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Topical Report on the Radiological Fuel Qualification Methodology for Dry Storage Systems

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TABLE OF CONTENTS

SUMMARY OF REVISION

Revision 0: Original Issue.

Revision 1: Document updated in response to NRC comments. All changes tracked with revision bars.

Revision 2: Document updated in response to NRC RAIs. All revision bars from Revision 1 were removed All new changes tracked with revision bars, except for Appendix E and F that were newly added to the document.

Revision 3: Document updated in response to NRC comments. Also, additional editorial changes throughout the document. All revision bars from Revision 2 were removed. All new changes tracked with revision bars, except for Appendix C which has been replaced in its entirety and no tracked changes are shown.

1.0 INTRODUCTION

For Dry Storage Systems, 10 CFR 72.236(a) requires a definition of the contents that is qualified to be loaded. The definition of the contents mainly consists of assembly type and condition, and limits on decay heat, and burnup, enrichment, and cooling time (BECT). It also includes specification of non-fuel hardware (NFH), including type, permissible location, and burnup and cooling time. Meeting the requirement of 10 CFR 72.236(a) also supports compliance with 10 CFR 72.236(d), to show that the design is capable of meeting normal and accident dose limits (10 CFR 72.104 and 10 CFR 72.106). The decay heat of the assemblies, and the corresponding limits, are also part of the requirements, and while they are not the subject of this TR, they are an important aspect and part of the motivation for this TR. Hence, they are included in the following discussion.

To ensure that applicable temperature limits are met, limits on the decay heat values of the assemblies must be implemented. In the early days of Dry Storage, such limits were identical for each location in the basket of a spent fuel storage cask (uniform loading). However, to optimize the cask loading from both a thermal and dose perspective, more and more sophisticated decay heat limit distributions (thermal loading patterns) within the baskets were developed over time. The culmination of this are thermal loading patterns where limits are defined almost on a cell-by-cell basis. This may be needed to efficiently empty the inventory of an entire spent fuel pool, with its large range of assembly decay heat values, into dry storage systems.

Given the importance of the thermal efficiency, the burnup, enrichment, and cooling time limits must be selected so that they do not result in an additional restriction, unless necessary from a radiological perspective. Expressed differently, the burnup, enrichment and cooling time limits for a given basket cell should ideally correspond to an assembly decay heat equal to or slightly greater than the decay heat limit for that cell.

While this sounds simple as a principal guide, it creates significant complications in its implementation. This is due to the fact that there is no easy and direct relation between the decay heat and the burnup, enrichment and cooling time of an assembly. Each decay heat value corresponds to an unlimited number of combinations of these parameters, and the combinations related to a single decay heat load value can be very diverse from a radiological perspective. For example, a combination of higher burnup and long cooling time can have the same decay heat as an assembly with short cooling time but much lower burnup, but these two conditions would be very different from a radiological perspective. This conundrum makes an efficient specification of burnup, enrichment, and cooling time limits in the Safety Analysis Report (FSAR), the corresponding Certificate of Compliance (CoC) or Technical Specification (TS) of a system extremely difficult. Two options to approach this, together with their advantages and disadvantages, are as follows:

1) Provide a small set of BECTs that would bound all decay heat load values for all assemblies.
    a) That approach would be easy from an implementation perspective.
    b) However, since dose rates presented in the FSAR are to be calculated using the limiting contents, it would result in excessive dose rates presented there. It would therefore NOT give a correct indication of the dose rates that would be expected for a loaded system. This results in an incorrect

characterization of the radiological performance of the system and does not provide the radiation protection departments at the licensee's site with any meaningful information.

2) Provide extensive lists, in the form of tables of BECTs, or coefficients of equations to calculate BECTs, closely aligned with or informed by the thermal patterns.

   a) This results in a significant burden on all parties involved. The FSAR and TS needs to be updated with a significant amount of information, creating effort on the side of the applicant to develop and generate the information and maintain it for the life of the document, and for the NRC to review and approve this information. On the licensee's side, it creates a large effort to implement the limits into the site-specific procedures, and to maintain them over time. The information may then also need to be updated with any change to the decay heat patterns.

   b) Dose rates would still be overestimated, and most likely by a significant amount. This is because it would be necessary to use the worst BECT for each location in a basket to calculate dose rates, and such condition would still be far away from any realistic BECT distribution. Hence dose rates in the FSAR would still not be representative.

   c) Nevertheless, given the comparatively loose connection between BECTs and decay heat values, there could still be assemblies that, based on their operation history, are below the decay heat limit but do not pass the corresponding BECT limits.

   d) Overall, this approach requires substantially more effort than the first option but provides comparatively little if any advantages.

This Topical Report (TR) provides an alternative to these two options to satisfy the regulatory requirement in 10 CFR 72.236(a), and hence also 10 CFR 72.236(d), where the specific contents can be defined in separate qualification reports that are prepared and maintained outside of the FSAR and CoC. For that, limiting dose rates are specified in the FSAR/CoC/TS instead of specifying BECTs, and separate qualification reports then establish the BECTs that assure these dose rate limits are met. Advantages of this approach, for the parties involved, are as follows:

1) BECT limits still have to be generated, but they are no longer presented in the FSAR/TS. This reduces the effort on the certificate holder's side significantly.

2) NRC does not need to approve the complex BECT derivations, only the dose rate limits, which are more directly linked to radiation safety. While the qualification reports are not submitted to NRC for review and approval, they will be available for inspection.

3) Licensees may be able to utilize a simplified set of BECT limits more specifically tailored to the fuel they need to load.

Finally, from a safety perspective, the limits in the FSAR or TS, being dose rates, are more closely linked to safety than the BECTs used until now.

This document outlines all requirements that need to be satisfied to apply this approach. Deviations from the requirements outlined here are not acceptable, unless specifically mentioned and discussed here. For this, the following terminology is used throughout this report:

- "shall" denotes a requirement that must be satisfied.
- "should" also denotes a requirement, but alternatives are permitted. Only the alternatives discussed are permitted, and the discussions may include criteria that must be satisfied for the alternative to be acceptable.

Throughout this document, two dry storage dockets, HI-STORM 100 [1] and HI-STORM FW [2], are frequently referenced. This is done for simplification, and since these are two of the predominant storage dockets. However, this is not meant to imply that this TR is limited to being applied to these dockets only, it can be applied to any storage docket, as long as the license amendment that would be submitted to include it in a docket addresses all requirement, such as those presented in Appendix B.

2.0 OVERVIEW OF THE APPROACH

This topical report defines the overall framework of defining and qualifying content for a dry storage system. The framework consists of several components as follows:

- The technical methodology to perform source term calculations for spent fuel and non-fuel hardware. This methodology is defined in this report, and it is essentially the same as that currently defined in the FSARs for various storage systems. Since it is common to various FSARs, it is defined here to avoid duplication of the approval process.
- The technical methodology to perform radiation transport calculations, i.e., to calculate dose rates for a given system and a given content. This is defined in the FSARs for the storage systems. Since it includes modeling details for the respective systems described in the FSAR, and hence is different for each system in that respect, this is not repeated here in order to avoid duplication of the many technical details. This part of the framework will be reviewed and approved as part of the process that includes the reference to this TR in each FSAR/TS. For each system, this technical methodology is also expected to be identical to the methodology that is already presented in each FSAR. Note that the specification of this methodology in the FSAR may limit aspects of the method that can be changed under 72.48. To assure consistency, Appendix B outlines the principal requirements that this technical methodology needs to fulfil in order to be acceptable as part of the process to define content. Appendix C contains an example of the subsection that may be added to an existing FSAR to meet the requirements in Appendix B.
- The acceptance criteria, which are dose rate limits at defined locations on the storage system. Since the locations and the limits are specific to each system, they are also defined in the respective FSAR, together with the methodology to calculate dose rates, not in this TR. Nevertheless, this TR contains guidance on the development of those criteria. The criteria would also become part of the TS, so they can only be changed through a license amendment application. Examples are also included in Appendix C.

- Qualification reports that finally define acceptable content, based on the methodologies and acceptance criteria discussed above. Appendix E contains an example of such a qualification report with a format and content that shall be followed for every actual qualification report. Appendix F shows the template to be used for this report. Additional examples of qualified content are included in Appendix A.

See Table 2.1 for a brief summary of these different aspects.

The following subsections contain additional clarifications on selected aspects of the approach.

2.1 FSAR vs. Qualification Report

A given FSAR/TS may already contain previously established BECT limits to satisfy 10CFR72.236(a) for some given conditions. When updating an FSAR / TS to allow the use of this TR, these could either be retained, or relocated to a qualification report. Relocating them would make for a more consistent approach. However, if these are already heavily referenced in the licensees' documents, it may be easier to retain them in the FSAR/TS.

2.2 Information in the Dry Storage Cask System FSAR/TS

To make the method generically applicable to different storage systems, the modeling and design details of the system and the details of the radiation transport analyses to calculate dose rates are not included and discussed in this report. They remain in the corresponding FSAR for each system.

The FSAR contains the descriptions of the systems for which the contents are to be qualified. This includes drawings, relevant design details, and descriptions of calculational models. Important in this respect is the level of detail that needs to be modeled for the calculations to be able to be used for the qualification. Also important is the specification of parameters that are considered inputs, such as material thicknesses of material types and densities, that can be changed (under the purview of 10CFR72.48) when performing the qualification. Part of this modeling description are also the dose points considered important for any given system.

The FSAR (or TS) then specifies the dose rate limits for the selected dose points. This provides the principal limits that the method uses to qualify approved contents. Note that a licensee using the system may elect to use lower dose rate limits to define contents for a specific site. But dose rate limits higher than those specified in the FSAR/TS are not acceptable.

The area of applicability of this TR is discussed in Section 2.8, with details provided in Table 2.2. The FSAR (or TS) may specify additional restrictions, i.e. may limit the applicability to a narrower range of certain parameters than those listed in Table 2.2.

Appendix B contains the principal requirements and guidelines for the information that needs to be defined in the FSAR, with an example in Appendix C. As stated before, the FSAR sections involved in defining

these are reviewed and approved in the context of adding the permission to use this TR for defining acceptable content.

2.3 Qualification Reports

The evaluations and analyses needed to demonstrate that given sets of content meet the acceptance criteria are documented in qualification reports. These reports define the contents to be qualified, the system that the contents are to be qualified for, and document the evaluations. They reference this Topical Report for the methodology and the FSAR for details located there.

Qualification reports may be generic in nature, qualifying a range of contents for a larger number of sites where a cask system will be deployed, or may be site-specific, just addressing the specific contents for casks at a single site. The qualification reports do not require NRC review and approval.

Appendix A contains three examples of the analyses that would be performed using the methodology. These are to be used as guidance for the implementation/qualification reports that define the allowable contents. Appendix E contains a principal example of such a report for a selected storage system and content.

2.4 Design Basis Assemblies

It would be certainly possible to model any fuel assemblies accurately in both the source term and the radiation transport analyses. However, such an approach would be highly impractical, not only because it would require an exorbitant number of analyses, but also because previous analyses have shown that the impact of the assembly type, when used consistently in source term and radiation transport analyses, is not that significant. This is due to competing effects of the assembly mass in source term and radiation transport analyses, where a larger mass increases the source term, but also the radiation self-shielding. Hence it is more practical to establish design basis assemblies that are to be used in both source term and radiation transport calculations, to be used for all analyses regardless of the actual assembly type.

The design basis fuel assemblies taken from [1] and [2] and specified in Tables 3.1 and 3.2 may be used in the analyses. For historic reasons, the two FSAR's ([1] and [2]) used different design basis fuel assemblies, and both are acceptable to be selected. However, source term calculations and radiation transport calculations that are combined to calculate dose rates for comparison with the dose rate limits shall both use the same design basis fuel. This is necessary so the self-shielding of the fuel assembly matches the source strength, and both depend on the mass of the fuel, which is slightly different between the assemblies in the tables. For both [1] and [2], the design basis assemblies are the assemblies with the higher or highest fuel mass and are used to bound all other assemblies qualified in the respective FSAR. This approach is supported by [9], where studies are presented on the importance of various fuel parameters for dose evaluations. The studies conclude that assemblies with higher fuel mass reasonably bound those with a lower mass. This is the result of the competing effects of source strength and self-shielding being a function of the fuel mass already discussed before, hence not unexpected. Using just one or two design basis assemblies for the entire range of assemblies to be qualified is highly advantageous since it simplifies the qualification effort quite substantially. Having two design basis assemblies for either BWR and PWR is advantageous in the current situation for qualification of assemblies for the storage systems in [1] and [2],

since these FSARs already used the different design basis fuel in the development of the radiation transport models, hence these models can be used directly without any modifications.

The design basis assemblies in Table 3.1 and 3.2 can be used without any further justification. A different design basis fuel can also be used, when defined in the FSAR that qualifies the radiation transport analyses for a given system. This aspect has then to be specifically reviewed and approved as part of the license amendment for the corresponding CoC to include the reference to this TR.

2.5 Loading Patterns

Inputs to the approach are candidate loading patterns for given casks and baskets, i.e., the fuel assembly types, and limits of burnup, enrichment and cooling times, for each cell in a candidate cask, including specifications of any NFH to be qualified for the cell. These could be generic in nature, i.e. to define patterns useable at various sites for the cask or basket, or could come out of the evaluation of pool inventories for a specific site. However, the development of those patterns is not part of this report and therefore not discussed here. In principle, a pattern could be completely unique, in the sense that every cell in a basket has different limits. For the fuel, the limits could be specified in the form of one or more limiting sets of burnup, enrichment, and cooling times for each basket cell, or in the form of equations that allow the calculations of the limits. For burnups, these will be upper limits, while for enrichments and cooling times these will be lower limits. For NFH, it would include the type, maximum burnup and minimum cooling time. While only a single NFH can be present in any fuel assembly, more than one NFH type may be qualified for a given location in the cask to provide flexibility for loading a larger range of casks. Limits or sets of limits may be applicable to individual cells, groups of cells with the same content limitations (in the following called regions), or the entire cask or basket. Appendix A of this TR provides some hypothetical sets of such limits for a given basket in Tables A.1 and A.3, with regions within the basket specified in Figures A.1 and A.2.

2.6 Acceptance Criteria

The principal acceptance criteria used to qualify fuel assemblies are dose rates around the casks.

1) Storage systems often consist of the storage cask and a transfer cask. Since these typically have different shielding performance, separate dose rate limits shall be defined for each of these.
2) The number and location of dose points will be selected in the FSAR to reasonably represent the contribution of all assemblies, including NFH, in a cask or canister.

Number, location and specific dose rate limits are to be defined in the license amendment request that incorporates this TR into the respective FSAR and TS. These are necessarily specific to the design qualified in the respective FSAR. They are therefore reviewed and approved as part of that license amendment, not as part of this TR. However, for the locations to be consistent with the purpose of defining cask content, the following aspects must be considered when selecting those:

- Dose locations must be selected to be on or close to the surface of the casks, so the dose rates will be representative of the impact of individual assemblies, not just the average assembly.

- Dose points must include areas of the surface/feature where highest dose rates are expected.
    - For example, for a vertical above-ground system, this would include dose locations on the side of the cask (where dose rates are more dominated by the contribution from assemblies on the periphery of the basket), and on the top of the cask lid (where dose rates are more dominated by the contributions in the center of the basket).
- If any NFH is expected to contribute significantly to the dose rates in certain areas, dose locations in those areas should be included
    - For example, for a vertical above-ground overpack containing control rod assemblies (CRAs), side surface points should include points where the activated portions of these components are located, at an axial height where the highest contribution to the side surface dose rates is expected
- Dose point locations should include those locations that are expected to contribute significantly to off-site dose and to occupational exposures. Different orientations of transfer cask and overpack during different stages of operations ((un)loading, transfer, storage) and for accident conditions need to be considered in that respect.
- Dose points need to be sufficient in number to represent the defined content of the cask.
    - For uniform loading or symmetric loading conditions (e.g., quadrant, octant), the symmetry may allow a smaller number of dose locations than that needed for a completely heterogeneous loading.
- Dose points shall not be placed at geometric discontinuities such as the inlet of outlet vents of a concrete overpack. In these locations, dose rates are more a function of the design than of the content, and are therefore not suitable to serve as limits to define the cask content.

In the qualification report, a set of dose calculations needs to be documented to demonstrate that the content to be qualified meets the dose rates at all defined locations. The extent of this set depends on the specification of the content to be approved as follows

The fuel content shall be defined in tables such as those shown for the examples in Appendix A, i.e. combinations of maximum burnup, minimum enrichments and minimum cooling times for specified basket locations. If the relationships are also specified as an equation, then at least 10 equidistant burnups that cover the range of content to be qualified need to be considered and listed in such a table with the corresponding enrichment and cooling time.

Two options can be used to qualify the tables, i.e. to show that the fuel specified by the tables meet the acceptance criteria, i.e. the dose rate limit. The first option represents a single step process and is more suitable for uniform loading patterns, or loading patterns with a small number of regions. The second option has a two step process and is more suitable for complex loading patterns, e.g. patterns with multiple regions with different fuel specification. Both options are defined below, and either option 1 or option 2 must be used.

- Option 1 - Evaluate all acceptable combination of BECTs in the pattern.
    - As an illustration, the BECTs listed in Table A.1 in Appendix A are used

        - There are three different patterns, each with 2 regions in the cask or basket, and for each region, there are 14 different BECTs. Hence Dose Rates for a total of, 3 * 14 ^ 2 = 588 configurations need to be evaluated. The maximum of these 588 combinations for each dose location must then be shown to be at or below the applicable dose rate limit.
    - The advantage of this approach is that there is only a single direct step.
    - The disadvantage is that the number of dose evaluations may become unmanageably large very quickly for larger number of regions. For example, for the 5 regions in Table A.3, each with 14 BECTs, all combinations would require a total of 14^5=537,824 dose calculations. Hence this option is only feasible for uniform or less complex patterns.
- Option 2 - The BECTs that will result in the highest dose rates are established first, and then those are combined into the final dose analyses.
    - In some cases, it may be obvious which BECTs will result in higher dose rates. For example, the assemblies in the inner locations of a basket provide predominantly neutron dose to the outside, since the gamma dose is shielded significantly by the outer assemblies, and since neutron dose is a strong function of burnup, but not of cooling times, the highest contribution would be from the BECTs with the maximum burnup. However, for other locations this may not be that clear. Gamma contributions may dominate from the assemblies on the periphery of the basket, and the gamma contribution depends strongly on both burnup and cooling time. So, whether a lower burnup with a very short cooling time provides higher dose than a higher burnup with a slightly higher cooling time is not immediately clear. Even a comparison of the gamma spectra may not be conclusive, since different gamma energy ranges have a different attenuation in the typical shielding materials. Then there may be assemblies on the inside of the basket but close to the periphery, where it also is not clear whether they more contribute through neutrons (highest burnup), or through gamma (potentially lower cooling time). Further, the selection of the bounding BECTs would depend on the dose location, since the contribution of the dose from a certain assembly location depends significantly on this. It is therefore necessary to apply a 2 step approach, where the first step identifies the bounding BECTs, and the second step then determines the dose rates.
    - Step 1: Separately for each region in the pattern, evaluate all applicable BECTs and for each dose location determine the BECT that results in the maximum dose rate from that region.
    - Step 2: For each dose location, perform a calculation with the bounding BECTs for that location determined in Step 1.
    - As an illustration, the BECTs from Table A.3 are used.
        - For this example, it is assumed there are 10 relevant dose locations.
        - For the 5 regions and 14 BECTs per region, 5 * 14 = 70 dose evaluations are performed.
        - The results are evaluated and the BECTs are determined that results in the highest dose rate. These are 5 * 10 BECTs.
        - Now one dose evaluation is performed for each dose location, for each region using the BECTs that result in the maximum dose from that region. In this case, this requires 10 dose evaluations.

        - In total, 80 dose evaluations are needed, much less than what be needed for Option 1 for this example.
    - Advantage: An overall limited number of dose evaluations.
    - Disadvantage: A two step process.

The qualification report shall only show the maximum dose rate for each dose location to show that is at or below the applicable limit. But for each dose location, it shall identify the BECT in each region that corresponds to this maximum dose rate.

For NFH, requirements also need to be specified, in the form of maximum burnup values as a function of cooling time. These could be specified as a single pair of limits for the entire cask, or may differ between regions, and several combinations may be set for a region. If a single pair of limits is specified for each cell, their contribution may be directly included in the calculations for the fuel discussed above. If more than one set of limits is applicable to any cell, calculations of the maximum contribution would need to be performed separately from those for the fuel, using one of the same approach options discussed above, and the maximum dose results are then added to those from the fuel before comparing with dose limits.

In all cases, for both fuel and NFH, note that different bounding conditions may be applicable to different dose locations.

2.7 Other Content Restrictions

This Topical Report establishes the principal Methodology to technically evaluate and qualify candidate loading patterns that satisfy given dose rate limits. Other restrictions or requirements may exist, for example decay heat limits, as specified in the FSAR or separate documents. None of these other restrictions are considered by the methodology described in this TR, and the conclusion that an assembly with certain burnup, enrichment and cooling time combination meets the dose rate requirements does not imply that it meets any other requirements such as heat load and temperature limits, and vice versa.

2.8 Area of Applicability

This topical report is applicable to all US PWR and BWR fuel assemblies that meet the requirements of the area of applicability summarized in Table 2.2. The table also specifies the basis for each parameter. In most cases, the parameter or range of parameters in the table are fixed, i.e. the TR can only be used if the parameters in the analyses match the parameter or parameter range in the table. However, for selected parameters, as clarified in the notes to the table, parameters outside the specified range can be acceptable under certain conditions. These conditions are then specified in the corresponding section referenced in the table.

For its use for a specific storage system, this report needs to be referenced in the respective CoC/TS of the system. The FSAR/TS that the TR is referenced in may specify additional restriction, i.e. only allow a subset of fuel characterized in Table 2.2.

Some of the requirements in Table 2.2 have a simple basis, while others require additional considerations and discussions. The simple bases are listed below, followed by separate subsections for the parameters that need additional considerations.

- Only UO_2-based fuel can be qualified through the method defined here. That does not introduce any significant restriction since MOX fuel is currently not utilized in US plants. However, if they ever would be used, they would be excluded from being qualified through this TR.
- Only zirconium cladded fuel can be qualified through the method defined here. This also does not pose any significant restrictions, since the vast majority of fuel has zirconium-based cladding.
- Burnup and enrichment ranges are defined by the limits in the predefined libraries in the source term code (see Section 3).
- Fuel types cover all types used in US BWR and PWR plants
- The cooling time limit for spent fuel is taken from 10CFR72.

Other fuel operation parameters are considered to be of low importance as discussed in [9]. Nevertheless, acceptable parameters or parameter ranges are defined in Table 2.2, to clarify the limits of the use of this TR. In practicality, these parameters and parameter ranges are not expected to limit the use of this TR for standard US fuel assemblies.

The following subsections discuss the basis for other parameters. If necessary, these discussions also provide the basis for the value or values listed in Section 3 that should be used in the analyses.

2.8.1 Soluble Boron (PWR)

This range is based on information from [10]. It is noted that the cycle-average soluble boron level for PWR is typically well below the maximum. However, under certain circumstances, assemblies may only be irradiated for part of a cycle, and hence be exposed to a higher soluble boron level.

Since the impact of the soluble boron on dose rates is minor, the TRITON libraries from [8] shall be used with their respective soluble boron specification.

2.8.2 Exposure to NFH (PWR)

During in-core operation, fuel assemblies may have NFH inserted into the guide tubes. This may locally and/or temporarily reduce the amount of water in the assembly, and/or provide additional absorption of thermal neutrons, both of which would harden the spectrum, and hence potentially affect the source terms determined for the assembly. This is different from the effect of NFH during the storage operation, which is concerned about the dose contribution from the NFH themselves. The different types of NFH are discussed separately below.

- TPDs. These are only present near the top of the assembly, typically not reaching into the active region. The effect of these on the source term of the assembly is therefore considered negligible.

- Burnable poison rods. Their effect was evaluated in [9], and also found to be very low. Additionally, these would only be present in a fraction of the assemblies, and only for a limited time, typically 1 irradiation cycle, for an assembly. Overall, their impact on the source terms can therefore be neglected.
- Control Rod Assemblies. These would have a stronger neutron absorption than the burnable poison rods discussed above. However, the majority of those are for shutdown operation, and never be inserted into the active region during full-power operation, and those used for power control would only be marginally inserted into the active region, so it does not negatively affect the plant efficiency and stability [10]. The allowed insertion is also tightly controlled by the plant operating procedures. Hence these can also be considered not significant enough to be modeled for the source term operation.
- Axial Power Shaping Rods. Their neutron absorption material has less of an absorption capability than that of the control rods assemblies, but they are present in the active region. However, they are only present in a small number of assemblies in the core, and only for a limited time, hence they are considered similar in the effect to the burnable poison rods.

Overall, it is therefore reasonable to neglect the effect of the NFH for the in-core-operation, and hence fuel assemblies irradiated with any type of NFH are permitted for storage and can be qualified through this TR.

2.8.3 Exposure to Control Components (BWR)

Similar to the discussion on control assemblies for PWR assemblies in the previous subsection, BWR control components are used for shutdown and power control, with insertion limited with respect to duration and insertion depth [10]. Their effect is therefore also considered not significant, and no restrictions are placed for those.

2.8.4 Water Density PWR

The in-core water density for PWR plants is a direct function of the pressure and average moderator temperature in the core, both of which only show minor variations between typical PWR plants; hence the density also only varies slightly. The range specified encompasses typical moderator temperatures. Due to the small variation, the source term analyses shall utilize the TRITON libraries, with the density used in each library, without any further adjustments.

2.8.5 Water Density BWR

The water density for BWR assemblies varies much more than that of PWR fuel, both over the irradiation history of an assembly, and also spatially, over the height of the assembly. Information in [10] and [11] show that it can vary from about 0.1 to about 0.75 g/cm^3, and TRITON libraries cover from 0.1 to 0.9 g/cm^3. However, utilizing densities changing with time or location is not the intent, so using a reasonable average is sufficient. Since a lower value is a more conservative assumption, the lower value of 0.3 g/cm^3 listed in [9] shall be used for the source term analyses here.

2.8.6 Fuel Density

Typically, fuel densities are kept close to the theoretical maximum of 10.96 g/cm^3, for efficiency purposes. However, in practicality, densities are somewhat lower, due to dishing and chamfering of the fuel pellets,

and the limitations in the manufacturing process, no more than about 96% of that value. Older assemblies may have had even lower density, potentially below 10 g/cm^3. The density affects the fuel to water ratio, and in this case, a higher value would be more conservative since it would result in a harder spectrum. Due to the small range of the density, the effect on source terms would not be significant. For the applicability, a generous range is therefore defined, and for the calculations, a fixed value of about 95% of the theoretical maximum should be used regardless of the fuel type, unless site or fuel specific values are available, then these may be used. However, these site or fuel specific values must be within the range listed in Table 2.2.

2.8.7 Specific Power

In [9], it is concluded that higher specific power values are more conservative, and for PWR fuel, values in the range of 20 to 40 MW/mtU are listed as realistic and typical values. In [1] and [2], values of about 40 MW/mtU are used for PWR assemblies, and 30 MW/mtU for BWR fuel. These are realistic and typical for most assemblies, specifically if they are fully burned, but individual assemblies in a core may have slightly higher values, specifically for assemblies irradiated only for a limited number of cycles before the plant shutdown. If the specific power of an assembly, averaged over its entire irradiation time, exceeds those values, then separate source term calculations shall be performed using a bounding specific power density. This must then be documented in the qualification report.

2.8.8 Fuel (Uranium) Mass

The fuel (Uranium) mass has a very small effect on dose rates, due to a compensatory effect between source term and radiation transport calculations. A higher mass results in higher source terms, which would tend to increase dose rates, but then also in higher self-shielding withing the cask, which would tend to reduce dose rates. The range specified in Table 2.2 is therefore the range from [1] and [2], slightly extended to account for variations in actual values. The important aspect is that for consistency between source term and radiation transport calculations, the mass in the radiation transport calculations cannot be more than that in the source term calculation for a specific fuel assembly. For that reason, the design basis fuel assemblies as presented in Section 3 should be used for all assemblies, including assemblies with lower uranium weight, to avoid multitudes of radiation transport analyses. However, it is acceptable to use site-specific uranium weights instead. In that case, this shall be documented in the qualification report, and it must be ensured that both the source term and radiation transport analyses are based on the same mass value. See also next section.

2.8.9 Fuel Condition

The condition of the fuel, e.g., undamaged, damaged, fuel debris, reconditioned, reconstituted fuel, affects predominantly the spatial distribution of the fuel in the dry storage system for the radiation transport calculations, but not the source term calculations which are still depending on burnup, enrichment and cooling time and any core conditions. Since this TR only specifies the methodology for the source term generation, and the radiation transport calculations are addressed in the corresponding FSAR/TS, there are no restrictions on the fuel conditions that need to be considered in this TR.

For some of the fuel conditions, the uranium weights may be lower than that of an undamaged assembly. When considering this in both the source term and radiation transport analyses, it may have some impact on external dose rates. However, due to the compensatory effect discussed in the previous section, the effect would be limited. Hence it is acceptable to model all assemblies as undamaged in both the source term and radiation transport analysis.

Table 2.1

SUMMARY OF THE ASPECTS OF THE FRAMEWORK AND METHODOLOGY

Information	**Document Location**	**Owner**	**Change Control**
Acceptance Criteria Dose Rates	Technical Specifications	NRC	Only via Amendment
Source Term Calculation Methodology	This Topical Report	CoC Holder	Only via Application
Radiation Transport (Dose Rate) Calculation Methodology	FSAR	CoC Holder	Strict 10CFR72.48 Method of Evaluation Controls
Acceptable Content	Qualification Report	CoC Holder / Licensee	Available to NRC for information, but not for approval

Table 2.2

AREA OF APPLICABILITY (Note 1)

Parameter	Applicability	Basis
Fuel	Spent PWR and BWR fuel	[1], [2]
Fuel Burnup	Up to 72 GWd/mtU for PWR fuel Up to 72 GWd/mtU for BWR fuel	TRITON Libraries [8]
Fuel Type	UO_2	Limitation set in this TR
Fuel Assembly Hardware	Standard fuel assembly hardware (upper and lower end fittings, guide tubes or water rods, grid straps, etc, made from steel, zirconium alloy, or Inconel). Assemblies may also contain irradiated or unirradiated steel or zirconium alloy rods in fuel rod locations.	See Section 3, specifically Section 3.2.2 for replacement rods
Non-Fuel-Hardware for PWR assemblies	Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs, also called ORAs), 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. Activated material in the NFH may be zirconium alloys, steel, Inconel, AgInCd or Hafnium.	[1], [2], and Section 3 of this report
Enrichment	0.5 wt% to 5.0 wt% ^{235}U	[1], [2] and range of TRITON Libraries [8]. For enrichment variations see Section 3.6 of this report
Cooling Time	Greater or equal to 1 year	10 CFR 72

In-Core Cycle Average Soluble Boron	0 to 2000 ppm	[10] See Section 2.8.1 of this report
In-Core exposure to control components	All NFH listed above, without any restriction	See Section 2.8.2 and Section 2.8.3 of this report
Water Density, g/cm^3	BWR 0.1 to 0.9; PWR 0.67 to 0.78	See Section 2.8.4 and Section 2.8.5 of this report
Fuel Density	9.0 to 10.96	See Section 2.8.6 of this report
Specific Power	up to 40 MW/mtU for PWR, up to 30 MW/mtU for BWR, without further justification. See Note 2.	See Section 2.8.7 of this report
Fuel arrays	BWR: 7x7 to 11x11, PWR: 14x14 to 17x17	[1], [2]
Fuel Mass (Uranium)	BWR: up to 205 kg; PWR up to 575 kg. See Note 2.	See Section 2.8.8 of this report
Fuel condition	Undamaged, damaged, fuel debris, reconstituted, reconfigured	See Section 2.8.9 of this report
Fuel Cladding	Zirconium based only	Limitation set in this TR

Note 1: The FSAR that this TR is referenced in may specify more restrictive values for some parameters
Note 2: Fuel with values outside this range can be qualified but require separate source term analyses. See Section 2.8.7 and Section 2.8.8 for details.

3.0 SOURCE TERM EVALUATIONS FOR QUALIFICATION OF FUEL

3.1 General

This section specifies the method for performing the source term analyses for the dose rate calculations to qualify fuel in accordance with this Topical Report. This methodology is principally the same as that defined in the FSARs for the various storage systems. Since it is common to various FSARs, it is defined here to avoid duplication of the approval process.

The code to calculate neutron and gamma source terms shall be the ORIGAMI module of the SCALE system, Version 6.2.1 [7] or higher, utilizing the TRITON data libraries supplied with the respective code version. The calculations shall be utilizing the 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. ORIGAMI and TRITON data libraries from SCALE 6.2.1 can be used without further justification. When using a newer version of the SCALE code, it shall be demonstrated, for a small set of BECTs that span the variations of the burnups and cooling times to be qualified, that the results (dose rates) are within 5% of those from SCALE Version 6.2.1. The value of 5% is a typical value for uncertainties of the radiation transport analyses, so any source terms from a different code version that keep the dose rate results within that 5% variation would indicate that the source terms are essentially the same as those from SCALE Version 6.2.1. These dose calculations to qualify a newer version shall be performed for the storage and transfer cask for which the fuel is to be qualified. The small set of BECTs shall be the same as that selected as representative in the corresponding FSAR (see Appendix B, Section B.4), i.e. one combination with shortest cooling time and corresponding lower burnup, and one combination with highest burnup and corresponding longer cooling times, both consistent with the dose rate limits specified, and using enrichments consistent with the burnups.

For SCALE 6.2.1, the TRITON libraries [8] supplied with the code as specified in Table 3.1 and 3.2 shall be used.

When performing the ORIGAMI calculations, a single full power cycle shall be used to achieve the desired burnup, since this has been shown to result in conservative source terms relative to actual multicycle power operation.

Unless a different design basis has been defined and qualified to be used in conjunction with this TR in the corresponding FSAR, source term calculations shall be performed for the design basis assemblies listed in either Table 3.1 or Table 3.2. Those have been shown in [1] to reasonably bound all assembly types in the corresponding FSARs.

3.2 Gamma Source

The gamma source term is comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from ^{60}Co activity of any

structural material in the fuel element, in the active region and above and below the active fuel region. These sources are determined through the source term calculations outlined here. The third source is from n-gamma reactions. This third source shall be considered directly in the radiation transport calculations.

3.2.1 Gamma Source from Fuel

Previous analyses (see Reference [1]) indicated that it is appropriate and necessary to include all photons with energies in the range of 0.45 to 3.0 MeV. Photons with energies below 0.45 MeV are too weak to penetrate the typical shielding constructions, while the effect of gammas with energies above 3.0 MeV was found to be insignificant since the source of gammas in this range (i.e., above 3.0 MeV) is extremely low.

To appropriately consider spectral effects, i.e., differences of source terms as a function of the gamma energy, a sufficiently fine energy group structure shall be used in the analyses. There are two options for selecting this energy structure:

- Use the energy structure from Table 3.5, taken from Reference [1]. The same energy structure must be used in the radiation transport analyses.
- Alternatively, the energy structure can be defined in the FSAR that documents the qualification of the radiation transport analyses. This source structure must then also be used in the source term calculations, and the structure must again be identical between source term and radiation transport analyses

The radiation transport analyses should apply the gamma source terms from the active region as a histogram, i.e. with equal probability of particle energies within each energy group.

3.2.2 Gamma Source from Activation of Non-Fuel Materials

An important source of activity in the fuel assembly arises from the activation of ^{59}Co to ^{60}Co in various non-fuel materials and components. These include the structural material above and below the fuel, guide tubes, water rods, channel boxes, and grid spacers, hold-down springs, etc. Additionally, assemblies may include irradiated metal rods that have been inserted to replace fuel rods with damaged cladding, or that have been part of the initial fuel assembly design. If any of these components are made from steel or Inconel, the activity can be substantial, and must be considered in the source term evaluation. If they are made from zircaloy, they can be neglected since it does not have a significant ^{59}Co impurity level. Reference [3] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. This impurity level is considered conservative for fuel which has been manufactured since the mid-to-late 1980s after the implementation of an industry wide cobalt reduction program. Based on this, Inconel and stainless steel in the non-fuel regions shall both be modeled with at least 1.0 gm/kg impurity level for fuel manufactured in or after 1990. However, it is recognized that materials used in earlier assemblies may have had higher impurity levels, with up to 2.2 gm/kg for steel and 4.7 gm/kg or more for Inconel. While such assemblies would now have undergone significant decay and corresponding reduction of the ^{60}Co source terms, these values should be used for assemblies manufactured before 1990 to assure the analyses are reasonably conservative. Lower values than those listed above may be used if documented records for those values are available. These

records will then need to be referenced in the qualification report, and the qualification would be limited to the assemblies that these records apply to.

Some of the fuel assembly designs utilized Inconel in-core grid spacers while others use zircaloy in-core grid spacers. In the mid-1980s, the fuel assembly designs using Inconel in-core grid spacers were redesigned to use zircaloy in-core grid spacers, which contain an insignificant amount of ^{59}Co. Source term calculations can be performed with or without considerations of Inconel grid spacers. Considering the presence of Inconel spacers bounds any type of spacers. If Inconel spacers are not considered, this shall be clearly stated in the qualification report, and the qualification can then only be used for fuel that does not contain them.

The non-fuel data listed in Table 3.1 were taken from References [3], [4], and [5].

In addition to the ^{59}Co activation, activated materials in CRAs may create an additional gamma source. CRAs are fabricated from various materials. The cladding is typically stainless steel, although Inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B_4C and hafnium has also been used. AgInCd produces a noticeable source term in the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs that shall be used.

There are principally two ways to evaluate the source terms for these activations, which are selected and used in combination with the way they are applied in the radiation transport analyses to calculate the dose rates:

- For the non-fuel materials that are always present in active region of the fuel, the materials will be considered directly in the ORIGAMI calculations, using the “nonfuel” and optional “fracnf” keywords (see Section 3.5). Through this, the activity of these materials is directly included in the gamma source term of the fuel assembly, and can then be directly considered in or combined with the results of the radiation transport analysis.
- For other non-fuel hardware that is present in every assembly but not in the active regions (e.g., top and bottom end fittings), or that may not be present in every assembly (e.g., NFH, steel rods, Inconel spacers), it is not practical to perform separate ORIGAMI calculations for each piece of hardware since that would complicate the application of the corresponding source terms in the radiation transport analysis. Instead, the source term for a fixed amount (e.g., 1 g of ^{59}Co, 1kg of AgInCd) is calculated, and then the source from each piece of hardware can be determined from the corresponding source term, the amount of the material in that piece (Tables 3.1, 3.2, 3.4, 3.7 or 3.8, as applicable), and the applicable flux factor (Table 3.3, Table 3.7 or Table 3.8, as applicable), and assigned to the appropriate location in the dose evaluation.

Typically, the mass of NFH devices is not considered in the radiation transport analyses as additional shielding. This is a conservative approach since it neglects material that would provide some additional shielding. If any credit for these masses is taken, then this must be justified in the FSAR describing the radiation transport analyses, and reviewed and approved as part of the corresponding CoC approval.

As stated above, assemblies may also contain steel rods replacing fuel rods, either as part of the design, or as the result of a repair process when damaged fuel rods were removed. They are treated in the same way as other steel components of the assembly, however, due to the potentially high local dose contribution from those, they deserve a separate discussion:

The principal activation and source term is calculated in the same way as for any other steel component of the fuel, namely by ORIGAMI, for the given burnup and cooling time, initially as a value of activity per unit mass. This is then multiplied by the mass of the steel rods and the flux factor in each of the axial section of the fuel assembly to get the total activity (see Table 3.3). The resulting activity is then added to the source for each axial region, and each assembly location that contains those rods.

While there are usually only limited number of assemblies with such rods, the source may be added to more assemblies in a basket if dose rate margins permit, just to simplify the evaluations. Also, the source strength of the active fuel that the steel rods are modeled or assumed in may be reduced according to the number of remaining rods, or it may be conservatively used as it is without the steel rods. However, these are part of the modeling details of the radiation transport evaluations, and need to be discussed either in the FSAR or the qualification report.

Depending on the number and location of these rods, the radiation transport model may be adjusted to account for the location of the source terms of the steel rods, and/or the potentially reduced self shielding of those areas. However, if the number of rods is small and/or they are not strictly localized, such adjustments are not considered necessary. In this context, 5% of the rods in an assembly are considered a small number. However, these are part of the modeling details of the radiation transport evaluations, and need to be discussed either in the FSAR or the qualification report.

Note that assemblies may also include unirradiated material added after removal from the core, such as ITTRs to support lifting of the fuel assemblies. These do not add to the source terms.

3.3 Fuel Neutron Source

The neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu contents in the fuel, which increases the inventory of other transuranium nuclides such as Cm. Because of this effect and in order to obtain conservative source terms, lower bound initial fuel enrichments shall be used in the analyses.

As for gamma sources, neutron source terms shall be generated by energy group in a suitable group structure. The same approach is applicable here that was discussed in Section 3.2.1 for gamma source terms, with the energy group structure for Neutrons from Reference [1] presented in Table 3.6.

The radiation transport analyses should apply the neutron source terms from the active region as a histogram, i.e., with equal probability of particle energies within each energy group.

3.4 Non-Fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), axial power shaping rods (APSRs) and neutron source assemblies (NSAs) are permitted for storage as an integral part of a PWR fuel assembly. If they are used, their source terms shall be evaluated based on the specifications below, and considered in the radiation transport analyses.

The burnup assigned to a NFH is the burnup that a fuel assembly accumulates while the NFH is inserted. If the NFH was present in several assemblies, as would have been typically the case for TPDs, CRAs, APSRs, and NSAs, the burnup assigned to the NFH is the addition of those burnups. It would be impractical to prepare separate source term calculations for each such NFH, with the applicable sequence of burnup exposures. Instead, a single calculation may be used with an upper bound burnup and corresponding cooling time, or a set of calculations with different burnups and corresponding cooling time limits. In all cases, the flux from the assembly shall be reset every 45 GWd/mtU in the calculation. There is no principal limit to the accumulated burnups of NFHs, hence no limits are specified in Table 2.2. However, the qualification report shall specify the burnup and cooling time combination(s) that are evaluated, and only NFH that meet these burnup and cooling time combination(s) are qualified for loading.

The specifications in the tables at the end of this section are considered reasonable and conservative for determining source terms for typical fuel assemblies and NFH, specifically with respect to the amounts and axial configurations of irradiated material. If there are indications that the amounts of components to be qualified are significantly higher from those, the differences need to be evaluated, source terms need to be adjusted accordingly, and this is to be documented in the qualification report. This adjustment would be made in the consideration of the NFHs as described in Section 3.2.2.

3.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different from that of a fuel assembly. Vibration suppressor inserts are considered to be in the same category as BPRAs for the purposes of this source term analyses since these devices have the same configuration (long non-absorbing thimbles which extend into the active fuel region) as a BPRA without the burnable poison.

TPDs are made of stainless steel and may contain a small amount of Inconel. These devices extend down into the plenum region of the fuel assembly but typically do not extend into the active fuel region. Since these devices are made of stainless steel, there is a significant amount of ^{60}Co produced during irradiation. This is the only significant radiation source from the activation of steel and Inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of Inconel in this region. Within the active fuel zone, the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel.

Since the systems are designed to store many varieties of PWR fuel, a representative TPD and BPRA was determined for the purposes of the analysis. This was accomplished by analyzing BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5] and [6] to determine the TPD and BPRA which produced the highest ^{60}Co source term for a specific burnup and cooling time. The TPD was determined to be the Westinghouse 17x17 guide tube plug and the BPRA was determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a single hypothetical BPRA. The masses of these devices are listed in Table 3.4. These should be used in the source term calculations, and in this case no further justification is needed. Masses different from those in Table 3.4 may be used, either lower (to take credit of actual insert weights), or higher (to consider designs not bounded by those considered here), the different masses need to be documented and justified in the FSAR or the qualification report.

Note further that since inserts are managed and handled separately from the fuel, the burnup and cooling time of an insert in an assembly may be different from that of the assembly.

3.4.2 CRAs and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are also an integral, yet removable, portion of a PWR fuel assembly. These devices are utilized for many years (upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant. Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, typically only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. As stated before, CRAs are fabricated from various materials, and the AgInCd CRAs are the bounding CRAs that shall be used.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W stainless steel clad APSRs: gray and black. According to reference [5], the black APSRs have 36 inches of AgInCd as the absorber while the gray ones use 63 inches of Inconel as the absorber. Because of the ^{60}Co source from the activation of Inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR and shall be used.

The materials and corresponding masses listed in Table 3.7 and Table 3.8 shall be used in the source term calculations, except for CRAs inserted more than 10%, where adjustments are required as discussed below.

These are based on a review of publicly available fuel information and are considered reasonable and sufficiently conservative.

The assumed insertion of CRAs of 10% during the entire irradiation is considered extremely conservative, since only a fraction of CRAs is permitted to be inserted into the active region at any given time during power operation, and since the insertion depth is limited and tightly controlled for both efficiency and stability reasons. However, if there are indications that individual CRAs have been inserted, on average, more than 10% into the active region, then this shall be considered in both the source term calculations for this, and the corresponding radiation transport analyses. This adjustment would result in a modification of the values in Table 3.7, as follows:

- The first row represents the depths (second and third column) and masses (fifth and sixth column) of the 10% insertion. These values have to be multiplied by the ratio of the next insertion to the 10% assumption.
- Then start and finish values for the 2 following rows have to be adjusted in accordance with the new insertion length.

3.4.3 Discrete Neutron Source

Neutron source assemblies (NSAs) are used in reactors for startup. There are different types of neutron sources (e.g., californium, americium-beryllium, plutonium-beryllium, polonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the guide tubes of a fuel assembly and are usually removable.

The neutron source term of these neutron source is usually negligible compared to those from fuel assemblies, specifically for the secondary sources. However, for some primary sources that may not be the case. Hence one of the following three options shall be used to consider the neutron source strength from NSAs:

- 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. The contribution can be considered negligible if an NSA present in all assemblies provides less than 1% of the total neutron source term of a cask.
    - This may be the preferred approach for antimony-beryllium source. In these, the neutron production is driven by the antimony decay, which as a half-life of only 60 days, hence the generation can be shown to be negligible after just a few years of cooling time.
- If the neutron source term of the NSA is not negligible but is quantified, it can be considered in the analyses to show compliance with the dose rate limits. In that case, the number and location of the NSAs qualified becomes part of the qualified content.
- If no evaluation is performed, only one NSA is permitted in a basket, and shall be located near the center of that basket, consistent with the approach in Reference [1].

Additionally, the stainless steel and Inconel portions of the NSAs become activated during in-core

operations, potentially producing a significant amount of ^{60}Co. Their design resembles a combination of a BPRA and a TPD. They typically contain a limited number of full-length rods (similar to a BPRA), with thimble plugs for the remainder of the guide tube positions that do not contain a full-length rod. For the source term calculations, they shall be considered a combination of the masses for the TPD (Table 3.4), combined with the in-core mass for a BPRA (also Table 3.4), but where this mass may be scaled down proportional to the number of full-length rods in the NSA.

3.5 ORIGAMI Calculations

There are numerous modeling and parameter options for performing ORIGAMI calculations. For the purpose of the source term calculations in accordance with this TR, it is sufficient to use the modeling option for a fully lumped assembly. The principal structure of an input file for such a calculation is shown in Table 5.4.1 of [7], although parameters may be different. The keywords and parameters that must be present are discussed below, unless specified as optional. For any ORIGAMI parameters not specified below, the code appropriately uses the default values stated in [7].

Keyword	Parameter	Comment
libs	TRITON Library	see Table 3.1
fuelcomp	fuel composition	use uox() with enrich= and dens= specification
options / mtu	fuel weight	uranium weight only
options / fracnf	total non-fuel mass as fraction of fuel mass	optional, may simplify non-fuel specification
pz and meshz	axial configuration	for a lumped assembly, pz=1 or omitted, meshz=any arbitrary number or omitted
nonfuel	combined composition of all nonfuel material in the active region	must include fuel cladding, guide tubes, water rods, grid straps, steel rods (if applicable), channel (BWR), all using their representative material composition, and cobalt content To establish the cobalt activation in NFH, a calculation should be performed where a defined amount of ^{59}Co is added, then the activity of ^{60}Co for that amount is extracted from the output file, and can be applied to the NFH, after scaling with the flux factors and scaling by the amount of ^{59}Co in the NFH.
hist / cycle	cycle information	only one cycle with a power>0 nlib should be set so the burnup steps are no more than about 5 GWd/mtU

ggrp and ngrp	energy group boundaries	see Section 3.2 and 3.3
modz	moderator density	BWR only, see Section 2.8.5
print	various	optional to generate additional output tables.

3.6 Fuel Assemblies with Axial Blankets

Some assemblies contain axial blankets, i.e., small sections at the top and bottom of the assemblies that have reduced enriched or natural uranium. How such assemblies are considered in the dose analyses, including the selection of the relevant burnup and enrichment parameters of the assembly, depends on the modeling approach taken in the radiation transport analyses. Since the modeling of the radiation transport analyses are part of the FSAR, not this TR, the modeling approach for blanketed assemblies and the definition of the relevant burnup and enrichment characteristics will be part of the review and approval of the CoC that incorporates the reference of this TR.

Table 3.1

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD FUEL

	PWR	**BWR**
Assembly type/class	WE 17×17	GE 10×10
Active fuel length (in.)	144	144
No. of fuel rods	264	92
Rod pitch (in.)	0.496	0.51
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.374	0.404
Cladding thickness (in.)	0.0225	0.026
Pellet diameter (in.)	0.3232	0.345
Pellet material	UO_2	UO_2
Specific power (MW/MTU)	43.48	30
Weight of UO_2 (kg)	532.150	213.531
Weight of U (kg)	469.144	188.249
No. of Water Rods/ Guide Tubes	25	2
Water Rod/ Guide Tube O.D. (in.)	0.474	0.98
Water Rod/ Guide Tube Thickness (in.)	0.016	0.03
Lower End Fitting (kg)	5.9 (steel)	4.8 (steel)
Gas Plenum Springs (kg)	1.150 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.793 (Inconel) 0.841 (steel)	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	6.89 (steel) 0.96 (Inconel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (Inconel)	0.33 (Inconel springs)
TRITON Library for SCALE 6.2.1 [7], [8]	w17x17	ge10x10-8

Table 3.2

DESCRIPTION OF ALTERNATIVE DESIGN BASIS ZIRCALOY CLAD FUEL

	PWR	**BWR**
Assembly type/class	B&W 15×15	GE 7×7
Active fuel length (in.)	144	144
No. of fuel rods	208	49
Rod pitch (in.)	0.568	0.738
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.428	0.570
Cladding thickness (in.)	0.0230	0.0355
Pellet diameter (in.)	0.3742	0.488
Pellet material	UO_2	UO_2
Specific power (MW/MTU)	40	30
Weight of UO_2 (kg)	562.029	225.177
Weight of U (kg)	495.485	198.516
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
Lower End Fitting (kg)	8.16 (steel), 1.3 (Inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (Inconel). 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (Inconel)	0.33 (Inconel springs)
TRITON Library for SCALE 6.2.1 [7], [8]	bw15x15	ge7x7-0

Table 3.3

SCALING FACTORS USED IN CALCULATING THE ^{60}Co SOURCE

Region	**PWR**	**BWR**
Handle	N/A	0.05
Upper End Fitting	0.1	0.1
Gas Plenum Spacer	0.1	N/A
Expansion Springs	N/A	0.1
Gas Plenum Springs	0.2	0.2
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15

Table 3.4

DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE

Region	**BPRA**	**TPD**
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of Inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 3.5
Energy Structure for Developing Fuel Gamma Source Terms

Lower Energy	**Upper Energy**
(MeV)	(MeV)
0.45	0.7
0.7	1.0
1.0	1.5
1.5	2.0
2.0	2.5
2.5	3.0

Table 3.6
Energy Structure for Developing Neutron Source Terms

Lower Energy (MeV)	**Upper Energy (MeV)**
1.0e-01	4.0e-01
4.0e-01	9.0e-01
9.0e-01	1.4
1.4	1.85
1.85	3.0
3.0	6.43
6.43	20.0

Table 3.7

DESCRIPTION OF DESIGN BASIS CONTROL ROD ASSEMBLY CONFIGURATIONS FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel[1]			**Flux Weighting Factor**	**Mass of cladding (kg Inconel)**	**Mass of absorber (kg AgInCd)**
Start (in)	**Finish (in)**	**Length (in)**			
0.0	15.0	15.0	1.0	1.32	7.27
15.0	18.8	3.8	0.2	0.34	1.85
18.8	28.25	9.45	0.1	0.83	4.57

Table 3.8

DESCRIPTION OF DESIGN BASIS AXIAL POWER SHAPING ROD CONFIGURATION S FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel[1]			**Flux Weighting Factor**	**Mass of cladding (kg Steel)**	**Mass of absorber (kg Inconel)**
Start (in)	**Finish (in)**	**Length (in)**			
0.0	63.0	63.0	1.0	5.29	24.89
63.0	66.8	3.8	0.2	0.32	1.51
66.8	76.25	9.45	0.1	0.79	3.73

1 This information shall be considered in the Radiation Transport Calculations, to correctly locate the source relative to the active region of the fuel assemblies

4.0 ANALYSIS PROCESS

Below is an outline of the principal steps of the overall analysis process. For each step, it identifies what methodology is used and where that methodology is defined, and references locations in this topical report that provide additional information for the respective step. Note that not all methodologies are described in this report.

Step 1: Generation and Collection of input parameters and input data

- Define what is to be qualified
    - Cask systems
    - Fuel and anticipated BECTs
    - NFH and anticipated burnups and cooling times
- Compile all parameters that are required for the analyses and that are required to verify the applicability of the TR.
- Identify and document any site-specific depletion parameters and other analytical aspects that may be different from the topical report, and where the TR allows such variations.
    - If any parameters are outside of the applicability of the TR, then the method in the TR can not be used to qualify corresponding fuel assemblies.
- Loading pattern(s), i.e. the identification of proposed or anticipated fuel characterization for the basket location or locations
    - Determination of the loading pattern(s) is a separate process not addressed here. It will reflect the fuel and NFH to be loaded, and potentially other aspects, such as thermal requirements or specific dose requirements other than those present in the context of this TR. The pattern could be developed using a manual process, or some specialized software.
- All relevant information on the fuel assemblies to be qualified, including NFH if applicable
    - See Section 2.8 and Table 2.2, in combination with any more specific requirements from the corresponding FSAR/TS for the information that may be required.
- Documentation: All this information needs to be either referenced appropriately in the qualification report, or directly documented in there.

Step 2: Source Term Calculations

- Note that there may be cases where only new loading pattern(s) are to be qualified for fuel where source term calculations were already performed. In that case, this step is skipped, and the previous calculations are referenced.
- Select the design basis assembly from either Table 3.1 or Table 3.2.
- For any NFH, develop the modeling in accordance with Sections 3.2, 3.4 or 3.5, as applicable
- Perform the ORIGAMI calculations with the TRITON libraries, for the fuel to be qualified.
    - See Section 3.1 and Section 3.5 for a discussion on ORIGAMI calculations
- Extract neutron, gamma and ^{60}Co source terms from the ORIGAMI outputs.
    - Energy structures for gamma and neutron sources shall meet the requirements listed in Section 3.2 and Section 3.3.
    - ^{60}Co for fuel and non-fuel hardware is calculated
- Documentation: If source term calculations are only performed for a small set, e.g. for a specific site, the calculations may be documented as part of the qualification report. If calculations are performed supporting different loading pattern(s) possibly in different qualification reports, it is more appropriate to document the source term calculations and the results of those in a separate report that can then be referenced accordingly.

Step 3: Radiation Transport and Dose calculations

- It is a common practice to perform radiation transport calculations normalized to a fixed number of starting particles, and then just combine those with the source terms to establish dose rates. This significantly reduces the calculation effort. If only new content (i.e. new fuel and/or new loading patterns) are to be qualified for an already qualified system, no radiation transport calculations may be needed, but new dose rate values need to be calculated with the existing radiation transport analyses and the newly generated source terms. This will typically be the case if the system to be qualified is the same as that evaluated in the corresponding FSAR, and no modifications are required to the systems. In that case, no new radiation transport calculations are needed, and the existing calculations need to be referenced appropriately.
- Select transfer and storage systems that the content is to be qualified for
    - Take the applicable radiation transport model that is consistent with the CoC revision that the fuel is to be qualified for. This may be from FSAR, including any applicable changes performed under 10 CFR 72.48 for that system. This may also be taken from an earlier qualification report for same systems, considering it is applicable to the CoC.

  - If need be, make changes to model. The extent of permissible changes is specific to the storage system, and would be defined and described in the corresponding FSAR/TS that references this TR. An example of how such permitted changes would be specified in the FSAR is shown in Appendix C, Section C.3. A 72.48 evaluation need to be performed for these changes, to ensure they do not require NRC approval
  - Any other changes that are made to the radiation transport model, as permitted by this TR, need to be documented in the qualification report.
- If changes to the model were made, confirm that the dose rate locations conform to the FSAR/TS requirements.
- If new models had been generated, perform radiation transport analyses.
- Select the set of burnup, enrichment and cooling times that need to be analyzed (follow guidance in Section 2.6)
- Determine dose rate limits and corresponding dose locations from applicable FSAR/TS
- Combine the source terms for the selected BECTs with the results of the radiation transport calculation to result in dose rates at the locations. See Section 2.6 on the options for considering BECT ranges and regions in the basket.
- To show that the content defined in Step 1 is qualified through the process in this topical report, dose rates shall be below the corresponding FSAR/TS limits.
- Documentation: All new and unique calculations are to be documented in a report. Pre-existing calculations for source terms and/or radiation transport calculations should not be repeated, but referenced appropriately.

5.0 CONCLUSION

This Topical Report provides the framework and part of the methodology for qualifying fuel loading patterns, and when referenced in a Certificate of Compliance will provide the ability to more efficiently load spent fuel into dry storage.

6.0 REFERENCES

[1] HI-STORM 100 FSAR, Holtec Report No. HI-2002444, Latest Non-Proprietary Revision [USNRC Docket 72-1014].

[2] HI-STORM FW FSAR, Holtec Report No. HI-2114830, Latest Non-Proprietary Revision [USNRC Docket 72-1032].

[3] A.G. Croff, M.A. Bjerke, G.W. Morrison, L.M. Petrie, "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, September 1978.

[4] J.W. Roddy et al., "Physical and Decay Characteristics of Commercial LWR Spent Fuel," ORNL/TM-9591/V1&R1, Oak Ridge National Laboratory, January 1996.

[5] "Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," DOE/RW-0184, U.S. Department of Energy, December 1987.

[6] "Characteristics Database System LWR Assemblies Database," DOE/RW-0184-R1, U.S. Department of Energy, July 1992.

[7] B.T. Rearden and M.A. Jessee, Eds., SCALE Code System, ORNL/TM-2005/39, Version 6.2.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2016).

[8] B. Ade and B. Betzler, ORIGEN Reactor Libraries, Oak Ridge National Laboratories, April 8, 2016

[9] "Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks", NUREG/CR-6716, ORNL/TM-2000/385, Oak Ridge National Laboratory, Oak Ridge, Tennessee, March 2001.

[10] "Fuel Assembly Reference Information for SNF Radiation Source Term Calculations", ORNL/SPR-2021/2093, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 2021.

[11] "Axial Moderator Density Distributions, Control Blade Usage, and Axial Burnup Distributions for Extended BWR Burnup Credit", NUREG/CR-7224, ORNL/TM-2015/544, Oak Ridge National Laboratory, August 2016.

APPENDIX A EXAMPLES FOR FUEL QUALIFICATIONS

A.1 Overview

To illustrate the application of the methodology articulated in this TR, three example fuel qualifications are presented in this Appendix. The first example is for a general set of fuel qualifications, including several systems and various fuel types, and a rather generic distribution of fuel in a basket. The second example shows an evaluation for a site-specific fuel contents, for a single system, a single assembly type, and a simple single BECT. The third example is also an evaluation for site-specific content but for a very specific distribution of fuel in the basket. The main focus of these examples is to show how content may be specified for the different generic or site-specific approaches. For a more extensive outline of the corresponding activities and analyses in each case, see Section 4 of the main part of this report. Further see the example of a qualification report in Appendix E, based on Example 3 shown below.

A.2 Example 1, Generic Fuel Qualification

The principal steps are as follows:

Step 1: Define inputs

Canister: 32 Assembly Canister A, with regions defined in Figure A.1

Storage Cask: Storage Casks A, B, C

Transfer Casks: Transfer Casks A, B, C

Burnup, Enrichment and Cooling times (BECTs), see Table A.1. In this example, three different sets are defined.

For NFH limitations, see Table A.1a

Fuel Types: W17x17, BW15x15

Dose Rate Limits and corresponding locations for all systems listed above, as defined in the corresponding FSAR/TS

Confirm the content to be qualified conforms to the Area of Applicability (Table 2.2)

Step 2: Perform Source term analyses for all fuel types, and BECTs, consistent with the methodology in Section 3.0 of this TR. Use either of the design basis assemblies in Tables 3.1 and 3.2.

Step 3: Perform dose rate analyses, consistent with the methodology in this Topical Report, and utilizing the shielding models and corresponding parameters from the FSAR, for the fuel selected in Step 2.

Ensure that the qualification covers all systems, fuel assemblies and BECTs. For the BECTs in Table A.1, with 2 regions, 3 cases, and 14 burnup steps, use Option 1 from Section 2.6, that requires $3x14^2$ dose evaluations for fuel, and some additional dose evaluations for the NFH, and for each dose location determine the maximum dose rate over all these evaluations.

Ensure that the calculated dose rates meet the dose rate limits (see Step 1).

An example result table is shown in Table A.2

A.3 Example 2, Site-Specific Fuel Qualification, Typical Plant Operation

The principal steps are as follows:

Step 1: Define inputs

Canister: 32 Assembly Canister A, Uniform Loading

Storage Cask: Storage Casks A

Transfer Casks: Transfer Casks A, with site specific (possibly reduced) shielding thicknesses.

Burnup, Enrichment and Cooling times (BECTs):
Maximum Burnup 55 GWd/mtU
Minimum Enrichment 4.0%
Minimum Cooling time 5 years

Fuel Types: W17x17

No NFH are qualified

Dose Rate Limits and corresponding locations for all systems listed above, as defined in the corresponding FSAR/TS

Confirm the content to be qualified conforms to the Area of Applicability (Table 2.2)

Step 2: Perform Source term analyses for all fuel types, and BECTs, consistent with the methodology in Section 3.0 of this TR.

Step 3: Perform dose rate analyses, consistent with the methodology in this Topical Report, and utilizing the shielding models and corresponding parameters from the FSAR.

Ensure that the qualification covers all systems, fuel assemblies and BECTs.

Ensure that all calculated dose rates meet the dose rate limits.

A.4 Example 3, Site-Specific Fuel Qualification, Decommissioning Operation

The principal steps are as follows:

Step 1: Define inputs

Canister: 32 Assembly Canister A, with regions defined in Figure A.2

Storage Cask: Storage Casks A

Transfer Casks: Transfer Casks A

Burnup, Enrichment, and Cooling times (BECTs), see Table A.3

For NFH limitations, see Table A.3a

Fuel Types: W17x17

Dose Rate Limits and corresponding locations for all systems listed above, as defined in the corresponding FSAR/TS

Confirm the content to be qualified conforms to the Area of Applicability (Table 2.2)

Step 2: Perform Source term analyses for all fuel types, and BECTs, consistent with the methodology in Section 3.0 of this TR.

Step 3: Perform dose rate analyses, consistent with the methodology in this Topical Report, and utilizing the shielding models and corresponding parameters from the FSAR.

Ensure that the qualification covers all systems, fuel assemblies and BECTs.

For the BECTs in Table A.3, with 5 regions and 14 burnup steps, use Option 2 from Section 2.6, that requires 5x14 dose evaluations for fuel to determine the bounding BECT for each dose location, and then one calculation for each dose location to determine the dose rate for the combination of those BECTs. Some additional dose evaluations are needed for NFH.

Ensure that all calculated dose rates meet the dose rate limits.

An example result table is shown in Table A.4

Table A.1 BECTs for Example 1

Case		1		2		3	
Region (See Figure A.1)		1	2	1	2	1	2
Maximum Burnup	Minimum Enrichment	Minimum Cooling Time (Years)					
5000	1.1	1	1.5	1	1	1.25	1
10000	1.1	1.25	2.5	1.75	1.75	2	1.5
15000	1.6	1.75	3	2.25	2.25	2.5	1.75
20000	1.6	2	3.75	2.75	2.75	3.25	2.25
25000	2.4	2.5	4	3.25	3.25	3.5	2.75
30000	2.4	2.75	5	3.75	3.75	4	3
35000	2.9	3	7	4	4	4.5	3.5
40000	3.2	3.5	9	4.5	4.5	5	3.75
45000	3.6	3.75	11	5	5	6	4
50000	3.6	4	16	6	6	8	4
55000	3.9	4	21	8	8	11	5
60000	3.9	5	27	11	11	16	6
65000	4.5	6	31	13	13	20	7
70000	4.5	7	36	18	18	24	9

Table A.1a Burnup and Cooling Time Requirements for NFH

Minimum Cooling Time (Years)	Maximum Burnup, MWd/mtU	
	BPRAs	TPDs, NSAs, CRAs, APSRs
3	30,000	180,000
10	50,000	630,000

Table A.2 Dose Comparison for Example 1

Dose Location (see Reference [1], Figure 5.1.13)	Maximum Calculated Dose Rate, mrem/hr	Dose Rate Limit for fuel qualification, mrem/hr
Storage Cask B (bounds A and C)		
1	100	200
2	200	300
3	200	300
4	30	100
Transfer Cask C (bounds A and B)		
1	500	800
2	600	900
3	500	600
4	50	100

Note that these are arbitrary values for illustrative purposes, not the results of any specific calculation.

Table A.3 BECTs for Example 3

Region (See Figure A.2)		1	2	3	4	5
Maximum Burnup	Minimum Enrichment	Minimum Cooling Time (Years)				
5000	1.1	2.25	1.5	1.25	1	1
10000	1.1	3.5	2.5	2	1.5	1
15000	1.6	4	3	2.5	2	1
20000	1.6	6	3.75	3	2.5	1
25000	2.4	9	4	3.5	2.75	1.25
30000	2.4	17	5	4	3.25	1.5
35000	2.9	22	7	4.5	3.5	1.75
40000	3.2	30	9	5	3.75	1.75
45000	3.6	43	11	6	4	2
50000	3.6	51	16	8	5	2.25
55000	3.9	57	21	10	6	2.25
60000	3.9	n/a	27	14	7	2.5
65000	4.5	n/a	31	17	9	2.75
70000	4.5	n/a	36	22	11	3

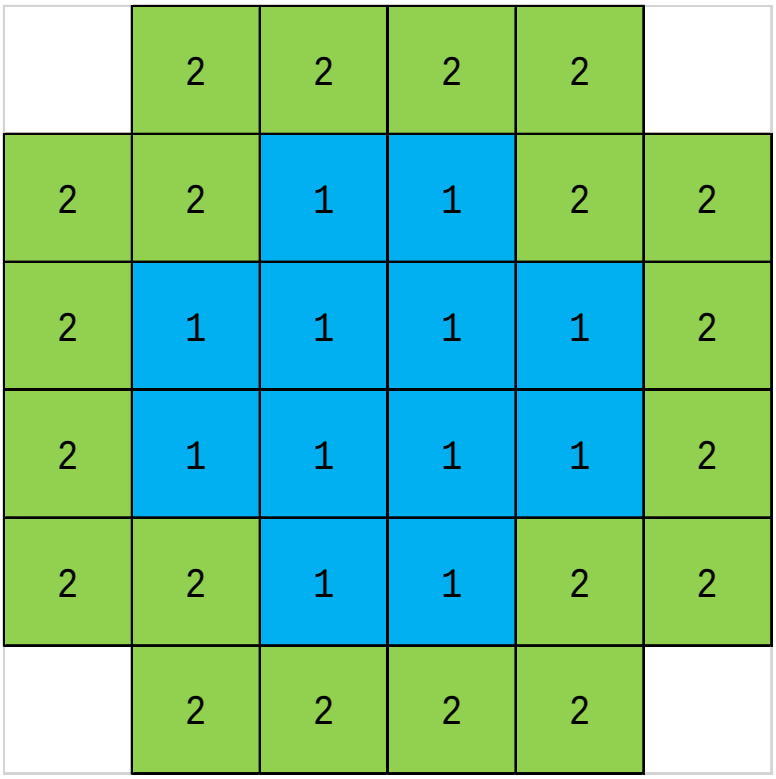
Table A.3a Burnup and Cooling Time Requirements for NFH

Minimum Cooling Time (Years)	Maximum Burnup, MWd/mtU	
	BPRAs	TPDs, NSAs, CRAs, APSRs
3	30,000	180,000
10	50,000	630,000

Table A.4 Dose Comparison for Example 3

Dose Location (see Reference [1], Figure 5.1.13)	Maximum Calculated Dose Rate, mrem/hr	Dose Rate Limit for fuel qualification, mrem/hr
Storage Cask		
1	100	200
2	200	300
3	200	300
4	30	100
Transfer Cask		
1	500	800
2	600	900
3	500	600
4	50	100

Note that these are arbitrary values for illustrative purposes, not the results of any specific calculation.



NFH is permitted in the following regions:

BPRAs, TPDs	Region 1 and 2
CRAs	Region 1
NSAs	1 NSA per cask, in Region 1

Figure A.1 32 Assembly Basket Layout with the Region Number identified in each Cell for Example 1

	5	3	3	5	
5	4	2	2	4	5
3	2	1	1	2	3
3	2	1	1	2	3
5	4	2	2	4	5
	5	3	3	5	

NFH is permitted in the following regions:

BPRAs, TPDs	All regions
CRAs	Regions 1 and 2
NSAs	1 NSA per cask, in Region 1

Figure A.2 32 Assembly Basket Layout with the Region Number identified in each Cell for Example 3

APPENDIX B REQUIREMENTS FOR FSAR/TS CONTENT

The designs and calculational models for the radiation transport evaluations are documented in the corresponding FSARs, together with any applicable acceptance criteria and specification of the area of applications. Requirements for the information that needs to be provided in the FSAR are summarized below.

B.1 Calculational Models

1) The calculational models shall represent the designs with sufficient and reasonable level of detail. Modern Monte Carlo codes for radiation transport evaluations, such as MCNP, are capable to represent a geometry without any significant simplifications that may affect the quality of the results.
   a) Overall dimensions, and extension and properties of major shielding materials can be modeled realistically or in a bounding fashion. In this context, bounding fashion would be modeling with a lower bound thickness or density.
   b) However, for local details, specifically inside of the system, modeling of intricate details is not necessary, as long as the overall shielding effect is reasonably represented.
2) Streaming paths need special attention, and a higher level of detail may be needed there to assure the streaming is considered.
3) Design Basis Fuel assemblies are acceptable to be modeled with several axial sections of different materials, one of them being the active region, with a homogenized material mixture in each section representing the materials in that section.
4) The statistical uncertainties of dose rates to be compared to the acceptance criteria should be reasonable. As general guideline, overall uncertainty should be no more than 5%, with individual contributions (i.e., gamma, ^{60}Co, neutrons, n-gamma) no more than 10% each, consistent with Reference [1].
5) The masses that are considered in the model for self-shielding of fuel shall be consistent with (i.e., the same or lower than) the masses utilized in the source term calculations.
6) The calculations shall consider the axial burnup distribution of the fuel assemblies.
7) Deleted
8) The text needs to identify the aspects of the design that can be changed under 10CFR72.48

B.2 Acceptance Criteria

1) Acceptance criteria are dose rates in selected locations around the transfer or storage casks.
   a) Number and location of the dose points should be selected to be representative of the contents of the cask. For example, for a vertical above-ground system, dose rate locations on side of the cask and on the top of the lid may be needed. The locations on the side will be more representative for

the fuel in the periphery cell locations of the basket, while the dose rate on the top lid will be more representative of the contribution from the assemblies in the center of the basket.

b) Dose rates on the surface of the casks at local discontinuities such as inlets and outlets are less suitable. If the areas of these dose rates are small, they would not represent a significant contribution to any occupational or site boundary dose, hence the level of the dose rate at the location is of little relevance. Controlling such locations through individual limits could therefore unnecessarily restrict the contents, without any related safety benefits.

c) Removed.

For further guidance, see Section 2.6 of the main part of this report.

B.3 Area of Applicability

1) For fuel, the area of applicability shall be specified in the form of the list of assemblies and assembly types that can be loaded (or a reference to such a list), and maximum burnup, minimum cooling time, and any enrichment limits if applicable.

2) For the casks, the area of applicability may include limits of changes permitted to the systems, such as changes in dimensions, materials, or material densities.

B.4 Representative Contents

1) To demonstrate the overall performance details of the systems, doses and dose rates are presented in the FSAR, including dose rates in the vicinity of the cask at locations other than those specified as acceptance criteria, occupational dose rates during loading and unloading of the casks, and dose rates for selected cask arrays at selected distances from the array to demonstrate the system meets the requirements of 72.236(d), 72.104, and for calculations to demonstrate compliance accident dose rates under 72.106.

2) For these analyses, one or more representative contents shall be selected, such that the dose rates used as acceptance criteria are met at the respective locations. For any given location, the total dose rates are either dominated by gamma source terms (fuel gamma and ^{60}Co contribution), or by neutron source terms (neutron and n-gamma). The analyzed source distributions may therefore include the following:

   a) Low cooling time, and corresponding (low) burnup so the dose acceptance is reached. This will maximize dose in locations where gamma contribution dominates; or

   b) High burnup, and corresponding (longer) cooling time so the dose acceptance is reached. This will maximize dose in locations where neutron contribution dominates.

   These could be used as uniform conditions, or with different distribution in different areas. For example, a condition where fuel on the basket periphery use distribution a) and on the inside of the basket use distribution b) may result in slightly higher dose rates than uniform distributions throughout the basket, and hence be a better representative content.

3) The casks shall be analyzed with each selected representative content, and for each dose location the highest value shall be reported or utilized.

4) For accident conditions, all selected representative content shall be evaluated to ensure that the maximum accident does rate is identified. For example, for a transfer cask with water on the outside for neutron shielding, the accident could be the loss of this water. Under this accident condition, the source distribution that maximizes the neutron doses may be more bounding, even if the contribution that maximizes gamma dose is more bounding under normal conditions for the same cask.

B.5 Accident Conditions

For the conclusions from the calculations for accident condition, there are the following two options, and it should be clearly identified which of these apply:

- If the calculations under accident conditions with the representative content show that the requirements of 10 CFR 72.106 are met, then no further site specific accident calculations are needed, and that should be stated.
- If the calculations under accident conditions with the representative content show that the requirements of 10 CFR 72.106 are not met, then further site specific accident calculations are required, and that should be stated.

APPENDIX C EXAMPLE OF FSAR SECTION

This appendix contains an example for an Appendix added to the FSAR Shielding Chapter to utilize the method and framework outlined in this TR. Further to this addition, the TR shall be referenced in the corresponding CoC/TS. The example is based on the FSAR for the HI-STORM FW, where the shielding safety analyses are in Chapter 5 and Chapter 11. For consistency with the nomenclature of this appendix example, section and subsection numbers start with C.

This example only contains general discussions, where information for a specific system are shown as "TBD". Different or additional discussions may be necessary for any specific system.

C.1 RADIOLOGICAL QUALIFICATION OF CONTENT

Fuel needs to be clearly qualified so the regulatory requirements in 72.236(a) and (d) are met. This means for a given fuel assembly proposed to be loaded into a certain basket cell (including any non-fuel hardware (NFH) that may be present), a clear decision can be made if loading that assembly into that cell is permitted (qualified) or not. Since content is often defined as a pattern for an entire basket loaded with fuel, the qualification may also depend on the pattern, i.e., on the specification of other assemblies in the basket, not just on the parameters of the assembly proposed for that cell.

From a radiological perspective, the qualification needs to include the fuel assembly type, burnup, enrichment, and cooling time (BECTs), and specification of any other NFH. Note that decay heat requirements that are discussed in Chapter TBD are independent of the radiological qualification discussed here. Specifically, qualification of an assembly from a radiological perspective does not imply that fuel meets any decay heat requirements, and vice versa.

Two approaches for this radiological qualification of the content are implemented in this FSAR:

1. Fuel type and BECTs, and NFH characteristics, that are directly specified as tables or equations. These may be linked to, or depend on, some defined loading patterns. These are based on and supported by the analyses in the main part and other appendices of this shielding chapter, and those in the radiation protection chapter.
2. Using the method defined in the Topical Report [C.1], which allows to determine BECTs and NFH characteristics separate from this FSAR, but based one dose rate limits specified in this FSAR.

This appendix to the FSAR shielding chapter implements the second approach for content qualification. It defines and justifies the acceptance criteria (dose rate limits and corresponding locations), the area of applicability, and present results of dose and dose rate calculations consistent with the requirements in [C.1].

C.2. ACCEPTANCE CRITERIA

The dose rate limits that are to be used as acceptance criteria are listed in Table C.1 for the Transfer Cask and in Table C.2 for the HI-STORM. Bases and justifications for the locations and the values are as follows:

- Dose Locations:
  - Dose locations are selected so they capture not just the integral effect of all the content, but, to the extent practical and feasible, the contribution from individual assemblies.
  - Dose rates on the sides of the cask are more dominated by the contribution from the assemblies on the periphery of the basket, specifically for gamma dose rates. Contribution from assemblies closer to the center of the basket have a larger effect on the dose rates on the top and bottom of the casks. Therefore, to assure contributions from all assemblies are considered to some extent, dose locations both on the side and top are defined, and for the Transfer Cask also on the bottom of the cask.
  - Both overpacks have different shielding characteristics, hence dose performance of fuel in the Transfer Cask cannot reliably derived from the dose performance in the HI-STORM and vice versa. Hence separate limits are specified.
  - Due to its shielding characteristics, local dose rates around the Transfer Cask are more directly impacted by the potential presence of NFH in PWR assemblies. Therefore, additional dose locations are specified for the Transfer Cask for PWR fuel assembly to represent this impact. The qualification of the NFH is performed through the dose rates for these additional locations on the Transfer Cask, thus it is not necessary to specify such extra locations for the HI-STORM, where the impact of those would also be less pronounced.
  - All dose rates should be calculated on or near the surface, where the impact of individual assemblies would be maximized.
  - Calculating dose rates typically involves averaging values over certain surface areas rather than calculation values for point like locations. The minimum size of these areas is governed by the following:
    - TBD
  - Each principal dose location contains several areas, such as the circumferential sections as stated above. The maximum value calculated over these sections should be used for comparison with the corresponding limit.
- Dose Limits: These are generally selected to be consistent with the values presented in Section 5.1, and that are of the order that can be handled by the individual plants under their Radiation Protection program.

Note that these dose rate limits are independent from the dose limits stated in 10CFR72.104, and meeting above dose rate limits does not imply any of those are met.

C.3 AREA OF APPLICABILITY

All contents must meet the requirements stated in Table 2.2 of [C.1]. Additional requirements are listed in Table C.3

C.4. CALCULATIONAL MODELS

The models used for the dose rate calculations in this appendix to the FSAR shielding chapter are the same models used for the results presented in Section 5.1 and as described in Section 5.3 of this chapter, with the following shielding-specific parameters:

- Transfer Cask : TBD
- Storage Cask : TBD

For the dose locations that show the effect of the NFH for PWR fuel at the top and bottom of the active region, the following areas are selected:

- Dose Location A: TBD
- Dose Location C: TBD

C.5 REPRESENTATIVE CONTENT AND CORRESPONDING DOSE RATES

Representative content specifications have been developed, by selecting characteristics that result in dose rates at the relevant location that closely match the dose rate limits presented in Subsection C.2. This content and the corresponding calculated dose rates are presented in this Subsection. This content is then used in subsequent calculations presented in Subsection C.6., for example to show annual doses at the controlled area boundary.

The details for the developed representative content are as follows:

- Spent Fuel
    - Two uniform and one regionalized loading configuration are developed, with BECTs listed in Table C.4
        - The first uniform configuration is established to represent gamma dominated fuel. For this, a short cooling time is selected, and then the burnup is determined that results in dose rates consistent with the limits.
        - The second uniform configuration is established to represent neutron dominated fuel. For this, a high burnup is selected and then the cooling time is determined that results in dose rates consistent with the limits. Note that the selected burnup is slightly above the value listed in Table C.3. However, that does not affect the process and the principal conclusions.
        - A regionalized loading configuration is then selected where the gamma dominated fuel are on the periphery of the basket, and the neutron dominant fuel are closer to the center. This is to account for the fact that the dose rates on the side of the cask would get little gamma contribution from any neutron dominated fuel in the center, hence this regional loading combination may represent a more comprehensive impact of all fuel assemblies on the side of the cask.
- NFH
    - Since a fuel assembly can only contain a single NFH device, several variations of NFH content are analyzed, in addition to the case where no NFH is present in any assembly. In all cases, the source terms of the NFH are the same for all devices in a cask, and independent from those of the fuel assemblies.

It is important to note that none of these assumptions pose any additional limitation. As long as the calculated dose rates meet the limits, different fuel with lower cooling times or lower burnups, and NFH with different quantities and cooling times or source terms can be loaded.

Calculated dose rates for the locations listed in Table C.1 and Table C.2, and the content specified in Table C.4 are presented in Tables C.5 through C.16 as follows:

- Tables C.5 through C.7 show the results for the Transfer Cask without NFH for all loading configurations.
- Tables C.8 through C.10 show corresponding results with NFH insertion. only the NFH contribution and the total values are shown in each case.
- Tables C.11 through C.16 show the same as Tables C.5 through C.10, but for the Storage Cask.

A review of the various tables shows that highest dose rate at any location is close to, or exceeds the selected limit shown in Table C.2 and Table C.3, showing that the goal of the selection of the representative content is met.

C.6 OTHER DOSE CALCULATIONS

The representative content is then used in other dose calculations to demonstrate the shielding capabilities of the systems, with results presented at the end of this Appendix. The following cases are analyzed.

Dose rates consistent with those presented in Section 5.1 for normal conditions are presented in Table C.17 for the Transfer Cask and Table C.18 for the Storage Cask.

Dose rates for the accident condition for the Transfer Cask are shown in Table C.19. These meet the requirements of 10 CFR 72.106, hence no further accident dose analyses are required on a site-specific basis.

Annual doses for cask arrays at a distance, consistent with the results presented in Table 5.TBD are listed in Table C.20.

With respect to the occupational dose rates (see Chapter TBD), TBD.

C.7 CHANGES TO CASK DESIGNS

The following requirements apply to changes made to the cask designs under 10CFR72.48. If these requirements are not met, the methodology in [C.1] can not be used to qualify the content for the casks.

TBD

C.8 SUMMARY

This appendix establishes dose limits, dose locations and area of applicability to qualify fuel using the topical report in [C.1]. It also clarifies which changes to the transfer and storage cask can be made under 72.48 to remain consistent with [C.1].

C.9 REFERENCES

[C.1] Topical Report on the Radiological Fuel Qualification Methodology for Dry Storage Systems, Holtec Report HI-2210161, Revision 3.

Table C.1 Dose Rate Limits for the Transfer Cask

Dose Rate Location		Dose Rate Limit (mrem/hr)
A	TBD	TBD
B	TBD	TBD
C	TBD	TBD
D	TBD	TBD
E	TBD	TBD

Table C.2 Dose Rate Limits for Storage Cask

Dose Rate Location		Dose Rate Limit (mrem/hr)
F	TBD	TBD
G	TBD	TBD

Table C.3 Applicability requirements, in addition to Table 2.2 in [C.1]

Parameter	Requirement	Basis or Justification
Maximum assembly average burnup for PWR (GWd/mtU)	TBD	TBD
Maximum assembly average burnup for BWR (GWd/mtU)	TBD	TBD
Changes to the Transfer or Storage Cask Design	TBD	TBD

Table C.4 Loading Configurations for Representative Content

Configuration	Fuel Specification, Burnup / Enrichment / Cooling time, (GWd/mtU / wt% / years)	
	Basket Periphery	Basket Center
Uniform 1	TBD	TBD
Uniform 2	TBD	TBD
Regionalized	TBD	TBD

The remaining pages need to show the tables with dose rates according to the list in Section C.5 and C.6

APPENDIX D DELETED

This appendix presents the required approach and format of documenting the qualification, using Example 3 from Appendix A. This form consists of a summary table providing all the relevant information, with appendices for information that would not fit into the table. The calculational details are not included and other calculation reports are referenced instead. A blank template with additional guidance is included as Appendix F to the report.

Parameter	**Requirements outlined in Topical Report**	**Qualified Value(s) or Parameter(s)**	**Justification or Reference**
Storage Cask			
Cask Systems	Specification of the casks systems that the fuel is qualified for	Storage CASK A Transfer Cask A 32 assembly basket	[1], [2]
Shielding Design Changes and Site Specific Parameters	Shielding design can be changed via 72.48, some designs have variable thickness transfer casks, however changes and/or site-specific parameters must be included in the shielding analyses used as a part of the FSAR method for dose rate calculations	The following design parameters for the cask system were modified for the qualification documented here: - Storage Cask: TBD - Transfer Cask: TBD	[5]
Allowable content definitions			
Fuel assemblies and characteristics that can be loaded	Allowable fuel assemblies are in Table 2.2 of the topical; FSAR will have more specific allowable fuel assembly characteristics; different masses of assemblies may be analyzed as long as the same mass is used in the source term and dose rate analyses	All PWR fuel assemblies specified in [1] and [2].	[1], [2]
Fuel Hardware	Section 3.2 of the topical states that if source term does not consider Inconel spacers then qualification must be restricted to fuel without them	Fuel assemblies do not contain Inconel grid spacers, hence these were not considered. This qualification report can therefore not be used for fuel	[3]

		that contains Inconel grid spacers.	
Fuel Conditions	Damaged or reconstituted fuel is allowed however the method for modeling this fuel is not part of the topical	Fuel to be loaded may include damaged or reconstituted fuel. Locations for such fuel are governed by different requirements in the corresponding CoC	[1]
Inserts/non-fuel hardware (NFH) that can be loaded (PWR)	Allowable NFH are in Table 2.2 of the topical; FSAR will have more specific allowable fuel assembly characteristics; masses are in tables 3.4, 3.7 and 3.8; however different masses may be used, and if different masses are used this must be stated and loaded NFH are restricted to these masses	BPRAs, TPDs and CRAs are qualified through this report. There is no indication that any of the NFH exceed the masses in [3]. The locations and maximum number for each NFH are specified in Appendix B to this report. Burnup and cooling time limits are specified in Appendix A to this report	[3]
Co-59 impurity levels in NFH	For assemblies manufactured before 1990, a higher Co-59 impurity level in Stainless Steel and Inconel must be used than for assemblies manufactured after that.	Different Co-60 activation levels were used in the dose analysis depending on the cooling time of the assembly. Since the manufacturing date is not always known, the higher levels are conservatively used for all assemblies discharged on or after 2020.	[4]
CRA Insertion	If there is indication that any CRA was inserted into the core more than 10% during power operation, then the radiation transport model needs to be adjusted for that insert as discussed in Section 3.4.2 of the topical report, and the source term used in the dose	There is no indication of any insertion larger than 10%, hence the standard parameters were used in the dose analyses.	[4]

	calculation for that insert also needs to be adjusted as discussed in that section of the topical report. If the insertion of 10% is not exceeded, the standard parameters in Table 3.7 are used.		
Neutron source assembly (NSA; PWR only)	Three options are explained within Section 3.4.3 of the topical. (1) no limit to NSAs if source is determined to be negligible, (2) quantify and consider NSA source in calculation and number and location of NSAs is part of the content, (3) perform no evaluation and NSA are limited to 1 at the center of the basket	Only a single NSA is permitted in each basket. hence no additional analyses are required.	[3]
Burnup/enrichment/cooling times and loading patterns	Can vary based on qualified content, maximum burnup allowed is 72 GWd/mtU for both PWR fuel and BWR fuel; enrichment range is 0.5 wt% to 5.0 wt% ^{235}U; cooling time is greater than or equal to 1 year	Fuel loading patterns are shown in Appendix B to this report. Burnup, enrichment and cooling time are shown in Appendix A to this report.	[4]
Specific Power	If the specific power of an assembly averaged over its entire operation in the core exceeds the value in Table 3.1 or 3.2 of the topical, then the increased specific power needs to be used in the source term calculations for that assembly. Otherwise, the specific power in Table 3.1 or 3.2 is used.	All assemblies to be qualified were operated at specific powers below the value listed in Table 3.1, hence the specific power in that table is used in the source term analysis	[4]
Analysis Method			

Design Basis Assembly	Topical gives the option of using assembly from Table 3.1 or 3.2 of the topical, this needs to be consistent with what is used in the FSAR	The WE 17x17 from Table 3.1 of the TR [3] is used.	No justification needed
SCALE Code Version	Topical allows for newer version of the SCALE code system than 6.2.1. If newer versions of the code are used, topical requires a comparison per section 3.1 must be performed	Source term calculations are performed with SCALE Version 6.2.1	[4]
Gamma/neutron group structures	Gamma and neutron group structure is documented in Tables 3.5 and 3.6 and is allowed to change slightly per Section 3.2 and 3.3 of the topical	Group Structures from Section 3.2 and 3.3 of the TR are used in the source term analyses and the radiation transport calculations	No justification needed
Results			
Acceptance criteria	Dose rates must meet acceptance criteria as established in transport method defined in FSAR	Acceptance criteria were taken from FSAR/TS [1],[2]. Calculations are documented in [4]. The comparison is presented in Appendix C, and shows that all criteria are met.	[4], Appendix C
Justify acceptance criteria is valid if there are design changes	FSAR will include criteria/method for demonstrating that acceptance criteria dose rate points are still valid if there are design changes from FSAR version where these were originally approved	The FSAR specified that the acceptance criteria are applicable for a range of design variations. The design variations used here are within those ranges.	No justification needed

References

[1] Storage CoC

[2] Storage FSAR

[3] Topical report HI-2210161

[4] Calculation Report

[5] 72.48 evaluations

Appendix A: Fuel Qualification Tables (Fuel and NFH)

Fuel:

Region (See Appendix B)		1	2	3	4	5
Maximum Burnup	Minimum Enrichment	Minimum Cooling Time (Years)				
5000	1.1	2.25	1.5	1.25	1	1
10000	1.1	3.5	2.5	2	1.5	1
15000	1.6	4	3	2.5	2	1
20000	1.6	6	3.75	3	2.5	1
25000	2.4	9	4	3.5	2.75	1.25
30000	2.4	17	5	4	3.25	1.5
35000	2.9	22	7	4.5	3.5	1.75
40000	3.2	30	9	5	3.75	1.75
45000	3.6	43	11	6	4	2
50000	3.6	51	16	8	5	2.25
55000	3.9	57	21	10	6	2.25
60000	3.9	n/a	27	14	7	2.5
65000	4.5	n/a	31	17	9	2.75
70000	4.5	n/a	36	22	11	3

NFH:

Minimum Cooling Time (Years)	Maximum Burnup, MWd/mtU	
	BPRAs	TPDs, NSAs, CRAs, APSRs
3	30,000	180,000
10	50,000	630,000

Appendix B: Loading Pattern

	5	3	3	5	
5	4	2	2	4	5
3	2	1	1	2	3
3	2	1	1	2	3
5	4	2	2	4	5
	5	3	3	5	

NFH is permitted in the following regions:

BPRAs, TPDs	All regions
CRAs	Regions 1 and 2
NSAs	1 NSA per cask, in Region 1

Appendix C: Acceptance Criteria and Results of Dose Analyses

Table C.1. Dose Comparison

Dose Location (see Reference [2])	Maximum Calculated Dose Rate, mrem/hr	Dose Rate Limit for fuel qualification, mrem/hr ([1],[2])
Storage Cask		
1	100	200
2	200	300
3	200	300
4	30	100
Transfer Cask		
1	500	800
2	600	900
3	500	600
4	50	100

Note that these are arbitrary values for illustrative purposes, not the results of any specific calculation.

Dose rate calculations are based on the Option 2 from the approach outlined in Section 2.6 of [3]. The burnups and cooling times that result in the highest dose rate at each dose location and that are used to calculate the dose rates listed above, are shown in Table C.2. The table also shows the bounding NFH burnup/cooling time for the highest dose rate in each region.

Table C.2 Burnup Combinations corresponding to Maximum Dose Rates

Region (See Figure in Appendix B)	1	2	3	4	5
Dose Location	Burnup (Assembly / NFH) (for corresponding enrichment and cooling times see Appendix A)				
Storage Cask					
1	15000 / 30000	25000 / 30000	10000 / 30000	10000 / 30000	60000 / 30000
2	...	...	...		
3					
4					
Transfer Cask					
1	...	...	...		
2					
3					
4					

Required content of the qualification report

Parameter	Topical Requirement – information in this column does not change and is to be carried over into the qualification report for reference use	Qualified Value(s) – Information in this column is input by the user; they must state values that are applicable to this qualification report; if a table (such as BECT) include pointer to this information	Justification – Information in this column is input by the user; they must add a reference to where appropriate justification (as required by topical) such as calculation file, etc.
Storage Cask			
Cask Systems	Specification of the casks systems that the fuel is qualified for	Specify cask and basket type(s) to be qualified	Reference FSAR an CoC. They must include the reference to the topical report
Shielding Design Changes and Site Specific Parameters	Shielding design can be changed via 72.48, some designs have variable thickness transfer casks, however changes and/or site-specific parameters must be included in the shielding analyses used as a part of the FSAR method for dose rate calculations	include reference FSAR, differences in design and/or site-specific parameters that deviate from the design basis FSAR that have been incorporated into this qualification	Provide references to applicable 72.48 reports or updated FSARs as appropriate
Allowable content definitions			
Fuel assemblies and characteristics that can be loaded	Allowable fuel assemblies are in Table 2.2 of the topical; FSAR will have more specific allowable fuel assembly characteristics; different masses of assemblies may be analyzed as long as the same mass is used in the source term and dose rate analyses	Include reference to TS/FSAR and Table 2.2 of the topical that include allowable assemblies; if there are restrictions associated with an assembly used as an analysis parameter (i.e. if a different/lower assembly mass	Include a reference to the calculation report documenting different mass used for source term and dose rate calculations

		is used) that needs to be stated here	
Fuel Hardware	Section 3.2 of the topical states that if source term does not consider Inconel spacers then qualification must be restricted to fuel without them	State if fuel with Inconel spacers is allowed and if they were considered in the source term evaluation	Include a reference to the calculation report documenting how Inconel spacers was considered within source term
Fuel Conditions	Damaged or reconstituted fuel is allowed however the method for modeling this fuel is not part of the topical	State if and how much damaged and reconstituted fuel is allowed and include a loading pattern(s)/locations that include this information (could be an appendix to this table)	Include reference to Include reference to calculation file for source term and dose rate
Inserts/non-fuel hardware (NFH) that can be loaded (PWR)	Allowable NFH are in Table 2.2 of the topical; FSAR will have more specific allowable fuel assembly characteristics; masses are in tables 3.4, 3.7 and 3.8 of topical; however different masses can be used, if different masses are used this must be stated and loaded NFH are restricted to these masses	If PWR, include inserts/NFH that are allowed to be loaded; reference topical or FSAR as appropriate, and/or include allowable masses of inserts if different	If NFH mass is different from topical assumptions, include a reference to the calculation report documenting different NFH mass used for source term calculations
Co-59 impurity levels in NFH	For assemblies manufactured before 1990, a higher Co-59 impurity level in Stainless Steel and Inconel must be used than for assemblies manufactured after that.	State if assemblies manufactured before 1990 are to be qualified, with the higher Co-59 impurity levels.	Include reference to calculation report that shows correct application of the impurity levels
CRA Insertion	If there is indication that any CRA was inserted into the core more than 10% during power operation, then the radiation transport model needs to be adjusted for that insert as discussed in Section	State if CRAs with insertion exceeding 10% are qualified. If so, include maximum insertion level and the corresponding masses that are considered	If insertion exceeds 10%, include reference to the calculation file where the details are documented.

	3.4.2 of the topical report, and the source term used in the dose calculation for that insert also needs to be adjusted as discussed in that section of the topical report. If the insertion of 10% is not exceeded, the standard parameters in Table 3.7 are used.		
Neutron source assembly (NSA; PWR only)	Three options are explained within Section 3.4.3 of the topical. (1) no limit to NSAs if source is determined to be negligible, (2) quantify and consider NSA source in calculation and number and location of NSAs is part of the content, (3) perform no evaluation and NSA are limited to 1 at the center of the basket	State which option is selected and if option (2) include the allowable number and location of NSAs	If (1) is selected, provide reference to justification that source is negligible, if (2) is selected provide reference to analyses showing source term and compliance with dose rate limits
Burnup/enrichment/cooling times and loading patterns	Can vary based on qualified content, maximum burnup allowed is 72 GWd/mtU for both BWR and PWR fuel; enrichment range is 0.5 wt% to 5.0 wt% ^{235}U; cooling time is greater than or equal to 1 year	Include allowable FQT or burnup/enrichment/cooling times (can be appendix to this table)	Include reference to calculation file for source term and dose rate
Specific Power	If the specific power of an assembly averaged over its entire operation in the core exceeds the value in Table 3.1 or 3.2 of the topical, then the increased specific power needs to be used in the source term calculations for that assembly. Otherwise, the specific power value in Table 3.1 or 3.2 is used.	State if the fuel assemblies with the specific power exceeding the value in Table 3.1 or 3.2 are qualified. If so, include the specific power that is considered	If the specific power exceeds the value in Table 3.1 or 3.2 of the topical, include reference to the calculation file where the details are documented.

Analysis Method			
Design Basis Assembly	Topical gives the option of using assembly from Table 3.1 or 3.2 of the topical, this needs to be consistent with what is used in the FSAR	State here which design basis assembly was chosen for source term and transport calculations	No justification needed
SCALE Code Version	Topical allows for newer version of the SCALE code system than 6.2.1. If newer versions of the code are used, topical requires a comparison per section 3.1 must be performed	State here which version of SCALE is used to perform source term calculations	If SCALE version is a newer than 6.2.1, provide reference to documentation of comparison per Section 3.1
Gamma/neutron group structures	Gamma and neutron group structure is documented in Tables 3.5 and 3.6 and is allowed to change slightly per Section 3.2 and 3.3 of the topical	State if the group structures from the topical have been used or state what the group structures are and if they are different; similar to BECT, can be after this table in an appendix to this report	Include reference to calculation file that includes justification of different group structure
Results			
Acceptance criteria	Dose rates must meet acceptance criteria as established in transport method defined in FSAR	Include comparison to acceptance criteria (similar to FQT, won't fit in this box so may include as appendix, Appendix XYZ to this table, etc.)	Reference calculation file with dose calculations
Justify acceptance criteria is valid if there are design changes	FSAR will include criteria/method for demonstrating that acceptance criteria dose rate points are still valid if there are design changes from FSAR version where these were originally approved	If there are design changes, include results of criteria/method used to demonstrate acceptance criteria dose points are still acceptable	Reference calculation file with dose calculations

Example Appendices

Appendix A: FQT

[include FQTs with burnup/enrichment/cooling time]

Appendix B: Loading Pattern

[Include loading pattern, locations of damaged fuel, inserts, etc.]

Appendix C: Group structures

[Include gamma and neutron group structures if different from topical]

Appendix D: Dose Rates compared to Acceptance Criteria

[Include results of dose rate calculations demonstrating that acceptance criteria has been met]