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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.

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). 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 overarching requirement, 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 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 approach 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 was 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 the same as that 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. 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 D contains an example of such a qualification report with a format and content that should be followed for every actual qualification report, with any deviation justified. 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 point 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 any given set of contents meet the acceptance criteria are documented in qualification reports. These reports define the contents to be qualified, define the system

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

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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 D contains a principal example of such a report for a selected storage system and content.

2.4 Design Basis Assemblies

The design basis fuel assemblies taken from [1] and [2] and specified in Tables 3.1 and 3.2 shall 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.

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. 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. 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. 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 consists 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 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 that would be needed for a completely heterogeneous loading.

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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 specified.

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. Any fuel assembly that is outside any of the parameters or parameter ranges specified in Table 2.2 cannot be qualified through this TR.

Some of the requirements in Table 2.2 are straight forward and 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 does also 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 in [9]. Nevertheless, acceptable parameters or parameter ranges are defined in the Table, 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 subsection discuss the basis for other parameters. If necessary, these discussions also provide the basis for the value or values listed in Chapter 3 that should be used in the analyses.

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Deleted: no limits or restrictions are specified in the corresponding FSARs. This is consistent with the studies presented in [9], which considers these of low importance and characterized them as "unrestricted".

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 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 each 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 as to not negatively affect the plant efficiency and stability. The allowed insertion is tightly controlled by the plant operating procedures. Hence these can also be considered not significant enough to be modeled for the source term operation.

Overall, there is no need to consider any of the NFH for the in-core-operation, and hence all NFH permitted for storage are also permitted to be be present in fuel assemblies during power operation.

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. 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

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2.8.5 Water Density BWR

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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 [11] shows that it can vary from about 0.1 to 0.75. 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 listed in [9] is defined for the source term analyses here

2.8.6 Fuel Density

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Typically, fuel densities are kept close to the theoretical maximum of 10.96 g/cm2, 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/cm2. The density affects the fuel to water ratio, and in this case, a higher value would be more important 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 96% 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

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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 BWR fuel. These are realistic and typical for most assemblies, but individual assemblies in a core may have slightly higher values, specifically for assemblies irradiated for just a limited number of cycles before the plant shutdown. Hence in order to not exclude such assemblies, higher upper bound values are specified in Table 2.2 For the design basis assembly calculations (Table 3.1), the values from [1] and [2] are considered appropriate.

2.8.8 Fuel (Uranium) Mass

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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. 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.

Some of the fuel conditions not undamaged may have uranium weights lower than that of an undamaged assembly. When considering this in both the source term and radiation transport analyses, this 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.

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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

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
Non-Fuel-Hardware for PWR assemblies	Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, instrument tube tie rods (ITTRs), vibration suppressor inserts, and components of these devices such as individual rods. Activated material in the NFH may be zirconium alloys, steel, Inconel, or AgInCd (in CRAs)	[1], [2], and Section 3 of this report
Enrichment	0.5 wt% to 5.0 wt% ^{235}U (Enrichments may be axially constant or there maybe axial enrichment variations, e.g., assemblies may have axial blankets)	[1], [2] and range of TRITON Libraries [8]. For enrichment variations see Section 3.1 of this report
Cooling Time	Greater or equal to 1 year	10CFR72
In-Core Cycle Average Soluble Boron	0 to 2000 ppm	[10] See Section 2.8.1 of this report

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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	BWR 0.1 to 0.75; PWR 0.7 to 0.75	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 53 MW/mtU for PWR, up to 40 MW/mtU for BWR	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 200 kg; PWR up to 500 kg	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 that the FSAR that this TR is referenced in may specify more restrictive values for some parameters

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3.0 SOURCE TERM EVALUATIONS FOR QUALIFICATION OF FUEL

3.1 General

This section specifies the requirements 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 those 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 libraries supplied with the respective code version. The calculations shall be utilizing data libraries with the maximum number of energy groups available for the respective code version. For SCALE 6.2.1, this is the 252-group library based on ENDF/B-VII.1 nuclear data. ORIGAMI and TRITON 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 A), 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.

Source term calculations shall be performed for the design basis assemblies listed in either Table 3.1 or Table 3.2, which have been shown in [1] to reasonably bound all assembly types in the corresponding FSARs.

Some assemblies contain axial blankets, i.e., small sections at the top and bottom of the assemblies that have reduced enriched or natural uranium. The assembly average burnup and enrichment used in the source term calculation can be either calculated with or without including those blankets. However, if the average burnup is calculated by including the blankets, then the corresponding enrichment must also be calculated by including the blankets

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ORIGAMI calculations have several options and parameters, and these shall be used as follows:

- TBD

3.2 Fuel Assembly 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.

<u>Gamma Source from Active Fuel Region</u>

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 documentation to qualify 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.

<u>Gamma Source from Activation of Structural Materials in the Fuel</u>

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. Additionally, assemblies may include irradiated steel rods that have been inserted to replace fuel rods with damaged cladding, or that have been part of the initial fuel assembly design. If these 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 1.0 gm/kg impurity level for fuel manufactured in or after 1990. It is recognized that materials used in earlier assemblies may have

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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.

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].

The calculations are then to be performed with the following steps:

1) The activity of the ^{60}Co is calculated using ORIGAMI. The flux used in the calculation is the in-core fuel region flux at full power.
2) The activity calculated in Step 1 for the region of interest is then modified by the appropriate scaling factors listed in Table 3.3.

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 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 used for the NFH is the burnup of fuel assembly that the NFH was present in. If the NFH had been present in several assemblies, as would have been typically the case for TPDs, CRAs, APSRs, and NSAs, the burnup used for the NFH is the addition of the burnups of all assemblies it was located in. In the

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source calculation, this should be modeled as a sequence of exposures, with a reset of the assembly burnup every 45 GWd/mtU.

The specifications in the tables at the end of this chapter are considered reasonable and mostly conservative for determining source terms for typical fuel assemblies and NFH, specifically with respect to the amounts of irradiated material. If there are any indications that the amounts of components to be qualified are significantly higher from those, the differences need to be evaluated, and source terms need to be adjusted accordingly.

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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. The stainless steel clad BPRAs create a significant radiation source (^{60}Co) while the zircaloy clad BPRAs create a negligible radiation source. Therefore, the stainless steel clad BPRAs are bounding.

In general, the radiation source term for the TPDs and BPRAs shall be determined in the same way as the structural materials in the fuel assembly. In the calculations the ^{59}Co impurity level shall be assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for Inconel. The calculations shall then be performed by irradiating the appropriate mass of steel and Inconel using the flux calculated for the design basis or specific fuel assembly. For TPDs which can be repeatedly placed into fuel assemblies, the flux level shall be restarted every time a burnup of the assembly of 45 GWd/mtU is reached. The mass of material in the regions above the active fuel zone shall then be scaled by the appropriate scaling factors listed in Table 3.3 to account for the reduced flux levels above the fuel assembly. It is recognized that materials used earlier may have had higher impurity levels, mainly because the detrimental effect of ^{60}Co on the dose rates around

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casks was not appropriately recognized. However, these materials would now have undergone significant decay and corresponding reduction of the ^{60}Co source terms, and more recently utilized materials have been typically been selected to have lower ^{59}Co content. The values prescribed here are therefore considered appropriate.

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. Alternatively, lower masses can be used with appropriate reference and documentation. Note that since inserts are managed, handled and can be qualified separately from the fuel, the burnup and cooling time of an insert in an assembly may be different from that of the assembly.

Previous analyses (see Reference [1]) have indicated that dose effects from BPRAs are generally bounding. However, since they may have separate and different acceptance criteria with respect to burnup (this is the burnup of the fuel assembly that they were inserted in while being in the core) and cooling time, they both need to be considered separately from each other to qualify them with the respective parameters.

Note that the mass of BPRAs or TPDs shall not be considered in the radiation transport analyses as additional shielding. This is a conservative approach since it neglects material that would provide some additional shielding.

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, 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. CRAs are fabricated of 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.

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. These are based on a review of publicly available fuel information and are considered reasonable and sufficiently conservative. In the calculations a cobalt-59 impurity level of 0.8 gm/kg shall be used for stainless steel and 4.7 gm/kg for inconel.

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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 (i.e. adjust mass of irradiated material), and the corresponding radiation transport analyses (adjust length of insertion).

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In all cases, ORIGAMI calculations shall be performed where a standard amount of the respective material (^{60}Co or AgInCd) is added to a calculation of the corresponding source terms, based on the burnup that the components were exposed to in the fuel assemblies in the core. Since both CRAs and APSRs are present for a longer exposure time than a single assembly, the assembly flux shall be restarted every 45 GWd/mtU of the assembly. The final activity levels to be used in the dose calculations is then based on the ORIGAMI results, the mass of the component and the corresponding flux factor.

Since the level of activation of CRAs and APSRs can vary, the quantity that can be stored in an MPC is typically limited. As for BPRAs and TPDs, the qualification shall be consistent with the loading of those components, so these components are only loaded in locations specifically considered in the qualification.

Additionally,, the masses of those components shall not be considered in the radiation transport analyses as additional shielding, which is conservative.

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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.

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During in-core operations, the stainless steel and Inconel portions of the NSAs become activated, producing a significant amount of ^{60}Co. 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. The design shall be reviewed to identify the appropriate source distribution that should be considered in the radiation transport analysis.

Note that the 60 Co source from structural components of NSAs may put additional restrictions on the number and placement of NSAs, in addition to any restrictions based on the neutron source strength discussed below. In that respect, the top of the NSA is to be considered identical to a TPD, and the portion of the NSA in the active region shall be considered to be similar to a BPRA, but with appropriately reduced number of full-length rods.

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 it an NSA present in all assemblies provides less than 1% of the total neutron source term of a cask.
- 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].

3.5 Determination of Cobalt Sources for Fuel and Non-Fuel Hardware

The determination of the cobalt sources to be used in the radiation transport analyses shall follow the following steps:

- Perform an ORIGAMI calculation for the fuel assembly, with a reference amount of ^{59}Co added, e.g. 1 gram.
    - For inserts that may be in the core for an extended period of time, the fuel burnup is to be restarted every 45 GWd/mtU.
- For each component, scale the result of the first step to the corresponding mass listed in Tables 3.1, 3.2, 3.4, 3.7 or 3.8, as applicable.
- Then, for each component, scale the result with the applicable factor from Table 3.3, Table 3.7 or Table 3.8, as applicable

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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.

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

- Define what system, fuel and BECTs is to be qualified
- 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 differences.
  - if any parameters are outside of the applicability of the TR, then the method in the TR can not be used to qualify 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 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 Table 2.2, in combination with any more specific requirements form 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.

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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, 3.5 or 3.5, as applicable
- Perform the ORIGAMI calculations with the TRITON libraries, for the fuel to be qualified.

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- See Section 3.1 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. Changes are limited to as described in Appendix B, Section B.1 and 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.
- Confirm that the dose rate locations conform to the FSAR/TS requirements.

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- 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 Appendix B, Section B.4 and example in Appendix C, Section C.4)
- 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. Typically, there will be a range of BECTs, so the maximum dose rate over all of those BECTs needs to be established.
- 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 the qualification report. Pre-existing calculations for source terms and/or radiation transport calculations should not be repeated, but referenced appropriately.

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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] ML21258A368 (appears not to be publicly accessible on Adams)

[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.

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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. These examples provide the templates of the evaluations that need to be performed for any fuel to be qualified through the approach in this TR. An example of a qualification report is shown in Appendix D, based on Example 3 shown below.

A.2 Example 1, Generic Fuel Qualification

The principal steps are as follows:

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Step A: 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.

Fuel Types: W17x17, BW15x15

Step B: Define Acceptance Criteria

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

Step C: Perform Source term analyses for all fuel types, and BECTs, consistent with the methodology in Section 3.0 of this TR. Select the BW 15x15 assembly as design basis assembly since it bounds the W17x17 in terms of fuel weight

Step D: 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.

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 A: 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

Step B: Define Acceptance Criteria

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

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

Step D: 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 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 A: 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

Fuel Types: W17x17

Step B: Define Acceptance Criteria

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

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

Step D: 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.

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
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.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
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.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.

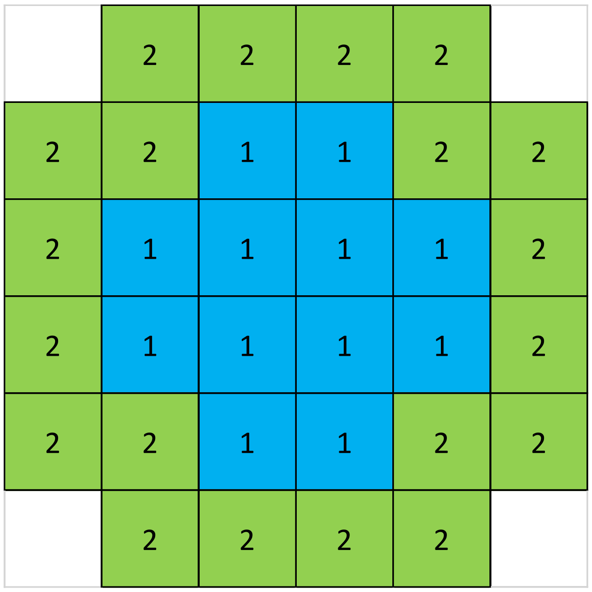

Figure A.1 32 Assembly Basket Layout with the Region Number identified in each Cell for Example 1

	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	

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) Fuel conditions other than undamaged fuel may need additional considerations with respect to their spatial distribution of the material and the applied source term.
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, 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). Hence one of two source distributions would result in a representative and conservative dose rates:

   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.

3) For each dose rate analysis with representative content, both conditions (items a) and b) stated in previous bullet) shall be analyzed, and for each dose location the higher value shall be reported or utilized.

4) For accident conditions, both source distributions 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

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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.

APPENDIX C EXAMPLE OF FSAR SECTION

This appendix contains an example for a Section added to an FSAR 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. However, for consistency with the nomenclature of this appendix, section and subsection numbers start with C.

C.1 Radiological Qualification of Content

This subsection discusses the two ways the content of the cask, i.e., the fuel assemblies, can be qualified. The qualification discussed here includes burnup, enrichment and cooling time (BECTs) of the fuel assemblies, and certain other parameters. Decay heat requirements are not part of this subsection, they are discussed in Chapter 4, and are independent of the discussions presented here. Specifically, the qualification process specified here does not imply that fuel meets any decay heat requirements, and vice versa.

Fuel needs to be clearly qualified so regulatory requirements in 72.236(a) and (d) can be met. That means that for a given fuel assembly proposed to be loaded into a certain basket cell, a clear decision can be made if loading that fuel 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 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.

Two alternative approaches are specified in this FSAR to perform this qualification:

1. BECTs are directly specified in the approved content section of the technical specifications. They can be specified as tables or as equations, linking providing a relationship between the BECTs, and these can vary between loading patterns. These are based on and supported by the analyses presented in this chapter, including dose rates presented in Section 5.1 around the casks, and for the possible locations at the controlled area boundary.
2. A method defined in a topical report is used to define and qualify the content for a given cask. The results of the process (i.e. the tables or equations) are documented in a separate qualification report. But the process relies on technical details documented in this FSAR.

The remainder of this subsection addresses all technical details that are needed and important for the second approach stated above. It addresses the modeling, acceptance criteria, and area of applicability. Some of the details and limits are included in the technical specification, either by repeating values in there, or by including parts of this subsection by reference into the technical specification.

C.2 Acceptance Criteria

The acceptance criteria are dose rates, and dose rate limits are defined in this subsection and specified in the TS. Limits are specified for both the HI-TRAC transfer cask and the HI-STORM overpack. Dose rates around the HI-TRAC are typically higher, and hence more important from an ALARA perspective during cask loading operations, while dose rates around the HI-STORM are more important for storage operations and site (e.g. owner controlled area) boundary. For certain design combinations, dose rates from the HI-TRAC may be more limiting, whereas for others dose rates form the HI-STORM may be more limiting. For that reason, dose rate limits for both systems are defined, and need to be independently confirmed for any given content.

For each system, dose rate limits are defined separately for the side and the top of the cask since top and side have a different relevance from an operational perspective.

Dose rates can exhibit significant variations across a given surface, due to design details of the cask and the characteristics of the content such as axial source distribution and loading pattern. Consequently, a sufficient number of dose locations have to be defined to capture the highest dose rate.

Additionally, a minimum area is specified for the maximum dose rate. This maximum dose rate is determined as an average over an area of no more than about TBD ft^2. Larger areas would possibly mask the local effect of fuel content, such as the effect of fuel distribution throughout the basket, whereas smaller areas would possibly shift the importance to local effects of the cask design rather effects of the content. Note that this selection is based on the need to determine a clear and unambiguous acceptance criteria for the content. If there are local discontinuities in the cask design that result in higher local does rates for smaller areas, these need to be considered by the RP personnel and taken into account for loading and other operations.

C.2.1 HI-TRAC

The different areas of the HI-TRAC are the side and the top of the cask.

The limits for these areas are selected as follows:

- Side
  - Maximum TBD rem/hr
- Top
  - Maximum TBD rem/hr

The dose rate limits were selected to be comparable to, but slightly lower than those dose rates calculated before and documented in Subsection 5.1.

C.2.2 HI-STORM

The different areas of the HI-STORM are the side and the top of the cask.

The limits for these areas are selected as follows:

- Side
  - Maximum TBD mrem/hr
- Top
  - Maximum TBD mrem/hr

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As for the HI-TRAC, dose rates are selected based on, but slight lower then shown in Subsection 5.1.

Note that for the HI-STORM, this is only part of the necessary dose considerations, and additional dose limits exist based on the need to show compliance with 72.104. These are often more restrictive than the general limits from the qualified content defined here, but are specified for the entire ISFSI and compliance is demonstrated on a site-specific basis.

C.3 Calculational Models

The calculations to show compliance with the above dose rate limits shall be those described in Section 5.3 of this chapter, but the following changes or adjustments are permitted:

- Thicknesses of the main materials relevant for shielding, i.e. steel, concrete, lead and water, can be changed, i.e., increased or reduced.
- Concrete density can be increased or reduced.
- Overall height of the casks can be changed, i.e., increased or reduced.
- Modifications to inlet or outlet air paths.

The following changes are not permitted:

- Introduction of shielding materials not currently used.
- Reduction of the level of detail in modeling specific design details, such as homogenization of materials beyond what is currently applied.

Note that any change or adjustment has to be validated against 10 CFR 72.48 and all other safety requirements, not just shielding.

For any changes that are made under 10 CFR 72.48, it must also be verified that dose locations are still meet with the requirements stated in Section $$ of the topical report [C.1], or those location need to be revised accordingly

Overall, the models described in Section 5.3, and required to be used for the shielding calculations to qualify fuel, meet the guidance in Appendix B of the topical report [C.1]. Specifically,

- They model the geometry with sufficient detail, i.e., without any significant simplifications of the geometry.
- They use MCNP, a state-of-the-art Monte-Carlo program
- Inputs and outputs for the airflow path in the HI-STORM, which are main concerns from a streaming perspective, are modeled accurately.
- Fuel assemblies are modeled with separate axial sections, one of them for the active region, each using a homogenized material.
- Uncertainties of the results are generally of the order of 5% or less.
- Fuel masses for self-shielding are less or equal to those used in the source term analyses.
- The axial burnup profile is considered in the source term definition.

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Area of Applicability

Qualification is limited to the fuel assemblies that are explicitly listed in the TS, and the following burnup, enrichment and cooling time limits:

- Maximum assembly average burnup
  - PWR assemblies – 68.2 GWd/mtU
  - BWR assemblies – 65 GWd/mtU
- Minimum cooling time
  - TBD years
- Minimum enrichment
  - TBD

From a cask perspective, the qualification is limited to the HI-TRAC and HI-STORM in this FSAR, with modification permitted as discussed before in this subsection.

C.4 Other doses and dose rates

In this subsection, doses and dose rates for other dose location and other conditions (i.e., not for dose rate limits stated in the previous subsection) are evaluated and presented, that are consistent with the dose limits stated above. For this, representative content is developed for the casks, and used in the dose analyses. It is necessary to develop this representative content here, since the content that would be qualified based on the dose rate limited is not known yet. To cover the different conditions, this representative content is developed separately for the HI-TRAC and HI-STORM, and also separately for more gamma and more neutron dominated content. The development of the content follows the steps outlined below:

- Gamma or Neutron
  - To represent more gamma dominated content, the cooling time is set to the minimum (TBD years in all cases), and then the burnups are adjusted until the dose rate limits are approximately met.
  - To represent more neutron dominated content, the burnup is set to the maximum, and then the cooling times are adjusted until the dose rate limits are approximately met.
- "approximately met" is understood that the limits are in general slightly exceeded. This way, the derived other doses and dose rates would be expected to be upper bound values that would not be exceeded when the fuel is loaded to the qualified limit.
- Side and top of casks
  - The side dose rates are more determined by the assemblies on the periphery of the basket, whereas the top dose rates are generally more determined by the assemblies in the center of the basket. Hence the assemblies on the periphery of the basket may be selected with different burnup/cooling time values than those in the center of the basket, to match the respective limits.
- Surface average vs. maximum

  - Even within the cells on the periphery and the cells in the center of the basket, variations in burnups and cooling times may be selected to match the maximum dose rate.

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- Cask version
  - For normal conditions, nominal cask design are used for the evaluations presented here, specifically the HI-TRAC with TBD inches of lead, and the HI-STORM FW with a concrete density of TBD pcf in the wall.
  - These choices are not critical for the evaluations presented here, since it is to be expected that for the same surface dose rates, the other dose rates presented here would also be similar, regardless of the specific characteristics of the cask used in the evaluations.
- For the accident condition of the loss of water from the HI-TRAC outer water shield, a model with minimum lead thickness is evaluated.
  - This represents a bounding condition for the accident dose rate at 100 m distance. No further site-specific accident evaluations are therefore necessary.

The representative content identified based on the above is as follows:

- HI-TRAC
  - Gamma dominated
    - Outer assemblies TBD GWd/mtU, TBD years
    - Inner assemblies TBD GWd/mtU, TBD years
  - Neutron dominated
    - Outer assemblies TBD GWd/mtU, TBD years
    - Inner assemblies TBD GWd/mtU, TBD years
- HI-STORM
  - Gamma dominated
    - Outer assemblies TBD GWd/mtU, TBD years
    - Inner assemblies TBD GWd/mtU, TBD years
  - Neutron dominated
    - Outer assemblies TBD GWd/mtU, TBD years
    - Inner assemblies TBD GWd/mtU, TBD years

This content is then used to evaluate doses and dose rates at the following locations / under the following conditions

- Normal conditions
  - HI-TRAC and HI-STORM, surface and 1 m distance, at the same locations that were evaluated in Section 5.1. This is not to demonstrate compliance with any regulatory requirement, but to give an indication of the maximum dose rates in those locations.
  - Occupational dose rates, equivalent to those presented in Chapter 11.
  - HI-STORM, annual dose for various cask arrays at selected distances, to show compliance with 72.236(d)

- Accident condition
    - HI-TRAC with loss of water from the water jacket, at 100 m from the cask. This is to generally show compliance with 72.106.

For each dose location, results from the more gamma dominated and the more neutron dominated representative content are calculated, and the maximum values are presented in the tables at the end of this subsection.

C.5 Summary

The information in this subsection is to be used, in combination with the information presented in the TR [C.1] to qualify content (fuel assemblies) for the casks in the FSAR. The qualification is documented in one or more qualification reports, as also outlined in the TR [C.1].

For details on which information in this subsection can or cannot be modified under 72.48 see Section C.3.

C.6 References

[C.1] HI-2210161

Table C.1 Normal, HI-TRAC (equivalent to FSAR Table 5.1.1)

Table C.2 Normal, HI-STORM (equivalent to FSAR Table 5.1.5)

Table C.3 occupational (equivalent to FSAR Table 11.3.2, but summary only)

Table C.4 HI-STORM arrays (equivalent to Table 5.1.3)

Table C.5 HI-TRAC, Accident (equivalent to Table 5.1.4)

APPENDIX D EXAMPLE OF FUEL QUALIFICATION REPORT

The following example report outlines the structure and required content of the qualification report that should be followed. Any deviation from this structure and/or content requires justification that the report satisfies the original intent of documenting the qualification process.

Commented [SA46]: need to address #14h

Since this is an example for a separate report, it has its separate Table of Content, and page and section numbers. Information that would be site specific or depend on the specific implementation of this topical report in the corresponding FSAR and CoC are listed as "TBD".