflect this as a requirement.

d) In reference to loading MOX fuel, Table 7.D.6 of the application states, in Note 2, that "the remaining 28 storage cells must be loaded with UO2 fuel with the same cooling time for each burnup as indicated in the table above (i.e., UO2 fuel with a burnup of 50000 MWD/MTU must have a cooling time of 18 months)." Based on the loading tables for UO2 only assemblies (Table 7.D.5 of the application), the staff finds that this is conservative for most of the assemblies in that table; however, the staff finds that there is no minimum enrichment specified in Table 7.D.6 of the application. The source term is affected by the enrichment assumed within the depletion analysis. Lower enriched fuels produce higher source terms similar to burning fuel with higher power densities. If a fuel assembly has a lower enrichment then to achieve the same amount of burnup as a higher enriched assembly it must have a higher thermal flux which will then increase the amount of radionuclides produced by capture. Therefore if the fuel assembly loaded has a lower enrichment than that assumed within the analysis, the analysis could be non-conservative. The applicant needs to add the minimum enrichment as a requirement for this table.

This information is required by the staff to determine compliance with 10 CFR 71.35(a), 71.47(b) and 71.51(a)(2).

Holtec Response to RAI 5-4:

a) We apologized for the oversight. As noted in the RAI, and stated in Section 5.E.1.1.1, the adjustment is applicable to both heat load and shielding calculations. Section 7.D, Table 7.D.1 has been revised accordingly.

b) The footnote and the proposed option to qualify assemblies other than those listed in Table 7.D.2 has been withdrawn.

c) To provide more clarity and avoid misinterpretation, the cycle requirements have now been directly integrated into tables 7.D.4, 7.D.5 and 7.D.6, and Table 7.D.7 is removed. Additionally, to provide added flexibility for selecting and qualifying assemblies, the tables have been extended as follows: a) a separate column has been added to tables 7.D.4 and 7.D.5 to qualify assemblies irradiated in a reduced number of cycles, with a different (higher) minimum cooling time; b) for BWR assemblies, a further column was added for assemblies with a higher number of cycles and slightly reduced minimum cooling times; and c) and intermediate burnup level is added to each table. All this is supported by calculations and evaluations documented in Section 5.4.7.2.

d) Table 7.D.6 has been revised and now specified the minimum enrichment for the UO2 assemblies that can be loaded together with the MOX assemblies. The table also shows the corresponding minimum number of irradiation cycles for all assemblies. Note that in the revised calculations, one single UO2 specification is evaluated and qualified for all MOX assemblies.

Chapter 6 – Criticality Evaluation

NRC RAI 6-1

6-1 Pertaining to the criticality safety analysis of the mixed UO2 and MOX package:

1. Demonstrate that mixed MOX and UO2 loading patterns are bounded by the uniform loading pattern of UO2 assemblies for all possible MOX fuel designs/configuration (including Pu vector, fissile material loading and fuel configurations),

2. Demonstrate that the MCNP computer code is appropriate for criticality safety analyses of this proposed mixed fuel systems, and

3. Provide benchmarking analyses to determine the bias and bias uncertainty of the MCNP code for performing criticality analyses for the proposed package containing mixed load of MOX and UO2 fuel assemblies.

In its response to RAI 6-4, dated February 7, 2017, the applicant states that it performed criticality safety analyses for two cases for the F-32B basket with a MOX and UO2 mixed loading. Both cases contain a mixed loading of spent boiling water reactor (BWR) MOX and UO2 fuel assemblies in the F-32B basket with a limit of four MOX fuel assemblies in the center locations of the basket. The applicant further states: “The results are presented in Table 6.2.7, and show that for both cases the mixed MOX and UO2 loading patterns are bounded by the uniform loading pattern of UO2 assemblies.”

During the reviewe of the analyses, the staff found that, in both cases, the applicant used approximately 4.16 wt% Pu fissile material in heavy metal for the MOX fuel and 5 wt% of U-235 in UO2 fuel. The staff finds that these sample cases may not be representative of typical MOX fuel plutonium vector (typically in the range of 3 to 7% fissile material in heavy metal) and does not represent all mixed MOX designs because the plutonium vector of MOX fuel varies significantly from different Pu streams as pointed out by the International Atomic Energy Agency (https://www-nds.iaea.org/wimsd/critmox.htm). The staff is concerned that the amount of fissile materials for the MOX fuel used in the sample cases may be not bounding for all BWR MOX fuel assemblies. Therefore, the applicant needs to either justify that the composition of the MOX fuel as modeled is bounding for all possible MOX assemblies that can be loaded in the HI-STAR 80 or provide analyses using a bounding MOX composition.

The staff is also concerned that the MCNP code has not be verified and validated for modeling the proposed mixed load of MOX and UO2 fuel assemblies. The applicant needs to provide information on validation and verification (V&V) of the MCNP code for modeling this type of systems. With regard to code V&V, the applicant may include any general publications from the code developer or users that discuss using the MCNP code to model a mixed MOX and UO2 system. In addition, as previously requested by the staff in RAI 6-4, the applicant needs to perform adequate code benchmarking analysis to provide the bias and bias uncertainty of the MCNP code for the criticality safety analysis of the mixed MOX and UO2 system. If the information necessary to conduct V&V and benchmarking consistent with common nuclear criticality safety

practices is not available, then the applicant should justify the keff value for the bounding MOX and UO2 loading pattern based upon more conservative consideration of uncertainties of the mixed MOX and UO2 composition.

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

Holtec Response to RAI 6-1:

While the RAI contains 3 sub-questions, they are related to two separate issues, namely a) appropriateness of the plutonium vector; and b) Mixture of MOX and UO2 fuel. These are addressed separately below.

a) Plutonium Vector

The acceptable Plutonium vector is explicitly specified in the definition of the acceptable content in Appendix 7.D. Specifically, the total plutonium mass and the percentages of the fissile isotopes are specified as upper limits, and those are the values that are used in the safety analysis. Hence, while there may be MOX fuel with compositions different from those analyzed, it would not be permitted to be transported in the cask.

b) Mixture of MOX and UO2 fuel

To alleviate any concerns that may exist about the mixture of MOX and UO2 fuel in the basket, the following changes were made:

- The fuel assemblies that only contain a small number of MOX rods have been removed from the approved content and the safety analysis, and only those assemblies containing all MOX rods are retained. This eliminates any concerns of arbitrary mixtures of MOX and UO2 rods within an assembly.

- The number of a maximum of 4 MOX assemblies per basket is retained, but the locations that these assemblies are permitted to be loaded have been changed. Previously, those 4 MOX assemblies were located in the center of the basket, creating a larger area of MOX rods. For the revised content and analysis, these assemblies moved outwards, so individual MOX assemblies are no longer placed face-to-face to each other, instead, they are separated by UO2 assemblies.

Analyses are not only performed for the proposed mixture of MOX and UO2 assemblies, but also for a (hypothetical) uniform loading of MOX assemblies in the entire basket. This case shows a significant reactivity margin compared to the mixed loading or the loading with UO2 fuel only.

Together, the reduction of the number of qualified MOX assembly types, the modified location of the MOX assemblies in the basket, and the margin should be sufficient to alleviate the concerns about the mixture of UO2 and MOX assemblies in a basket.