Holtec International Report HI-2210161, "Topical Report on the Radiological Fuel Qualification Methodology for Dry Storage Systems"

Request for Additional Information

By letter dated May 31, 2021, as supplemented August 18, 2021, Holtec International (Holtec) submitted the subject topical report (TR) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

This request for additional information (RAI) identifies additional information needed by the NRC staff in connection with its review of this approval request. Each RAI below describes information needed by the staff to complete its review of the subject request.

RAI 1:

Clarify the area of applicability for the TR to address assemblies with guide tubes, water rods, or channel boxes that are made of materials that differ from the fuel rod cladding (i.e., are not zirconium alloy materials). If these assemblies are to be included, modify the TR to address the source term from these assemblies.

The staff is aware from past review experience that the material specifications for the identified assembly hardware may not always be the same as the fuel rod cladding material and that for assemblies with zirconium alloy cladding these hardware components may be steel. Thus, the TR should clearly indicate in the area of applicability the allowed materials for these assembly hardware components, making the needed changes to the TR's guidance to ensure they are adequately addressed (e.g., Table 2.2, Sections 3.2 and 3.4).

This information is needed so that the staff can determine if the method will satisfy Part 72 of Title 10 of the *Code of Federal Regulations* (10 CFR) 72.236(a).

Holtec Response:

Several changes have been made to address this RAI. Specifically, Table 2.2 has been expanded with a row on fuel assembly hardware, and the discussions in Section 3.2.2 has been expanded to more clearly explain the evaluations for any non-fuel material.

RAI 2:

Modify the TR, including the area of applicability, to address assemblies with axial blankets, particularly how the blankets are to be considered in determining the maximum assembly average burnup and minimum assembly average enrichment.

The TR does not state if assemblies with axial blankets are within the scope of the TR. If they are within the area of applicability, the TR should describe how these assemblies are to be

treated, particularly for the purposes of calculating their radiation source terms. Burnup and enrichment are important factors in the source term calculation, and those calculations typically use maximum assembly averages for burnup and minimum assembly averages for enrichment. Whether or not the axial blankets are included in the determination of those average burnup and enrichment values can have a noticeable effect on the source terms for the assemblies, particularly with blankets of sufficient length that are either low enrichment or natural enrichment. The staff understands that the axial burnup profile associated with blanketed versus non-blanketed fuel would be a subject for the method that includes the radiation transport and dose rate calculations. However, the staff notes that technical specifications for current storage systems have included definitions of these averages for burnup and enrichment that state whether or not the blankets are included.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

<u>Holtec Response:</u>

Section 3.6 is added to address axial blankets. However, since the considerations of axial blankets in the source term analyses depends on how this is treated in the radiation transport analyses, the solution is deferred to the FSAR.

RAI 3:

a) Provide a range of applicability for depletion parameters within Table 2.2 of the TR that are marked as “unrestricted.” Address this for the following parameters: In-Core Cycle Average Soluble Boron, Water Density, Fuel Density, Specific Power, Fuel Mass (Uranium).
b) Provide a baseline value for water density, or state that this will need to be defined in the qualification report and add a description of this to the methodology.

With respect to RAI 2 a), although the depletion parameters stated as “unrestricted” are in agreement with the same terminology from NUREG/CR-6716, “Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks,” March 2001, Table 5, this table was written in terms of “fuel technical specifications and common restrictions.” The methods used to determine the source terms in the Final Safety Analysis Reports (FSARs) typically make or are based on certain assumptions for these parameters. Although these parameters are not typically part of the technical specifications, staff still reviews these inputs to determine that they are represented in a reasonable way (NUREG-2216, Section 6.5.2.5). The TR should specify the ranges of the values for those parameters that are acceptable to use in the method.

Although some of these depletion parameters have been shown in NUREG/CR-6716 to have a lower impact on dose rates, there would still be restrictions on the range of parameters that can be used that would be based on any one, or a combination of, the following: (1) realistic

possible range, (2) range that has been studied within the references, and (3) range that is available in the pre-generated TRITON reactor libraries.

For “fuel mass (uranium),” this is considered in NUREG/CR-6716 to have “intermediate” importance on dose rate and is listed in Table 5 of this reference as one of the “commonly used restrictions” within technical specifications. The fuel mass should be restricted to that of the source term that is modeled (i.e., fuel assemblies to be qualified for loading must have fuel masses that do not exceed the fuel mass used in the source term calculation). As the goal of the TR is to provide flexibility in specifying contents without excessive conservatism, this may be an area (similar to burnup, enrichment, and cooling time) that could be specified on a plant-byplant basis and established in the qualification report as a loading restriction. In this case, the procedure for changing mass within the source term modeling would need to be established, and there needs to be a condition that any FSAR referencing this TR that allows plant-specific fuel mass specifications, would need to alter the shielding model to accommodate this as well. The HI-STORM 100 and HI-STORM FW FSARs (references 1 and 2 of the TR) have been cited as justification of the unrestricted range of these parameters. The staff was unable to locate the information on how the variation of these parameters impact dose rates within these documents. State the applicable sections of these documents and/or provide clarifying information on how they are to be used to justify the “unrestricted” range of applicability for these parameters.

With respect to RAI 2 b) water density, while NUREG/CR-6716 shows that moderator density would likely have a lower impact on dose from pressurized water reactor (PWR) fuel, boiling water reactor (BWR) fuel experiences a much higher variation in moderator density. As stated in NUREG/CR-6716, Section 3.4.2.5, *“The net impact of moderator density on cask shielding is expected to be low for PWR fuels. However, the axial variation in moderator density in BWR reactors can have a measurable effect on the axial dose rate profile on a cask, and increase the dose rate near the top of the assemblies where the moderator density is lowest.”* Even for PWR fuel, this section of NUREG/CR-6716 states that the neutron dose rate varied by as much as 30% over the moderator density range studied. The NUREG states that calculations assumed 40 gigawatt-days per metric ton of uranium (GWd/MTU) and a moderator density range of 0.3 – 0.7 grams per cubic meter (g/cm^3). As allowable burnup can be as high as 68.2 GWd/MTU for PWR fuel and 65.0 GWd/MTU for BWR fuel (per Table 2.2 of the TR) for higher burnups such as these, neutron dose has a much higher contribution to total dose. Also, modern operating regimes (such as those needed to support extended power up rates) can see moderator density as low as 0.107 g/cm^3 (NUREG/CR-7224, “Axial Moderator Density Distributions, Control Blade Usage, and Axial Burnup Distributions for Extended BWR Burnup Credit,” August 2016). Also, the gamma/neutron contribution to dose depends on the cask design and conditions (i.e., accident conditions usually have no neutron shield). Therefore, moderator density should be treated in a realistic or conservative way for both PWR and BWR fuel, but especially BWR fuel.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

<u>Holtec Response:</u>

All entries of “unrestricted” have been replaced with specific limits, and new sections 2.8.1 through 2.8.9 have been added to discuss and justify those. In some cases, options to extend

the limits are permitted, and this shown in Table 2.2, and discussed in the corresponding subsection of 2.8.

RAI 4:

Revise Section 2.6 of the TR to provide additional parameters or considerations for determining the needed locations and number of locations for where dose rate criteria need to be established. Ensure that the appropriate sections in Appendices B and C also address these parameters and considerations.

Section 2.6 of the TR discusses the acceptance criteria for qualifying the contents to be loaded into a cask. This section states that the number and location of dose points are to be selected in the FSAR to ensure the contribution of all assemblies in the cask are reasonably represented. However, there is no description of what would constitute that reasonable representation. Additionally, the section does not address representation of the contribution of non-fuel hardware to be qualified for loading.

Additional description of factors relevant to establishing what reasonable representation would entail should be provided in the TR. An example of information that the staff would find acceptable is the following:

- Ensuring the contribution from non-fuel hardware is addressed.
- Consideration of transfer cask and overpack orientations during different stages of operations ((un)loading, transfer, storage) and for accident conditions, ensuring that dose rate locations include locations on surfaces and features that contribute significantly to off-site dose and to occupational exposures for those conditions.
- Ensuring dose points are in regions of the important surfaces and features where, for a particular content, the contributions are most significant or pronounced (e.g., for a vertical above-ground overpack with control rod assemblies (CRAs), side surface points include points at the axial height of the overpack where the activated portions of these components are located – at an axial height that maximizes their contribution to the side surface dose rates).
- Dose points include areas of the surface/feature where dose rates are predicted to be at maximum values.
- Ensuring dose points are sufficient in number and location so that the contribution of an assembly to the dose rate can be shown to be adequately captured. Or for uniform loading or symmetric regionalized loading, the points are sufficient in number to represent that loading region (e.g., over a quadrant or octant of the cask because of the uniformity of the loading over that quadrant or octant).

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

Holtec Response:

In order to respond to this RAI, a description similar to that suggested in the RAI has been added to Section 2.6 on a preliminary basis. However, in order to be able to propose a final version, the following issues need to be discussed and resolved:

- Non-fuel hardware only results in minimal dose increase, specifically in the locations where maximum dose rates are expected. Areas where the contribution is larger are those with dose rates well below the maximum values. Specifying dose limits for those locations could be problematic since those may be affected by variations in the cask design (e.g. material changes). In general, while it is important to include the NFH in the dose analyses, it does not seem practical to link dose rate limits to NFH due to low impact on dose rates
- It is not clear what the wording “contribution of an assembly to the dose rate can be shown to be adequately captured” implies. Dose rates are always integral values to a large extend, and it is not realistic to expect that dose rate changes can be shown to be correlated to changes in the BECT of single assemblies. This should be discussed to clarify any expectations with respect to the selection of the dose locations in the corresponding FSAR/CoC LAR.

Section 2 in Appendix B has been revised to simply reference to Section 2.6 in the main report

RAI 5:

Clarify if the TR is intended to cover fuel assemblies with irradiated steel replacement rods and if so, ensure the process described in the TR, including the source term calculation method, adequately addresses these assemblies.

Section 2.8 of the TR states that the condition of the fuel mainly factors into the spatial distribution of the fuel assembly’s radiation source term versus having an influence on the assembly’s source term. Staff analyses demonstrate that assemblies where damaged fuel rods have been replaced by steel rods can have a significant impact on dose rates. While the fuel mass and its associated source term are reduced, the steel rod introduces a significant gamma source due to its cobalt content. There are also changes in the self-shielding (uranium vs. steel). Staff analyses indicate that for assemblies with these irradiated fuel rods, there is an overall increase in dose over that of an assembly with no replacement rods, especially for lower cooling times where the cobalt-60 (Co-60) source term is more significant. Because the presence of these rods is not simply a spatial distribution issue, the TR method should address source terms for these assemblies, or clearly state that the source term from these assemblies cannot be determined using this method.
This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

Holtec Response:

Table 2.2 and Section 3.2.2 are extended to recognize the possible presence and consideration of steel replacement rods.

RAI 6:

Define the values of the parameters to be selected when utilizing the ORIGAMI code in Section 3.1 of the TR.

Define the value of "NLIB" (i.e., number of burnup interpolations per cycle) that will be used (see Table 5.4.5 of the SCALE manual). Discuss options (or state if defaults will be used) that will be specified from Table 5.4.3 "Keywords in options block," and Table 5.4.6 "Keywords in srcopt block," of the SCALE manual (ORNL/TM-2005/39). State if any of the array values discussed in Table 5.4.8 of the SCALE manual will be used, and if so what the values of these parameters will be. These parameters can affect the source term as well as the code's accuracy. The staff needs to determine that the code will be used appropriately and the method is explicit enough to be considered "specifications" in lieu of the actual parameters of those listed in 10 CFR 72.236(a).

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

<u>Holtec Response:</u>

New Section 3.5 has been added to address and clarify the necessary parameters and corresponding values for ORIGAMI calculations.

RAI 7:

Discuss how gamma and neutron energy group structures will be used within the shielding (i.e., transport) calculations.

Section 3.2 and 3.3 of the TR discuss energy group structures to be used for gamma and neutron sources, respectively. Discuss how these group structures (TR Tables 3.5 and 3.6, or similar) will be used in the shielding evaluations (i.e., sample at each energy bin's mid-point or upper energy, sample from histogram distribution, etc.).

Considering the TR describes a "response function" method in Step 3 of Section 4.0, the staff assumes that the source term energies are represented as a single energy for each group established in Tables 3.5 and 3.6. The particular energy that is chosen to represent the group is essential for the staff to understand whether or not these group structures are appropriate. The staff needs this information to determine that the proposed group structures are appropriate to account for the energy of the gamma and neutron emissions from the spent fuel nuclides. The use of the energy group structure will become a condition of the method in this TR.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

Holtec Response:

Section 3.2 and 3.3 have been revised to clarify the energy structures, and that they are either to be used as listed in the TR, or that a different structure may be defined in the FSAR, reviewed and approved through an LAR.

RAI 8:

Provide additional information justifying the appropriateness of the 800 parts per million (ppm) Co-59 impurity assumption for the breadth of steel and Inconel spent fuel and non-fuel hardware contents within the scope of the TR. Alternatively, include this value as a limit for contents being qualified using this TR within the area of applicability in Table 2.2 of the TR. Alternatively, update the methodology to define this as a site-specific quantity that needs to be justified and included within the qualification report.

Reference 3 of the TR is not an appropriate reference for the 800 ppm Co-59 impurity level as it is referencing an input to another calculation. Justification of the cobalt level for steel and Inconel components needs to be based on data showing it is reasonable for this material, such as measurements of impurities. Although the staff understands in principle that there has been an effort by the industry to reduce the Co-60 as an activation product, the staff does not have information on what the actual impurity level is for Co-59 for more modern fuel assemblies, or for non-fuel hardware components, nor when the change to lower impurity levels occurred.

The staff notes that an impurity level of 800 ppm is significantly lower than the values listed for Inconel materials cited in available literature (see Section 6 of ORNL/SPR-2021/2093, Radulescu, et. al, “Fuel Assembly Reference Information for SNF Radiation Source Term Calculations,” September 2021 (https://doi.org/10.2172/1819561)), which are around 5,00010,000 ppm. Also, stainless steel could have impurity levels that have been reported to be as high as 2.2 g/kg (PNL-6906, Luksic, “Spent fuel assembly hardware: Characterization and 10 CFR 61 classification for waste disposal: Volume 1, Activation measurements and comparison with calculations for spent fuel assembly hardware,” June 1989 (https://doi.org/10.2172/5940840). Therefore, the staff needs additional information on the data justifying the 800 ppm Co-59 impurity for all fuel structural materials and non-fuel hardware for contents that will be qualified using this method. Alternatively, this parameter can be used as a plant-specific input to the method and become a restriction for the qualified contents. In this case, actual Co-60 levels for a given fuel assembly and/or non-fuel hardware can be used as long as they are justifiable by the plant.

In addition, Sections 3.2 and 3.4 of the TR include justifications for using lower cobalt levels of more recently fabricated assemblies and non-fuel hardware and why that would be bounding for older items fabricated with higher cobalt levels. If the analyst is seeking to qualify older fuel or non-fuel hardware with higher cobalt levels, the analyst needs to use the actual cobalt levels of those items in the analysis and require longer cooling times for these components if that is what is needed to reduce dose rates. The approach currently described in Sections 3.2 and 3.4 may be acceptable for a representative analysis, such as that presented within the FSAR using this TR, but fuel and non-fuel hardware analyses used to qualify fuel and non-fuel hardware need to use actual or bounding values for what will be loaded.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

<u>Holtec Response:</u>

Consideration of Cobalt has been revised (Section 3.2.2) to be more consistent with available references, and also more consistent between fuel assembly hardware and NFH. To address concerns with older assemblies, the levels to be assumed now depend on the age of the assembly. However, the option to use lower number is added, if such are available from valid references.

RAI 9:

Provide the following information related to the specification of non-fuel hardware (NFH)

a) Provide a mass limit for NFH within the area of applicability in Table 2.2 of the TR or modify the TR to state that higher and lower mass NFH will be reflected within the analyses and qualification report as appropriate. Additionally, for thimble plug devices (TPDs), ensure the TR's area of applicability and analysis methods address the TPD configurations for which the TR is to be applicable. According to DOE/RW-0184, "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," some TPD configurations include varying numbers of absorber or water displacement rods that extend into the active fuel zone. Thus, the TR's area of applicability and the analysis methods should address these TPD configurations or specifically exclude them.

   Section 3.4.1 and 3.4.2 of the TR discuss the procedure for determining the source from TPDs and burnable poison rod assemblies (BPRAs), and CRAs and axial power shaping rods (APSRs), respectively. It includes masses to be used in Tables 3.4, 3.7 and 3.8 of the TR that represent hypothetical TPDs, BPRAs, CRAs, and APSRs for use in activation analyses. A limitation for applicability must be set in Table 2.2 of the TR as this may not be appropriate for components not described in DOE/RW-0184 (References 5 and 6 of the TR). Section 3.4.1 of the TR states that *"lower masses can be used with appropriate reference and documentation."* If lower masses are assumed, then this must be included as a restriction within the qualification report. Alternatively, if higher mass NFH are needed, this could also be stated within the qualification report, as long as, the method is modified to address the increased source term associated with this option.

b) Provide additional information on how the burnup (i.e., exposure) for NFH will be determined. Section 3.5 of the TR discusses the process of determining the amount of Co-60 for NFH. This section does not discuss how the burnup within the ORIGAMI calculation is to be determined for these components. Sections 3.4.1 and 3.4.2 of the TR have some general information on the "burnup" these components might experience but does not specifically state what should be used as input into the calculation. Provide information stating what the "burnup" for each component is and include this within the

area of applicability in Table 2.2 of the TR. Alternatively, this could be determined on a site-specific basis and added to the qualification report. If so, modify the TR to add steps to ensure that this is clearly stated, and that this information will be added to the qualification report if necessary.

c) Clarify that the lengths specified in Table 3.7 of the TR represent a ten percent insertion of the CRA into the core (active fuel zone) and ensure the area of applicability (Table 2.2 of the TR) specifies the amount of CRA insertion for which the TR is to be applicable.

   The assumed amount of insertion of a CRA is important to determining the amount of activation and source term for the CRA. The amount of insertion is also important for determining the axial extent of the source within the radiation transport (dose rate) calculations. Alternatively, to be able to qualify CRAs that have been inserted more than 10%, the TR needs to establish a method where the length of the activated area is increased accordingly and the amount of CRA insertion needs to be included in the qualification report.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

<u>Holtec Response:</u>

a) Given that NFH only has limited effect on dose rates, hence variations in the NFH masses are considered to have a second order effect, and would therefore not be important enough to be considered in the area of applicability. Section 3.4 has been revised.
b) This has been clarified, see in Section 3.4
c) A discussion has been added to Section 3.4.2

RAI 10:

Provide the following modifications and clarifications to the TR to address the source term for neutron source assemblies (NSAs):

a) Revise the first option for determining allowable neutron source assemblies (NSAs) to account for the gamma source term from NSAs. Section 3.4.3 of the TR describes the method for determining allowable neutron source assemblies. There are three options. The first option states: *"If an evaluation is performed that shows that the neutron source term from an NSA is negligible, there is no limit on the number or location of NSAs in the basket… During in-core operations, the stainless steel and Inconel portions of the NSAs become activated, producing a significant amount of ^{60}Co."* Allowing an unlimited number of NSAs if the neutron source is negligible may be appropriate in terms of its impact on neutron dose, however it does not address the gamma component of the NSA's source term. This statement needs to be revised to indicate that there may still be limitations for the NSAs based on the gamma source.
b) Revise the TR to include a method for determining the gamma source from NSAs and revise Table 2.2 to update the area of applicability for allowable NSAs that are covered

by the method (i.e., maximum mass and exposure). If NSAs are to be bounded by other non-fuel hardware components (such as BPRAs), ensure the method and Table 2.2 clearly states that. Also, ensure the method and Table 2.2 include all configurations of NSAs for which the TR is applicable. According to DOE/RW-0184, “Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require LongTerm Isolation,” some NSA configurations include varying numbers of absorber rods in addition to the source rods. If the method is specific to NSAs (i.e., they aren’t bounded by other non-fuel hardware), a table similar to Tables 3.4, 3.7 and 3.8 of the TR may be needed for NSAs, which would then also need to be referenced in the appropriate steps in Section 3.5 of the TR.

c) Clarify if each individual NSA is to be less than 1% of the total cask neutron source, or if all NSAs combined would contribute less than 1% of the total cask neutron source. Section 3.4.3 of the TR states: “*The contribution can be considered negligible if it provides less than 1% of the total neutron source term of a cask.”* Although “less than 1%” could be interpreted as being negligible, current baskets can allow up to 37 assemblies. Future designs may have more assemblies than this. If each NSA contributed 1% to the total cask neutron source term, then each NSA could contribute up to 37% more to the neutron source of each basket cell in a 37-assembly basket. This is considered a non-negligible source increase and could have a significant contribution to occupational and public dose.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

<u>Holtec Response:</u>

In general, Section 3.4.3 has been revised to address the RAIs.
a) Section 3.4.3 has been updated to clarify that the gamma source from NSAs has to be considered, independent of the option for the neutron source.
b) See response to RAI 9a above.
c) In Section 3.4.3 it has been clarified that 1% means for all NSAs together, not for a single NSA. Also, a brief discussion has been added about secondary sources, where this condition is most likely to apply.

RAI 11:

Modify Section 4.0 “Analysis Process” to include the following necessary steps and information.

a) Include within “Step 1” the need to identify site specific depletion parameters and analytical deviations from the TR and that these must be appropriately documented and justified within the qualification report. The TR allows some analytical assumptions and depletion parameters to vary.
b) Revise the second bullet of Step 1 to read as: “See Section 2.8 and Table 2.2, in combination…” Section 2.8 also contains relevant guidance for defining the area of applicability.

c) Update Step 2 to add references to TR Sections 3.2, 3.4 and 3.5 to provide clear, explicit direction regarding modeling and calculation of the activation sources (cobalt-60 and, for CRAs, AgInCd) for the spent fuel assembly hardware and for the non-fuel hardware.
d) Include within "Step 3" of Section 4.0 in the step where changes are to be made to the model, the need to identify site specific parameters (i.e., changes that are not made using 10 CFR 72.48 which is already stated) and state that these must be appropriately documented within the qualification report. Some designs have variable parameters important to the shielding design such as variable thickness transfer casks or variable concrete density.
e) Revise the first bullet of Step 3 to clearly identify that dose rates for the new contents (i.e., the contents to be qualified) will always need to be calculated. As described in the current text, the staff understands that new radiation transport calculations may not be necessary due to use of the response function method; however, the dose rates for the contents will always need to be calculated.
f) Revise the first sub-bullet of the second bullet of Step 3 to clarify that the model that will be used will be one that is covered by the FSAR revision, including relevant changes modified per 10 CFR 72.48, that corresponds to the CoC amendment the contents are being qualified under. It should be made clear that the model used for an earlier qualification report may not be appropriate if it is for a different CoC amendment or does not contain applicable changes made using the 10 CFR 72.48 process.
g) Modify the second sub-bullet of the second bullet of Step 3 regarding references to Sections B.1 and C.3. Section B.1 does not describe any limits to changes to the radiation transport model beyond what is stated in this sub-bullet. Further C.3 is only an example of the kinds of limitations on such changes, which may differ for a different system.
h) Include a step to define inputs similar to Step A in the examples included in Appendix A. This is an important part of the process necessary for understanding the scope of the qualification calculations.
i) Modify Section 4 of the TR to ensure that it provides applicable guidance on source term development for contents qualification versus for FSAR demonstration analyses. Specifically, the fifth bullet of Section 4, Step 3 directs the analyst to Section B.4 and Section C.4 for directions on how to select the set of burnup, enrichment, and cooling times to be analyzed. Section B.4 is guidance for the CoC holder/applicant to develop a representative source for the FSAR demonstration analyses. Appendix C.4 is an example that illustrates the guidance in Section B.4. The contents qualification should use the burnup, enrichment, and cooling time combinations that the analyst seeks to qualify, and not the representative sources discussed in Section C.4 and B.4 of the TR. Additionally, the meaning and intent of the sixth bullet of Section 4, Step 3 and its appropriateness for the qualification analysis is also unclear.

Section 4.0 of the TR contains the method of evaluation that needs to be followed to meet 10 CFR 72.236(a). This information needs to be clear for the staff to be able to make a finding that the method is explicit enough to be considered "specifications" in lieu of the actual parameters of those listed in 10 CFR 72.236(a).

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

Holtec Response:

Section 4 is updated for inclusion or clarification of the various subsections of the RAI. However, note the following:

h) : Appendix A is essentially just to show examples of FQTs. In order to avoid duplication in the document, a reference to Section 4 has now been added there, instead of adding information to the appendix.

RAI 12:

Modify the examples in Appendix A to reflect the processes in Section 4 of the TR.

In the acceptance review, the staff identified that the TR did not include a description of the analysis process that the TR was meant to define. It did include an appendix of examples. The TR now includes a description of the analysis process steps in Section 4. However, Appendix A was not updated to reflect this process.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

Holtec Response:

Appendix A is essentially just to show examples of FQTs. In order to avoid duplication in the document, a reference to Section 4 has now been added there, instead of adding information to the appendix.

RAI 13:

Provide guidance regarding averaging of dose rates that ensures the respective limit will be meaningful for qualification of contents (fuel and non-fuel hardware) to be loaded into a cask.

The TR indicates that averaging of dose rates used as acceptance criteria may be part of the process covered by the TR (e.g., see Section B.2 and Appendix C). While averaging may be a useful component of the process, some guidance is needed that ensures the averages are a meaningful indicator of the contents' qualification. The examples in the TR should also reflect this guidance. An example of the information that the staff would find acceptable is the following:

- Only dose points for the same source contributor are averaged together and the limit for the average is for capturing that source contributor (e.g., for a cask with CRAs, there would be limits on average dose rate in the areas of the cask affected by the CRAs and separate limits for a zone minimally, or negligibly, affected by the CRAs).

- Only dose points for the same zone or region of behavior of the contents' source are averaged together and the limit for the average is for that zone or region (e.g., radial surface dose points at locations that are the peak of the axial dose rate profile of the contents aren't averaged with dose points that aren't at locations of the peak axial dose rates on the radial surface).
- Dose points that are averaged together and the corresponding limit are for an area of the cask surface/feature and contents configuration that have sufficient similarity or homogeneity.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

Holtec Response:

The option to define average in addition to maximum dose rate limits has been removed throughout the document.

RAI 14:

Provide the following clarifications or modifications:

a) Clarify what is meant by the "larger range of systems" discussed in the second paragraph of Section 2.3. It needs to be clear that qualification using this method would not apply to multiple dockets unless those systems have this TR as part of their licensing basis.
b) Clarify the language in the fourth paragraph of Section 2.2 and in the paragraph at the top of page 7 in Section 2.8 to describe that the FSAR/TS will define the storage system's allowable contents in terms of the ranges of authorized burnups and enrichments that will be within the area of applicability of the TR. The current language can be interpreted as meaning that the "area of applicability" for the TR will be defined in the FSAR/TS.
c) Modify Section C.3 to clearly state that the comparable section of an FSAR must establish criteria that determine or ensure that acceptance criteria dose rate points (locations and values) are still valid when changes are made. Modify the statement that follows the bullets of changes that are not permitted in Section C.3 to read as "Note that any change or adjustment has to be also validated against **10 CFR 72.48 and** all other safety requirements, not just shielding." (bold indicates requested text change)
d) Change the "may not" in the last paragraph of Section 3.4.2 to be "shall not", similar as to what is written for BPRAs in Section 3.4.1.
e) Add "or Table 3.7 (for CRAs) or Table 3.8 (for APSRs)" as well as any table for information related to NSAs (if any is added as a result of the other RAIs) to the last bullet of Section 3.5. These other tables have the appropriate flux factors that should be used for flux scaling for the respective non-fuel hardware components.
f) Change "location" to "locations" in the 7th bullet and "limit" to "limits" in the 8th bullet in Step 3 of Section 4.

g) Clarify what is being described in paragraph 3) of Section B.4. It is not clear what the words “each case” and “both conditions” are referring to.
h) Modify the third paragraph of Section D.7 to correctly describe what is described in Reference 6 of this example qualification report. Reference 6 is the TR. Contrary to this paragraph in Section D.7, the TR does not provide source terms (it provides the source term calculation method), nor does it describe the method of tallying.
i) Section 3.1 of the TR states, *“The codes to calculate neutron and gamma source terms should be the TRITON and ORIGAMI modules of the SCALE 6.2.1 system [7].”* Modify the term “should be” to “shall be.” This sentence indicates that TRITON is actively being used to calculate the gamma and neutron source terms as part of the method described in this TR. Although the staff agrees that TRITON is involved in the process by way of generating the supplied reactor libraries, it would not actively be exercised as part of this method. This needs to be modified to indicate that TRITON was used to develop the supplied cross section libraries that will serve as input to the ORIGAMI module of SCALE 6.2.1. This clarification will make clear to users that the ORIGAMI module is the module that must be used in applying the method.
j) Section 3.1 of the TR states, *“The SCALE calculations shall be utilizing data libraries with the maximum number of energy groups available for the respective code version. For SCALE 6.2.1, this is the 252-group library based on ENDF/B-VII.1 nuclear data.”* This statement is written in future tense as if it’s something that an applicant will be doing as a part of the method of evaluation when it is describing acceptable reactor libraries and the nuclear data used to generate these libraries. This sentence should be clarified to state that the ORIGAMI calculations will use the TRITON data libraries supplied with the code, using the 252-group libraries based on ENDF/B-VII.1 nuclear data.

The requested clarifications and modifications are needed to ensure a clear and correct understanding of the TR’s descriptions of the method and its application as well as the examples. It is also needed to ensure the examples, though not comprehensive, are meaningful and align with the TR’s descriptions of the method and its application.

This information is needed so that the staff can determine if the method will satisfy 10 CFR 72.236(a).

<u>Holtec Response:</u>

The report and appendices have been updated to address all subitems of the RAI. However, note the following.

d) While typically the NFH material is not credited in the radiation transport analyses, this would be a subject to be addressed in the discussion of the radiation transport calculations in the FSAR, not this TR. We have therefore left the option open here. This discussion is also now moved to Section 3.2.2, to cover all NFH.

e) This discussion has now been relocated to Section 3.2.2