U.S. NUCLEAR REGULATORY COMMISSION

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

HOLTEC INTERNATIONAL TOPICAL REPORT HI-2210161

"TOPICAL REPORT ON THE RADIOLOGICAL FUEL QUALIFICATION METHODOLOGY FOR DRY STORAGE SYSTEMS"

1.0 INTRODUCTION

By letter dated May 31, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession Package No. ML21152A257), as supplemented on August 18, 2021 (ML21230A336), February 7, 2022 (ML22038A940), February 1, 2023 (ML23032A449), and April 14, 2023 (ML23104A377), Holtec International (Holtec) submitted to the U.S. Nuclear Regulatory Commission (NRC) Holtec Report HI-2210161, Revision 4, "Topical Report on the Radiological Fuel Qualification Methodology for Dry Storage Systems," (Topical Report) for review and approval.

2.0 BACKGROUND

Spent fuel dry storage certificate of compliance (CoC) applicants perform dose rate calculations to demonstrate that their CoC provides adequate shielding of radioactive contents in compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 72.236. Ideally, this dose rate calculation uses design parameters that represent the minimal shielding capability of a system permitted and a source term that bounds all allowable contents from a dose perspective. Although there are many depletion parameters that affect the characteristics of the source term of a spent fuel assembly, the most significant are maximum allowable fuel mass, maximum burnup, minimum cooling time, and minimum enrichment. Other spent fuel parameters have less of an impact on source term (Ref. 1) and therefore can be treated in a bounding way within the analysis (versus being limited in Technical Specifications). Co-60 generated by activation of Co-59 impurities within fuel hardware and non-fuel hardware (NFH) is also characterized and accounted for within these evaluations.

When the NRC first began approving CoCs for dry storage systems, CoC applicants included a single bounding burnup, enrichment and cooling time (BECT) that defined the spent fuel source term and limited all allowable spent fuel contents in all locations of a basket. This configuration is referred to as uniform loading. However, over time, CoC applicants began requesting certification for systems with non-uniform configurations, sometimes referred to as "regionalized," or "zoned" loading patterns. Therefore, certain locations within a basket would have different allowable BECT limits than other locations. Some dry storage systems have evolved such that loading patterns have BECT limits defined almost on a cell-by-cell basis. These loading patterns can be developed to limit dose but more often are needed to ensure decay heat is limited. CoC applicants state that these highly heterogeneous loadings may be needed to efficiently empty the inventory of an entire spent fuel pool into dry storage systems.

Current CoCs no longer define allowable contents in terms of a single BECT but rather tables or correlations of allowable BECTs where lower enrichments and higher burnups are balanced with higher cooling times. These tables, sometimes referred to as fuel qualification tables (FQTs), may have hundreds of allowable BECT combinations. There may be multiple tables for a single basket where assemblies have different loading limits depending on its location within the basket. Applicants may also have separate FQTs for inserts and damaged fuel or fuel with different uranium mass loadings.

Demonstrating that a system has adequate shielding for all the allowable source terms often involves hundreds (or more) calculations. Applicants need to supply, and NRC staff subsequently has to review, a lot of information to justify and confirm that calculations performed bound all allowable contents. In addition, applicants are now designing systems that have variable shielding design parameters such as thickness of lead shielding transfer casks to meet the needs of sites based on lifting crane capacity or variable concrete densities. Based on this evolution of increasingly complicated shielding analyses for dry storage systems, the staff has been considering alternative means by which applicants could demonstrate compliance, and which the NRC staff could perform its regulatory reviews more efficiently, while maintaining the same level of safety.

Simultaneously, in November 2019, the National Energy Institute (NEI) submitted a paper to the NRC titled, “Defining Spent Fuel Performance Margins” (Ref. 2). The paper provided recommendations to improve the efficiency of dry storage system certification given known but undefined performance margins. For example, in the area of shielding, loaded casks often exhibit far lower measured dose rates than analyzed within the safety analysis reports. This is because when there are uncertainties in source term and dose rate calculations, conservative assumptions are made resulting in an increase in calculated dose (e.g., higher Co-59 impurities, etc.). Similarly, depletion parameters and allowable contents are not defined for an individual cask loading, but rather generically with additional margin so that multiple casks can be loaded using the same content specifications. In December 2019, NRC staff stated that it would explore the recommendations presented in the NEI paper (Ref. 3).

Throughout 2020, NRC staff and NEI held a series of workshops (Ref. 4 - 10) to discuss, amongst other things, an alternate approach to defining depletion parameters and allowable contents. In so doing, vendors and licensees could reduce some of the margin by creating allowable loadings that are more specific and do not include unnecessary conservatisms. NRC staff and NEI discussed alternative approaches for defining contents. The Topical Report, which is the subject of this safety evaluation (SE), is the result of those interactions. The Topical Report presents an NRC approved methodology that permits more efficient shielding reviews and fuel qualification for certification and oversight while still assuring the same level of safety.

3.0 REGULATORY EVALUATION

Certification of a spent fuel dry storage system is governed by Subpart L, “Approval of Spent Fuel Storage Casks,” of 10 CFR Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste.” Section 72.234, “Conditions of approval” states in paragraph (a), “The certificate holder and applicant for a CoC shall ensure that the design, fabrication, testing, and maintenance of a spent fuel storage cask comply with the requirements in 10 CFR 72.236.”

Section 72.236, “Specific requirements for spent fuel storage cask approval and fabrication,” paragraph (a) contains requirements for the content specifications. It states:

Specifications must be provided for the spent fuel to be stored in the spent fuel storage cask, such as, but not limited to, type of spent fuel (i.e., BWR, PWR, both), maximum allowable enrichment of the fuel prior to any irradiation, burn-up (i.e., megawatt-days/MTU), minimum acceptable cooling time of the spent fuel prior to storage in the spent fuel storage cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., intact assembly or consolidated fuel rods), the inerting atmosphere requirements.

Section 72.238, "Issuance of an NRC Certificate of Compliance," states: "A Certificate of Compliance for a cask model will be issued by NRC for a term not to exceed 40 years on a finding that the requirements in 10 CFR 72.236(a) through (i) are met."

Section 72.236(d) states: "Radiation shielding and confinement features must be provided sufficient to meet the requirements in 10 CFR 72.104 and 72.106."

The objective of the NRC's regulations for spent fuel dry storage system design is to assure that the system meets certain criteria. Pertaining to the shielding design, the requirements are the dose limits prescribed in 10 CFR 72.104 and 72.106.

Specifically, 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," requires that:

(a) During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid and 0.25 mSv (25 mrem) to any other critical organ as a result of exposure to:
    (1) Planned discharges of radioactive materials, radon and its decay products excepted, to the general environment,
    (2) Direct radiation from ISFSI or MRS operations, and
    (3) Any other radiation from uranium fuel cycle operations within the region.
(b) Operational restrictions must be established to meet as low as is reasonably achievable objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations.
(c) Operational limits must be established for radioactive materials in effluents and direct radiation levels associated with ISFSI or MRS operations to meet the limits given in paragraph (a) of this section.

Further, 10 CFR 72.106, "Controlled area of an ISFSI or MRS" states that:

(a) For each ISFSI or MRS site, a controlled area must be established.
(b) Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 0.05 Sv (5 rem), or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv (50 rem). The lens dose equivalent may not exceed 0.15 Sv (15 rem) and the shallow dose equivalent to skin or any extremity may not exceed 0.5 Sv (50 rem). The minimum distance from the spent fuel, high-level radioactive waste, or reactor-related [greater-than-class C] waste

handling and storage facilities to the nearest boundary of the controlled area must be at least 100 meters.

The requirements of Subpart L of Part 72 work in complement with one another. The source term, derived using specifications provided in accordance with 10 CFR 72.236(a), is used to demonstrate compliance with 10 CFR 72.236(d). The Topical Report and its conditions for implementation present an alternative means for demonstrating compliance with these regulations. Instead of explicitly defining parameters within the CoCs (e.g., burn-up (i.e., megawatt-days/MTU), minimum acceptable cooling time of the spent fuel prior to storage in the spent fuel storage cask), CoC applicants can provide specifications using the radiological fuel qualification methodology, i.e., the Topical Report. The Technical Specifications of the CoC would incorporate the NRC approved methodology to qualify all fuel contents. The Technical Specifications would also specify acceptance criteria, such as defined surface source or key dose rate points (Ref. 9). Use of the methodology, and its limited scope of applicability, combined with the acceptance criteria, provides an alternative, more efficient approach to demonstrating compliance with 10 CFR 72.236(a) and (d).

There is precedent for this approach in the licensing of nuclear power reactors under 10 CFR Part 50 that use a “Core Operating Limits Report” or COLR, as discussed in Generic Letter 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications” (Ref. 11). This generic letter states that cycle-specific parameter limits in Technical Specifications can be replaced by a methodology for determining the limit. It further states that: “the NRC review of proposed changes to TS for these limits is primarily limited to confirmation that the updated limits are calculated using an NRC-approved methodology.”

Similar to the approach outlined in GL 88-16 (Ref. 11), where cycle specific operating limits are no longer required to be documented in the Technical Specifications but can be instead documented in a COLR, when loading a spent fuel dry storage cask licensees can follow the methodology in the Topical Report and document the allowable contents in a “Qualification Report.” A Qualification Report in conformance with the Topical Report will contain a description of the allowable spent nuclear fuel (SNF) specifications for each unique SNF loading or group of loadings. The Qualification Report will include parameters such as burnup, enrichment, cooling time, fuel mass as well as any depletion parameters associated with that fuel that need to be restricted and information on allowable inserts (for pressurized water reactors). A Qualification Report also provides NRC inspectors a means to verify whether the licensee has followed the methodology for determining the loading requirements.

Using a methodology to determine allowable source terms means that a storage system could be modeled using a source term with more accurate characteristics, rather than overly conservative characteristics as the dose rate analysis is flexible enough to encompass an as-loaded storage cask, a specific site, or group of sites. Also, since applicants would not have the need to have so many permutations of loadings to identify all potential variations of fuel and depletion parameters used at sites, calculations for the systems should be more straightforward, with users analyzing their loadings to confirm it is within the bounds of the specification (rather than performing hundreds of calculations to support certification to determine bounding/representative source terms for these systems) that can be more easily verified by NRC review staff and inspectors, improving regulatory oversight of the systems.

4.0 TECHNICAL EVALUATION

4.1 Implementation of this Topical Report

Implementation of this methodology to a 10 CFR Part 72 CoC application (i.e., for an initial application or an amendment) is discussed in section 2.0 of the Topical Report. The Topical Report implementation is expected to be similar to COLR evaluations for 10 CFR Part 50 applications, as discussed in section 3.0 of this SE. As part of a 10 CFR Part 72 CoC application, the applicant would propose use of the Topical Report as a condition in the Technical Specifications of the CoC. The NRC staff recommend doing so in the administrative controls section. Additionally, the applicant would propose certain acceptance criteria in the Technical Specifications (e.g., see tables C.1 and C.2 of appendix C to the Topical Report). During its review, the NRC staff would need to determine that the storage system design is within the area of applicability of the Topical Report, see section 2.8 and table 2.2 of the Topical Report and section 4.2.1 of this SE for more information. The NRC staff would also review the acceptance criteria to ensure that a storage system that meets the acceptance criteria would also meet regulatory limits in 10 CFR 72.104 and 10 CFR 72.106. At this time, the NRC would also review and approve the dose rate calculation method (referred to as the Radiation Transport Calculation Methodology in the Topical Report) to be used as part of the methodology for developing loading patterns.

4.2 Description of Methodology

The methodology for determining allowable spent fuel is discussed in section 4.0 of the Topical Report and includes several steps. The steps can be summarized as follows:

- Step 1: Generation and Collection of Input Parameters and Input Data
- Step 2: Source Term Calculations
- Step 3: Dose Rate Calculations

The staff's evaluation of the methodology follows.

4.2.1 Step 1: Generation and Collection of Input Parameters and Input Data

In Step 1, the user defines the input parameters that will be used for this methodology. The relevant inputs are the loading patterns of the fuel assemblies into the cask and the relevant properties of the fuel assemblies, which include the fuel properties, the depletion parameters of the fuel, the BECTs of the fuel and any associated pressurized water reactor (PWR) NFH.

The user must determine if their fuel is within the area of applicability defined in table 2.2 of the Topical Report including allowed deviations, as discussed, and document the findings within a Qualification Report. The staff finds the area of applicability acceptable for this methodology. The staff's evaluation of each of the parameters within the area of applicability follows:

- Fuel and Fuel Type: The source term methodology described in the Topical Report is applicable to spent boiling water reactor (BWR) and PWR UO_2 fuel only. For example, it would not be applicable to advanced reactor fuel, mixed-oxide (MOX) or blended, low enriched uranium (BLEU) fuel.

BLEU fuel is created by downblending fuel from higher enriched uranium. Additional impurities, such as cobalt, which activates into Co-60, can be introduced in this process that change the source term to differ from traditional UO_2 fuel. An example of the composition of BLEU fuel with higher cobalt impurities is shown in Ref. 23.

The staff found that ORIGAMI from the SCALE 6.2.1 code system with the specified reactor libraries generated using TRITON are capable of simulating the spent fuel nuclides from BWR and PWR spent nuclear fuel (see section 4.2.2.1 of this SE for additional information).

- Burnup: The allowable fuel burnup is up to 72 GWd/MTU. The staff has accepted the use of TRITON/ORIGAMI for fuel burnup as high as 72 GWd/MTU. TRITON has the capability of simulating more detailed physics than 1-D codes that use a unit cell such as SAS2H. Per Ref. 12, the TRITON libraries supplied with SCALE 6.2.1 that are to be used as a part of the Topical Report methodology have been generated using burnup values up to 72 GWd/MTU. Therefore, the staff found that these codes are capable and acceptable for generating source terms with this maximum burnup.

- NFH for PWR Assemblies: Table 2.2 of the Topical Report limits the applicability of the methodology to burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), axial power shaping rods (APSRs), wet annular burnable absorbers (WABAs), rod cluster control assemblies (RCCAs), control element assemblies (CEAs), neutron source assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, instrument tube tie rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods.

  NFH components contain materials that become activated during normal operations. These activated materials contribute source terms that must be considered for storage in a spent fuel storage system. The primary source terms in NFH come from the activation of Co-59 impurities within structural materials (e.g., steel and Inconel), and absorber material within control rods (e.g., hafnium and silver-indium-cadmium (Ag-In-Cd)).

  For calculating an activation source, the most important parameters are: the amount of the Co-59 impurity (discussed in section 4.2.2.2 of this SE), the irradiation time, and the mass of the component. The staff does not have recent information on the design of these components as reference information is not available to the staff beyond documents published in the 1990s (Ref. 13). However, given the information in Ref. 13, the staff has determined that the mass of the NFH assumed in the Topical Report (tables 3.4, 3.7 and 3.8 of the Topical Report) is bounding for the components listed above. The codes and methodology for analyzing an activation source (discussed in section 3.5 of the Topical Report) are themselves not limited by mass as the amount of Co-60 (or Hafnium or Ag-In-Cd in control rods) can be scaled up to match the mass of the component. Therefore, the user could model more accurate masses of the inserts to more accurately calculate the amount of activation products. There is flexibility allowed within the methodology to adjust the mass if it is determined to exceed that in tables 3.4, 3.7 and 3.8 of the Topical Report. Likewise, if components are determined to have a much lower mass, it is also possible to reduce conservatism within the calculation by lowering the assumed mass of the component, however masses that deviate from tables

3.4, 3.7 and 3.8, must be documented and justified within the associated Qualification Report. If lower masses are used, the associated Qualification Report must document this restriction to allowable contents.

- Enrichment: The Topical Report allows fuel assemblies with an enrichment range from 0.5% to 5%. The staff found this acceptable based on the range of enrichment modeled within the TRITON libraries supplied with SCALE 6.2.1 that are to be used as a part of this methodology (Ref. 12).

  Section 3.6 of the Topical Report states that modeling of fuel assemblies with axial blankets and the definition of the relevant burnup and enrichment characteristics will be part of the review and approval of the CoC application that incorporates the reference to and use of the Topical Report. The staff found this acceptable as the change in source term from blanketed fuel assemblies can be accounted for within the dose rate calculations.

- Cooling Time: Minimum cooling time allowed for fuel assemblies is one year. The ORIGAMI module within SCALE 6.2.1 can determine nuclide concentrations in spent fuel immediately after discharge. However, the one-year restriction is based on 10 CFR 72.2 and 10 CFR 72.3, which establish that spent nuclear fuel that can be stored within a system licensed under Part 72 must have undergone at least one year of cooling time.

- In-Core Cycle Average Boron Concentration: The applicant restricted the in-core cycle average boron concentration from 0-2000 ppm. The W17X17 reactor library generated using TRITON that will be used with ORIGAMI in the Topical Report were generated using a boron concentration of 650ppm (Ref. 12).

  The boron concentration assumed within a depletion calculation can affect the source term due to the neutron energy spectrum shifting to a faster spectrum based on the high thermal absorption cross section of boron. This would cause an increase in actinides due to the higher absorption of neutrons by U-238. An increase in actinides results in a higher neutron source. Ref. 14 documents studies on the effect that soluble boron concentration from 0-1700 ppm has on source term. Higher boron concentration results in higher gamma and neutron source terms, however over the entire range studied, the change in gamma and neutron source terms was not significant (less than 10%). The information in section 3.1.2 of Ref. 13 shows actual boron concentrations up to 2000 ppm with cycle average boron concentrations much lower than 2000 ppm. Although outside of the studied range within Ref. 14, the staff found that the cycle average soluble boron range specified in table 2.2 of the Topical Report is acceptable based on the small change to source term demonstrated between 0-1700 ppm and actual cycle average boron concentrations being much lower than this.

- In-Core Exposure to Control Elements: Table 2.2 of the Topical Report states that there is no restriction to fuel assembly exposure to control elements or NFH listed above. Similar to the effect that boron concentration has on source term, the presence of NFH and control elements can also have an effect on source term based on these components absorbing thermal neutrons and creating a faster spectrum generally resulting in an increase in actinides relative to fuel assemblies that are not exposed to control elements or other NFH.

Oak Ridge National Laboratory (ORNL) studied the effect on source term from the presence of these components and documented their findings in Ref. 14. ORNL's studies conclude that there is an insignificant difference on gamma source term. Neutron source term increases due to the presence of these components, however for the range of expected burnup, this increase is less than 10%.

The reactor libraries generated using TRITON that will be used with ORIGAMI outlined in the source term methodology described in the Topical Report were generated assuming that there were no burnable absorbers or control components present. This cannot be changed without generating new reactor libraries that assume components are present.

The staff found it acceptable that reactor libraries that did not include the presence of these components will be used to calculate the source term. The staff found this acceptable because it would be atypical for an assembly to be exposed to components such as a BPRA or fully inserted CRA for their entire life. Not every PWR plant design uses BPRAs and not very many assemblies would be exposed to a BPRA, and they are typically only inserted for the first cycle. For BWRs, control blade positions vary throughout the life of the core and would not be fully inserted for the entire life of a single assembly. TPDs, or partially inserted control rods or blades, that could be inserted for a longer portion of an assembly's life, do not extend the entire length of the fuel. The top and bottom of the fuel assembly has a much lower source term due to the lower neutron flux during operation. Because not many assemblies would be exposed to NFH, and if exposed the exposure would be for a short duration, and because neutron source term increases were not significant, staff found the reactor libraries adequate.

- Moderator Density: Table 2.2 of the Topical Report states that the area of applicability for water density is 0.1 to 0.9 g/cm^3 for BWR fuel and 0.67 to 0.78 g/cm^3 for PWR fuel. Based on the information in sections 3.1.1 and 3.2.1 of Ref. 13, PWR moderator density range is roughly 0.65 – 0.75 g/cm^3 and BWR moderator density ranges from roughly 0.1 – 0.75 g/cm^3. A BWR will see a much higher range of density due to the change in void fraction due to boiling throughout the axial height of the reactor core.

  Moderator density is a user defined input to the ORIGAMI code. The ORIGAMI code can interpolate on moderator density if the reactor libraries were generated using multiple moderator densities. For the BWR fuel using the GE10X10-8 library, the moderator density can range from 0.1 to 0.9 g/cm^3. The W17X17 PWR fuel reactor library was generated at only one moderator density, 0.723 g/cm^3, and therefore the code does not interpolate on moderator density for PWR fuel.

  Since the applicant is using the W17x17 reactor library for PWR fuel, the ORIGAMI code will be using the density specified in the library as stated in section 2.8.4 of the Topical Report. For BWR fuel, the applicant specified in section 2.8.5 of the Topical Report that the default moderator density for BWR reactor fuel will be 0.3 g/cm^3.

  Ref. 14 contains the results of studies on the effect moderator density has on source term. This is discussed in sections 4.1.5 and 4.2.5 in Ref. 14 for PWRs and BWRs, respectively. Lower moderator density results in higher energy neutron spectrum due to less moderation and, similar to higher boron concentrations and the presence of inserts discussed earlier in this SE, the faster neutron spectrum results in additional actinides and higher neutron source. The results of the studies documented in Ref. 14 also demonstrate this phenomenon. There is a much larger variability in neutron source versus gamma source. Ref. 14 shows studies using a moderator density range from 0.6611 to 0.9996 g/cm^3 for PWR fuel and a void fraction of 0.05 to 1.0 for BWR fuel. The results of these studies show that the dose rate from PWR fuel can vary by as much as 40% over the studied range for various cooling time combinations and, despite the larger range of moderator density studies, as much as 35% for BWR fuel. Overall, lower moderator density produces a higher source term and dose rate especially from neutron dose.

The staff found the moderator density specified in the Topical Report to be a reasonable assumption for the area of applicability. Although the studies in Ref. 14 show that there can be a significant effect on neutron source term and dose rate from neutron radiation from variations in moderator density, the staff found that the effect is not so significant that it needed to be treated explicitly in the source term calculations as long as a reasonably conservative value was chosen to represent the moderator density. The W17X17 reactor library uses a reasonable average for what is seen by PWR fuel, and the moderator density of 0.3 g/cm^3 specified by the applicant for BWR fuel is conservative. Ref. 13 shows an average moderator density of 0.7 g/cm^3 for a PWR and 0.4 g/cm^3 for BWR fuel. Additionally, for a BWR, the areas with the lowest density would be toward the top of the assembly at the core exit and this area has lower burnup and less neutron dose rate than the axial center. The staff found the range of applicability for moderator density specified in the Topical Report in table 2.2 to be acceptable.

- Fuel Density: Table 2.2 of the Topical Report shows that the area of applicability for fuel density is 9.0 to 10.96 g/cm^3. Section 2.8.6 of the Topical Report states that the fuel density shall be modeled at 95% of theoretical density of 10.96 g/cm^3. This is consistent with fuel assumptions used in the TRITON models used to generate the reactor libraries per Refs. 12 and 21. Fuel density affects source term as it affects the U-238 resonance self-shielding. Studies documented in Ref. 14 show the effect of varying fuel density on source term. The density range studied was 10.26 – 10.96 g/cm^3. The results in Ref. 14 showed a negligible change in source term over this range, therefore applying the assumption of 95% of theoretical fuel density in the area of applicability allowed by the Topical Report (9.0 – 10.96 g/cm^3) would have a very small effect on dose rate.

- Specific Power: The applicant specified the range of specific power as up to 40 MW/MTU for PWR fuel and up to 30 MW/MTU for BWR fuel. The applicant states in tables 3.1 and 3.2 of the Topical Report that the specific power that will be modeled is 43.48 or 40 MW/MTU for PWR fuel and 30 MW/MTU for BWR fuel, therefore the applicability range is equal to or more conservative than the model and is acceptable.

  The TRITON calculations used to perform the reactor simulation and generate the reactor libraries used in the ORIGAMI calculations require a specific power as input. To be able to calculate discharge nuclides for a different specific power, ORIGAMI will change mid-cycle burnup accordingly. Since the average spectrum shape is not expected to change significantly based on temperature, then ORIGAMI can adjust the burnup in this way to account for changes in specific power. The staff found that the area of applicability for the specific power defined in table 2.2 of the Topical Report is broad enough to reasonably encompass PWR and BWR fuel (see sections 3.1.3 and 3.2.2 of Ref. 13), however, not so broad that the constant flux shape assumption within ORIGAMI becomes invalid.

Note 2 to table 2.2 of the Topical Report states that there are exceptions that can be made if fuel is outside this range. Since the specific power can be changed within the ORIGAMI code, source term calculations can be performed with higher or lower specific power if needed. Any restrictions to allowable fuel loadings with respect to specific power must be appropriately documented within the associated Qualification Report.

- Fuel Arrays: Table 2.2 of the Topical Report states that the methodology in the Topical Report is applicable to BWR 7x7 up to 11x11 and PWR 14x14 up to 17x17 fuel arrays. The reactor libraries that will be used with the methodology are based on a 17x17 PWR assembly array and a 10x10 BWR assembly array. Studies documented in section 6.1 of Ref. 14 analyzed various BWR fuel assemblies from 7x7 up to 10x10 and various PWR fuel assemblies from 15x15 to 17x17. The results of these studies showed that the specific assembly design has a negligible effect on dose rates. The differences in dose rates from the different fuel assembly types are likely due to the differences in mass between the different fuel assembly types. In addition, different fuel types may have different fuel hardware designs resulting in differences in fuel hardware (e.g., Co-60) source terms. However, since fuel hardware is at the top and bottom of the fuel assembly where flux is the lowest, the difference in hardware designs is expected to have a minimal impact on dose. Therefore, the staff found the 17x17 PWR and 10x10 BWR fuel assembly array types chosen by the applicant to represent all fuel assembly array types within the area of applicability acceptable because the specific assembly design has very little effect on dose rates so long as the mass of the fuel assembly is appropriately represented.

- Fuel Mass: Table 2.2 of the Topical Report states that area of applicability for fuel mass is up to 205 kg for BWR assemblies and 575 kg for PWR assemblies. Fuel mass has a very direct effect on dose rates in that an increase in fuel mass increases the source term. The Topical Report shows that the fuel mass is to be set at 469.144 or 495.485kg for PWR fuel and 188.249 or 198.516 kg for BWR fuel assemblies, respectively. Although the applicable range is outside that of the modeled assemblies, uranium mass tends to have a compensatory effect as the additional source term associated with more uranium also results in additional self-shielding, therefore the staff found the area of applicability acceptable.

  Because source term can be determined using the codes and methodology described in the Topical Report on a unit mass basis, fuel mass can be adjusted up or down. Therefore, there are no restrictions to fuel mass inherent to the source term calculation codes described in the Topical Report. Section 2.8.8 of the Topical Report states that a user could use site-specific uranium mass as long as it was adjusted in both the source term and the dose rate calculations and documented in the associated Qualification Report.

Some BWR fuel assemblies can have partial-length rods (PLRs) where the upper part of the assembly (sometimes referred to as the “vanished lattice”) has fewer fuel rods than the lower part of the assembly. The modeling of these assemblies will need to be specified within the dose rate calculation method. Typically, the PLRs are assumed to be full length, which is shown to be conservative in Ref. 14. Assumptions on how the allowable assembly mass is adjusted to account for assemblies with PLRs will need to be discussed within the dose rate calculation method if these assemblies are authorized for storage.

The staff found the area of applicability for fuel mass in table 2.2 of the Topical Report acceptable when using the fuel mass defined in tables 3.1 and 3.2 of the Topical Report. As stated in Note 2 of table 2.2., changes to the fuel mass would need to be justified and documented in the associated Qualification Report, with content restrictions, as necessary.

- Fuel Condition: Table 2.2 of the Topical Report states that the fuel condition can include undamaged, damaged, fuel debris, reconstituted, or reconfigured. It states in section 2.8.9 of the Topical Report that when representing damaged fuel and fuel debris in a dose rate calculation, the source term remains the same, however the geometry of the source changes. For example, fuel debris may be modeled as a source concentrated at the bottom of a storage system, but with the same source term as the intact fuel. The staff found that the source term methodology within the Topical Report could apply to damaged fuel and fuel debris provided that the geometry of the source term is accounted for within the dose rate calculations.

  Reconstituted fuel refers to fuel assemblies where there may have been a damaged fuel rod within the assembly early in its life and it is removed and replaced with a dummy rod to maintain structural integrity. Typically, these replacement rods are made of steel which contains Co-59 as an impurity that is activated to Co-60 when exposed to the neutron flux inside of a nuclear reactor. Because Co-60 has a strong gamma, it can have a significant contribution to dose rate. The fuel assembly would otherwise be missing the fuel rods, so the source term from spent fuel would decrease. However, the steel replacement rods are less dense than the uranium fuel rods and provide less self-shielding.

  Although there are a small number of assemblies with stainless steel replacement rods, Ref. 14 documents studies on the effect that stainless steel replacement rods have on dose rates in section 6.6 of that report and it is found to be significant. Therefore, the source term from the irradiated steel replacement rods needs to be appropriately accounted for. Section 3.2.2 of the Topical Report discusses the methodology for simulating the stainless-steel replacement rods. Similar to other activated non-fuel components, this source term will be determined using ORIGAMI by determining the Co-60 activity for a unit amount of steel based on flux factors and scaling up to the amount of steel associated with the replacement rods.

  This source term could be added to the spent fuel assembly source term, or the spent fuel assembly source term and associated self-shielding can be reduced to account for the replacement of the spent fuel rod(s). How the assemblies with stainless steel replacement rods are modeled with respect to source term and self-shielding will need to be specified within the transport method within the CoC certification application and reviewed and approved by the NRC.

- Fuel Cladding: Table 2.2 of the Topical Report states that the methodology in the Topical Report is applicable to fuel with zirconium-based cladding only. Older fuel assemblies that have stainless steel cladding can have a significant source term from the Co-59 impurities within the steel cladding and the applicant did not provide a means to handle activation products from steel cladding. The staff found restricting the methodology to zirconium based cladding appropriate.

Step 1 of the Topical Report methodology states that allowable deviations from the final safety analysis report (FSAR) model must be identified and documented in the application for a CoC that applies the Topical Report. For example, this might be if a system has a variable thickness shield for the transfer cask, or different density for the concrete in the storage system, or any allowable changes made via 10 CFR 72.48. The format and context of a Qualification Report, including in which allowable deviations must be identified and documented, are provided in appendix E and F of the Topical Report and are discussed in section 4.2 of this SE. The Qualification Report is maintained on-site. As discussed above, a dry storage system's Technical Specifications will reference the Topical Report for deriving allowable loading parameters.

The methodology defined in the Topical Report can be used to qualify any fuel/NFH that falls within the area of applicability as specified in table 2.2 of the Topical Report. Therefore, any user applying this Topical Report to their system will need to have contents within these bounds. For the reasons discussed throughout SE section 4.2.1, the staff found the bounds in the area of applicability to be adequate.

4.2.2 Step 2: Source Term Calculations

In Step 2 of the fuel qualification methodology the user performs source term calculations using the methodology outlined in section 4.0 of the Topical Report. The methodology employs the ORIGAMI module within the SCALE 6.2.1 code system (Ref. 20). It uses the reactor libraries, W17X17 and GE10X10-8, generated with the TRITON module and supplied with the SCALE code. The following sub-sections contain the staff's evaluation of the calculational methodology used for determining the source term.

4.2.2.1 Spent Fuel

To calculate the neutron and gamma source terms from spent nuclear fuel, the Topical Report states that a user will use the W17X17 and GE10X10-8 TRITON reactor libraries for PWR and BWR fuel, respectively, distributed with the SCALE 6.2.1 code, or newer, with the ORIGAMI module. The ORIGAMI code computes detailed isotopic compositions for light water reactor assemblies containing UO_2 fuel by using the ORIGEN transmutation code with pre-generated reactor libraries, for a specified assembly power distribution. ORIGEN is based on the point-depletion method and is considered an industry standard code and has been found acceptable by the staff for this purpose (Ref. 18, Ref. 15 in section 6.5.2.2).

The Topical Report states in section 3.1 that using the modules associated with newer versions of SCALE are acceptable as long as the newer codes demonstrate that for a small set of BECTs the dose rate results are within 5% of those calculated with the SCALE 6.2.1 system. The staff found this acceptable based on the rigorous testing procedures developed by ORNL for the

SCALE code system. Updates that cause a significant change from a previous version would not be incorporated unless it was well understood that the change resulted in more accurate physics. In addition, newer versions of the ORIGAMI module are not likely to have an effect on spent fuel nuclides for light-water reactor (LWR) fuel as updates to the code would likely be made to add features needed for depletion calculations of advanced reactor fuel. Newer nuclear data, such as ENDF/B-VIII, if incorporated, would likely have a larger effect, however, reactor libraries generated with the newer data would not be incorporated until ORNL was able to determine that the reactor libraries and data were appropriate.

The process for calculating the spent fuel source term using the ORIGAMI code is outlined in section 3.1 of the Topical Report. The staff found that using a single full power cycle is more conservative than accounting for the additional cooling time that would be experienced between cycles and found this assumption acceptable. Design basis fuel assumptions for inputs related to the source term are listed in table 3.1 or 3.2 of the Topical Report. Deviations from certain assumptions may be used if justified and documented in the Qualification Report. Allowable deviations are discussed in the previous section 4.2.1 of this SE.

The gamma energy range of interest is 0.45 to 3.0 MeV. This is consistent with the review guidance in section 6.5.2.3 of NUREG-2215 (Ref. 15). Therefore, the staff found this acceptable.

The user will extract the neutron, gamma and Co-60 source terms from ORIGAMI outputs into the energy group structure. The energy group structure is defined in tables 3.5 and 3.6 for gammas and neutrons, respectively. Sections 3.2.1 and 3.3 of the Topical Report state that an alternative group structure may be used. Alternate group structures will be submitted with the CoC application for review by the NRC. The group structure used within the dose rate calculations should be consistent with tables 3.5 and 3.6 of the Topical Report, or consistent with the proposed alternate group structure.

The staff performed independent calculations using the UNF-ST&DARDS (Used Nuclear Fuel – Storage, Transportation & Disposal Analysis Resource and Data System) code (Ref. 16) which demonstrated that grouping neutrons and gammas using the energy structures in tables 3.5 and 3.6 of the Topical Report produces more conservative (higher) dose rates as compared to a much finer energy group structure that more accurately models the gamma energies. The default group structure in UNF-ST&DARDS contains 999 gamma and 200 neutron groups (Ref. 17). For these reasons, the staff found the group structure used in tables 3.5 and 3.6 of the Topical Report acceptable.

4.2.2.2 Fuel Hardware

The process for calculating the source term from activation products from spent fuel hardware is discussed in section 3.2 of the Topical Report. Co-60 is the dominant contributor to dose from activation products due to the relatively high energy gamma from activation of Co-59 impurities within the stainless steel and Inconel components in the fuel hardware (e.g., top nozzle, bottom nozzle, grid spacers).

For these analyses the amount of the Co-59 impurity level must be either accurately determined or estimated using a conservative margin of error in addition to quantifying the total mass of the hardware component. The Topical Report states that the user must assume an impurity level of 1.0 g/kg impurity for Co-59 for all stainless steel and Inconel components and assumes there is no Co-59 in the zircalloy components. This impurity level is applicable for all fuel assemblies manufactured in or after 1990. Analyses performed for assemblies manufactured prior to 1990 will need to include the assumption of a higher Co-59 impurity of 2.2 g/kg for steel and 4.7 g/kg for Inconel. The staff found these Co-59 impurity levels consistent with published data (Refs. 13, 19). Although the staff does not have more recent data than what is cited on the impurity levels of Co-59 within nuclear reactor fuel and associated components, the staff is aware that there has been an effort throughout the industry to lower these impurities and therefore, based on its judgment, found that these values would be reasonably bounding for use within these source term evaluations.

If a user can justify using a lower Co-59 impurity through documented impurities for specific assemblies, these lower values can be used in a specific analysis as long as they are documented within the Qualification Report and allowable loadings are restricted to these values. This was discussed in section 3.2.2 of the Topical Report and the staff found this acceptable.

Section 3.2 of the Topical Report states that if the source term does not consider Inconel spacers then they must be excluded from the allowable contents. Table 3.1 of the Topical Report contains the data for fuel hardware. This data is consistent with the design basis assemblies, and although there are going to be differences in hardware design for different assemblies, the source term coming from fuel hardware at these locations is lower due to decreased flux. The applicant shows the factors it uses to adjust the source term in its calculations for fuel hardware in table 3.3 of the Topical Report. These factors account for reduced flux at the periphery of the core. The factors in table 3.3 are consistent with those from Ref. 19 and are determined to be appropriate based on the information in Ref. 18 and Ref. 13. The scaling factors in table 3.3 do not include an allowance for the uncertainty, which is +/-50%, however, given the top and bottom of the assembly are areas of lower flux and lower contribution to dose rate and the information in section 6.2 of Ref. 13, which suggests these factors are conservative, the staff found the values in the Topical Report acceptable for generic use.

Other fuel components such as guide tubes or instrument tubes made of non-zirconium-based cladding such as stainless steel or Inconel are not discussed within the Topical Report. Although activation of these materials could increase the Co-60 source term of an assembly, the staff found it acceptable not to account for this within the source term evaluation documented within the Topical Report because these would be on the interior of an assembly and will be shielded by the surrounding fuel rods, therefore minimizing any impact on external dose rates. In addition, the staff reviewed the information in U.S Department of Energy (DOE)/RW-0184 appendix 2A (Ref. 21) that shows that the only assembly with zirconium based cladding and stainless steel / Inconel guide tubes are the Exxon / ANF 16x16 Westinghouse PWR manufactured for Yankee Rowe, which has been shut down since 1991. For these reasons, staff found it reasonable that the source term evaluation in the Topical Report does not consider activation of non-zirconium based cladding.

4.2.2.3 Non-fuel Hardware

For PWR assemblies that will have NFH, as defined in table 2.2 of the Topical Report, section 3.4 of the Topical Report discusses how to calculate the source from these components. Similar to the fuel hardware, the main source of radiation from these components is due to Co-59 impurities in the material that is activated into Co-60. The exception is CRAs that have a different source term due to activation of absorber material such as hafnium or AgInCd. This is discussed in section 4.2.2.3.2 of this SE. The process for determining the sources from NFH is summarized in section 3.5 of the Topical Report. The staff reviewed this process and found it appropriate for determining the source term from NFH. Unless treated in a bounding way, it is expected that all NFH will have limitations on cooling and exposure time and that these limitations will become part of the Qualification Report. Assumptions related to the specific groups of components are summarized in the following sub-sections.

4.2.2.3.1 TPDs and BPRAs

The methodology uses a representative BPRA to take into account both BPRAs and wet annular burnable absorbers (WABAs), and a representative TPD to take into account TPDs and water displacement guide tube plugs, orifice rod assemblies, instrument tube tie rods (ITTRs), vibration suppressor inserts, and "components of these devices such as individual rods" as described in table 2.2 of the Topical Report. The staff has compared the assumed composition and mass to available literature in Ref. 13 and found that it is appropriately representative for these components. The masses to be activated are shown in table 3.4 of the Topical Report.

4.2.2.3.2 CRAs and APSRs

The Topical Report discusses the assumptions for modeling CRAs and APSRs in section 3.4.2 of the Topical Report. The methodology uses a representative CRA and APSR to account for all CRAs, APSRs, RCCAs and CEAs described in open literature. Unlike other NFH components, for CRAs, there is significant source from nuclides other than Co-60 due to the activation of absorber material (i.e., AgInCd, B_4C, hafnium). The applicant determined that the bounding material is AgInCd based on the activation of Ag. The staff has compared the specifications listed in tables 3.7 and 3.8 of the Topical Report to available literature in Ref. 13 and found that it is appropriately representative.

4.2.2.3.3 Neutron Source Assemblies

The Topical Report discusses the options for modeling the neutron source assemblies in section 3.4.3. The Topical Report states that the neutron source assemblies can have a gamma source from Co-60 equivalent to that of an activated BPRA due to activation of the structural materials, however, the neutron source can vary depending on the type of neutron source used. The Topical Report describes a method to calculate Co-60 from the activation of NSAs which is equivalent, including assumed masses, to the method used for a TPD combined with a BPRA.

Primary neutron source assemblies are used for the start-up of the plant and are only inserted for one cycle while secondary sources can be in the core for multiple cycles. It is not expected that there will be many NSAs at any given plant (~ 20). The staff reviewed the information related to the design of the primary and secondary NSAs from DOE/RW-0184 Vol 5 (Ref. 22). Nearly all of these NSAs have stainless steel mass similar to or bounded by the mass of the combined BPRA and TPD as described in table 3.4 of the Topical Report and therefore the staff found the assumption acceptable to account for the gamma contribution from the activation of NSAs. A few primary sources have stainless steel masses significantly higher than what Holtec states will be used in table 3.4 of the Topical Report. These are the Westinghouse primary sources and are only used for the start-up of the plant. Because they are only inserted for one cycle and will have significant time to decay before entering dry storage, the staff found that they would not have a significant contribution to dose rates. Therefore, the staff found the activation masses specified in table 3.4 for NSAs acceptable.

Depending on the neutron source used, the source can be negligible so the Topical Report provides three options when evaluating neutron source assemblies. These options require an evaluation of the specific neutron source at the site. If neutron source terms are quantified, their specific characteristics can be used within the shielding evaluations and they can become part of the qualified content described in the Qualification Report. Other NSAs have negligible source term as the half-life is short enough that the source has already decayed, such as antimony-beryllium sources. In this case, there would be no restriction on the neutron source from NSAs. Gamma source would need to be limited as discussed in the first paragraph of this section of the SE (section 4.2.2.3.3). Lastly, if no evaluation is performed, the Topical Report permits only one NSA in a basket and states that it shall be located near the center of the basket. The staff found that these options are reasonable especially considering the low number of NSAs expected for loading.

4.2.2.3.4 Determination of Co-60 from NFH

The Topical Report describes the process for determining activation sources for components described in sections 4.1.2.3.1, 4.1.2.3.2 and 4.1.2.3.3 of this SE in section 3.5 of the Topical Report. The Topical Report states that a reference amount of Co-59 will be added to the ORIGAMI calculation for the fuel assembly using an assembly burnup appropriate for the insert.

Then the amount of Co-60 will be scaled appropriately corresponding to the mass listed in tables 3.1, 3.2, 3.4, 3.7 or 3.8 of the Topical Report. Then the result will be scaled using the applicable flux factor from table 3.3 of the Topical Report (see section 4.1.2.2 of this SE).

The staff reviewed this process and found it appropriate for determining the Co-60 source from these components since Co-60 source is proportional to metal mass and neutron flux.

4.2.3 Step 3: Dose Rate Calculations

In Step 3 of the methodology the user will use the qualified source term to perform the dose rate calculations. The dose rate calculation method will be documented in the FSAR for the system that the Topical Report is applied to and must be reviewed and approved by the NRC at the time the Topical Report is applied to a CoC.

The Topical Report provides guidelines that will be used when establishing this method and an overall calculation process in Step 3 of section 4.0 of the Topical Report. That section discusses a common practice for calculating dose rates per particle and adjusting them to account for the gammas or neutrons per second per energy group for a particular source term.

Appendix B of the Topical Report contains information to be included in the FSAR/TS in accordance with the Topical Report. The staff reviewed these statements and found them to be reasonable and appropriate high-level criteria for establishing a method to calculate dose rates. Section B.1.3 states when describing the design basis fuel assembly that a homogenized mixture be used. Although this has been found acceptable, per Ref. 18, providing more detailed modeling, such as pin-by-pin assembly modeling would also be acceptable.

The Topical Report does not include a description of how secondary gammas and neutrons from sub-criticality multiplication are considered. Rather, these effects are to be accounted for within the dose rate calculations.

4.2.3.1 Acceptance Criteria

The acceptance criteria play a critical role in the Topical Report as the acceptance criteria permit staff to make a finding that a system meets 10 CFR 72.236(d) without depletion parameters and loading patterns in the Technical Specifications. The acceptance criteria will be specific dose rates at locations determined appropriate for each system. As there are many depletion permutations and loading patterns that can have equivalent dose rate, the acceptance criteria provide greater flexibility while assuring the same health and safety standards. The source terms developed using the methodology from the Topical Report can be used together with the dose rate calculation method (typically located in the FSAR and approved at the time of CoC application) to show that the system meets the acceptance criteria (dose rates). The applicant must also discuss how dose rate values and locations remain appropriate for representing the dose rate performance of the entire system when there are allowable changes to the system.

The acceptance criteria are proposed in the CoC application that requests use of the Topical Report. Staff reviews the acceptance criteria to confirm that they are appropriate for each system and that they represent enough locations and areas of interest that the staff can have reasonable assurance that a system that meets the acceptance criteria will similarly meet 10 CFR 72.236(d). The Topical Report includes some high-level criteria in section 2.6 and B.2 to consider when proposing the acceptance criteria. The staff found these high-level criteria appropriate because the criteria state that the number and locations shall be selected to be representative of the contents of the cask to include all assemblies, NFH, and locations where dose rate is expected to be highest but does not necessarily include localized locations such as vents where there may be high dose rates but smaller areas.

4.2.3.2 <u>Representative Source in Establishing Fuel Qualification</u>

The source term methodology outlined in the Topical Report is suitable for determining acceptability of as-loaded systems with specific loading patterns, however, it can also be used to define generic loading patterns. This may be for casks with a uniform loading and multiple BECT combinations, or patterns with multiple zones (or regions). The Topical Report allows for two options when analyzing allowable loadings with multiple BECTs.

For Option 1, each BECT combination is analyzed to ensure that it meets the acceptance criteria. The staff found Option 1 acceptable because it analyzes every BECT combination. However, if there are multiple BECTs allowed in multiple regions, the number of calculations needed to represent each combination can be extensive, therefore, the Topical Report defines an alternative option that provides more conservatism but reduces the number of total calculations. For Option 2, each allowable BECT combination is analyzed within a single loading zone/region to determine the one that gives the highest dose at every dose location. Once this is determined then dose contribution from all zones is analyzed together using the BECT combination that gave the highest dose rate at each location. Because this is conservative as compared to Option 1, the staff found it acceptable.

4.2.3.3 <u>FSAR Evaluations Representative Source</u>

The Topical Report provides a means to use a representative source term for demonstrating compliance with 10 CFR 72.236(d). Applicants propose the source term for the specific system as part of the CoC application. Considerations for developing the source term are discussed in section B.4 of the Topical Rep

The applicant should demonstrate that a source higher in gamma source (lower burnup with lower cooling time) as well as a source that’s higher in neutron source (higher burnup with higher cooling time) would meet acceptance criteria. A combined source could be used where higher gamma is used in the periphery where there is less shielding from basket contents and a higher neutron source is used in the central basket locations. The appropriateness of the source may depend on the shielding materials used and locations that contribute more to dose and could be different for dose rates calculated around the transfer cask versus dose rates calculated around the overpack. Accident conditions are usually limited by a loss of neutron shielding within the transfer cask. A source term that has a higher neutron contribution (high burnup, longer cooling time) may be more appropriate for this analysis. Applicants would need to provide justification at the time of application of the Topical Report to a CoC, justifying that the source term selected is appropriate for the specific configuration (e.g., transfer cask, storage overpack) and analysis conditions (e.g., normal versus accident conditions).

Representative source terms should be selected to achieve a dose rate at or slightly exceeding the acceptance criteria to demonstrate compliance with the regulatory limits in 10 CFR 72.104 and 72.106. This would provide reasonable assurance that a different source term, generated using the methodology in the Topical Report, that meets the dose rate acceptance criteria would also meet dose limits in 10 CFR 72.104 and 72.106.

In a vacuum, any source term with an equivalent dose at or near the cask surface, should also have an equivalent dose at the controlled area boundary as dose is only reduced inversely proportional to the square of the distance (i.e., $1/R^2$ rule) and not by attenuation. However, as air is incorporated in the models, as well as incorporating arrays of varying size and shape, gammas and neutrons of different energies would be attenuated differently, as well as experience different scattering from the air above the cask (i.e., skyshine), such that two sources that give the exact same dose at the surface at or near the surface, may actually have different doses at a distance where limits in 10 CFR 72.104 and 72.106 are specified.

Research by ORNL documented in section 5.2 in Ref. 14 shows the dose contribution from various source term components in terms of distance. This research shows that for large distances from an example storage array for a concrete storage overpack, the dose contribution from secondary gammas (i.e., gammas created from n-γ reactions) increases. This was especially true for the fuel with longer cooling times where the primary gamma source had more significant decay. This is because the secondary gammas are generated at high energies and can be generated inside cask shielding materials that do not have an opportunity to be attenuated, increasing the probability of contributing to dose at very large distances.

10 CFR 72.106(b) requires that the controlled area boundary be at least 100 meters from the spent fuel storage system. Historically, CoCs for general licensees have not required distances larger than 300-400 meters to meet the dose rate requirements in 10 CFR 72.104(a). However, this study shows that if larger distances are needed, then ensuring the representative source has a high neutron contribution (i.e., higher burnup fuel) and incorporating dose from secondary gammas is important. For lower cooling times and closer distances, the largest contributor to dose is Co-60 therefore the representative source should include activated fuel and NFH within the representative source term calculations.

4.3 Qualification Report

Once the methodology for determining allowable contents has been executed, the results must be documented in a "Qualification Report," in accordance with the conditions of the Topical Report. The Qualification Report more specifically defines the allowable contents that can be loaded into a system. It also references the analyses performed, which demonstrate that the methodology outlined in the Topical Report was appropriately implemented. The Qualification Report will be used by the general licensees to document allowable contents, but also by NRC inspectors to verify that the allowable contents are appropriate consistent with the terms in the Topical Report.

As stated in section 2.3 of the Topical Report, a “Qualification Report” can be generic in nature, qualifying a larger range of contents for a large number of casks, or specific reports that address the specific contents for a limited number of casks for a single site. The Qualification Report does not require NRC review and approval (i.e., via a CoC amendment), however, it shall be made available to the NRC upon request to support inspection activities. The format and contents of the Qualification Report as contemplated in the Topical Report are shown in appendix E. The staff reviewed this report and found that it contains the necessary information in an appropriate format to support users and NRC inspection activities.

5.0 LIMITATIONS AND CONDITIONS

The Topical Report evaluated in this SE establishes a methodology to technically evaluate and qualify candidate loading patterns that satisfy given dose rate limits. Other loading restrictions and requirements beyond those discussed in this SE and the Topical Report may exist, such as those related to decay heat or criticality safety requirements.

The Topical Report provides a means for a CoC applicant to demonstrate compliance with 10 CFR 72.236(a) and (d). The following limitations and conditions exist if an applicant or licensee uses the methodology of the Topical Report:

- The source term methodology defined in the Topical Report is applicable to spent BWR and PWR UO_2 fuel only. For example, it is not applicable to advanced reactor fuel, or MOX or BLEU fuel.
- Resultant parameters describing allowable fuel will be included in the “Qualification Report” in the format described in appendix E and F of the Topical Report. The “Qualification Report” must be made available to NRC to support inspection activities.
- A dry storage system must include the Topical Report as a condition in the Technical Specifications. The NRC staff recommend doing so in the administrative controls section.
- The contents for the dry storage system for which the Topical Report is to be applied must be within the area of applicability in table 2.2. Deviations in specific parameters discussed in section 4.2.1 of this SER may be allowed as long as they are appropriately justified and documented in the “Qualification Report.”
- All NFH are expected to have limitations on cooling time and exposure time within the Qualification Report unless it can be demonstrated that these analysis assumptions are treated in a bounding way.
- Acceptance criteria, in terms of dose rates, must be established and added as a condition of the CoC to the Technical Specifications using the guidelines in section 2.6 of the Topical Report for the dry storage system (transfer cask and storage system). In the CoC application incorporating use of the Topical Report, the applicant must also discuss how dose rate values and locations remain appropriate for representing the dose rate performance of the entire system when there are allowable changes to the system.
    - One of the two options discussed in section 2.6 for qualifying the allowable contents must be used.
- The CoC application incorporating use of the Topical Report must include a method for performing dose rate calculations (i.e., the Radiation Transport Calculation Methodology in the Topical Report).
- The CoC application incorporating use of the Topical Report must also include a demonstration that the storage system meets the acceptance criteria and the regulatory

limits in 10 CFR 72.104 and 10 CFR 72.106. A representative source may be used for this demonstration to calculate doses per the guidelines in section B.4 of the Topical Report.

- The dose rate calculation must use fuel assembly information from either table 3.1 or 3.2 of the Topical Report, and the chosen assembly must be the same assembly used for source term evaluations as outlined in the Topical Report.
    - Site specific characteristics that can be justified can be specified and limited in the Qualification Report as appropriate. This includes mass of inserts, Co-59 impurity for fuel and NFH, fuel mass, and specific power. For fuel mass, any changes to table 3.1 or 3.2 Topical Report masses must also be made to the transport method so the self-shielding characteristics are consistent.
- Mass of NFH shall not be considered in the dose rate calculation as additional shielding.
- The Topical Report does not describe how secondary gammas from (n,γ) reactions or neutrons from subcritical multiplication will be incorporated and this must be described in the dose rate calculation submitted with the CoC application applying the Topical Report.
- The source term for damaged fuel can be derived from the methodology in the Topical Report but the altered source geometry must be accounted for within calculations supporting qualification and CoC certification calculations. Assumptions regarding assemblies with stainless steel replacement rods shall also be described in the CoC application with respect to whether UO_2 rods are replaced (removing SNF source term and also self-shielding) or if Co-60 is added.
- The CoC application for incorporating the Topical Report shall discuss how blanketed assemblies are modeled within the dose rate calculations if these assemblies are part of the authorized contents.
- The energy group structure used within the dose rate calculations should be consistent with tables 3.5 and 3.6 of the Topical Report. If an alternate group structure is proposed for the source term evaluations, it shall be consistent with that used within dose rate calculations.
- If assemblies with partial-length rods are part of the authorized contents, the application for the CoC applying the Topical Report shall discuss how partial-length rods are modeled within the dose rate calculations and justify the allowable mass of these assemblies if they are modeled as full length.

6.0 CONCLUSIONS

The staff found that the Topical Report outlines a methodology that if followed provides a means for demonstrating compliance with 10 CFR Part 72.236(a) and (d). The radiological fuel qualification methodology is acceptable for use with dry storage systems that meet the area of applicability, and is subject to the limitations and conditions outlined in section 5 of this SE. The methodology permits CoC applicants and licensees to consider more realistic source terms for actual cask loadings and depletion parameters. Additionally, it provides efficiency for applicants, licensees, and the NRC while still maintaining safety, assuring compliance with the requirements for content specifications in 10 CFR 72.236(a) and the shielding design in 72.236(d).

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