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

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SUMMARY OF REVISION

Revision 0: Original Issue.

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.

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 in 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. An example of such a qualification report is included as appendix D. 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.

Finally, the FSAR (or TS) provides areas of applicability of each system, i.e., overall upper and/or lower bound values for certain important parameters. That would include global limits for burnup, enrichment, and cooling times, and limits for other fuel or design parameters.

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 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 larger range of contents for a larger range of systems, or specific reports 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.

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 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.

2.5 Acceptance Criteria

The 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. 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).
3) Dose locations will 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.

2.6 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.

2.6 Area of Applicability

This topical report is applicable to spent fuel storage systems, with general limitations listed in Table 2.2 For its use for a specific storage system, this report needs to be referenced in the respective CoC/TS of the system, and additional restrictions for it applicability may then also be specified in that context.

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 (or new Topical Report)	CoC Holder	Strict 10CFR72.48 Method of Evaluation Controls
Acceptable Content	Qualification Report	CoC Holder / Licensee	Submitted to NRC for information, but not approved

Table 2.2

AREA OF APPLICABILITY

Parameter	Applicability
Storage Systems	HI-STORM and HI-STAR
Fuel	Spent PWR and BWR fuel
Fuel Burnup	Up to 68.2 GWd/mtU for PWR fuel Up to 65.0 GWd/mtU for BWR fuel
Fuel Type	UO_2
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.
Enrichment	Up to 5.0 wt% ^{235}U
Cooling Time	Greater or equal to 1 year

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 codes to calculate neutron and gamma source terms should be the TRITON and ORIGAMI modules of the SCALE 6.2.1 system. As alternatives, it is acceptable to use these modules from newer version of the SCALE code. In that case it shall be demonstrated, for a small set of BECTs that span the variations of the burnups, enrichments and cooling times to be qualified, that the results 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.

Further, the standard TRITON libraries supplied with the code shall be used, unless no suitable library is available for the respective fuel type, in which case it is acceptable to specifically generate libraries. In such a case, the TRITON calculations shall be based on the relevant fuel characteristics, which are burnup and enrichment range, fuel rod and rod array information, and the active length of the fuel. This shall be documented together with the qualification reports.

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 should be performed for the design basis assemblies listed in Tables 3.1 or 3.2, which have been shown in [1] to reasonably bound all assembly types in the corresponding FSARs. However, it is also acceptable to perform the source term evaluations for other individual assembly types, for example for a site-specific qualification of acceptable contents. In this case:

1) The same information as that listed in Tables 3.1 and 3.2 for the design basis assemblies shall be specified for these individual assembly types, with justification.
2) The fuel qualification is limited to this assembly type, and this limitation is clearly stated in the qualification report.
3) The modeling in the radiation transport calculations must also consider this assembly type with respect to self-shielding. For that, it is important that for an assembly with a lower uranium weight, this lower uranium weight is also considered in the dose analyses so that the self-shielding is not overestimated.
4) A single full power cycle shall also be used, but the specific power may be adjusted to correspond to the assembly type to be qualified.

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 must be considered directly in the radiation transport calculations.

Gamma Source from Active Fuel Region

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. Table 3.5 presents an acceptable energy group structure, taken from Reference [1]. Other structures can be used, as long as they contain a similar or larger number of groups and comparable group limits.

Gamma Source from Activation of Structural Materials in the Fuel

The primary source of activity in the non-fuel regions of an assembly arises from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is impurities in the steel structural material above and below the fuel. Reference [3] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions shall be modeled with 0.8 gm/kg impurity level. The zircaloy in these regions, and in the active region of the fuel, can be neglected since it does not have a significant ^{59}Co impurity level.

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]. As stated above, a ^{59}Co impurity level of 0.8 gm/kg is used for both Inconel and stainless steel. Therefore, there is little distinction between stainless steel and Inconel in the source term generation and since the shielding characteristics are similar, stainless steel can be used in the radiation transport calculations instead of Inconel.

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, conservatively low initial fuel enrichments must be used in the analyses.

As for gamma sources, neutron source terms shall be generated by energy group in a suitable group structure. Table 3.6 presents a suitable group structure, taken from Reference [1]. Other structures can be used, if they contain a similar or larger number of groups and comparable group limits.

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.

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 the analysis in this chapter 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 0.8 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.

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 can be utilized in any source term calculation without any further justification. For specific qualifications, it is also acceptable to consider assembly and/or site-specific inserts, with appropriate justification of the specifications used. 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, so as a bounding approach, BPRAs can be assumed to be present in an assembly to represent either BPRAs or TPDs.

The qualification for BPRAs and/or TPDs shall be consistent with the loading of those components. For example, if these are only considered in the dose analyses in specific cells in the basket, they must be restricted to those cells in the loading plans to be implemented.

The mass of BPRAs or TPDs are often not considered in the radiation transport analyses as additional shielding, which is a conservative approach since it neglects material that would provide some additional shielding. However, it is acceptable to consider these masses. In that case, it needs to be limited to those assemblies in a basket where the source terms of the components are applied.

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.

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.

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 must 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 may not be considered in the radiation transport analyses as additional shielding, which is conservative. However, it is acceptable to consider these masses. In that case, it needs to be limited to those assemblies in a basket where the source terms of the components are applied.

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.

During in-core operations, the stainless steel and Inconel portions of the NSAs become activated, producing a significant amount of ^{60}Co. Comparisons of activation levels have been performed and it was concluded that activation from NSAs are bounded by activation from BPRAs. Hence from a gamma source perspective, these are treated as BPRAs.

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.
- 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].

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)

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)

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

4.0 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.

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:

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 4.0 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.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 4.0 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.

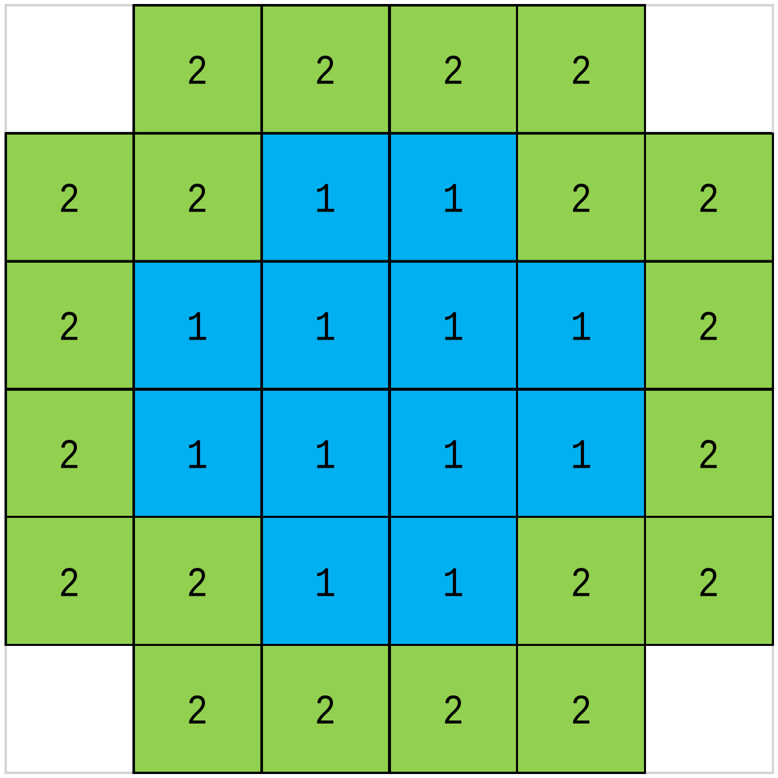

Figure A.1 32 Assembly Basket Layout with the Region Number identified in each Cell for Example 1

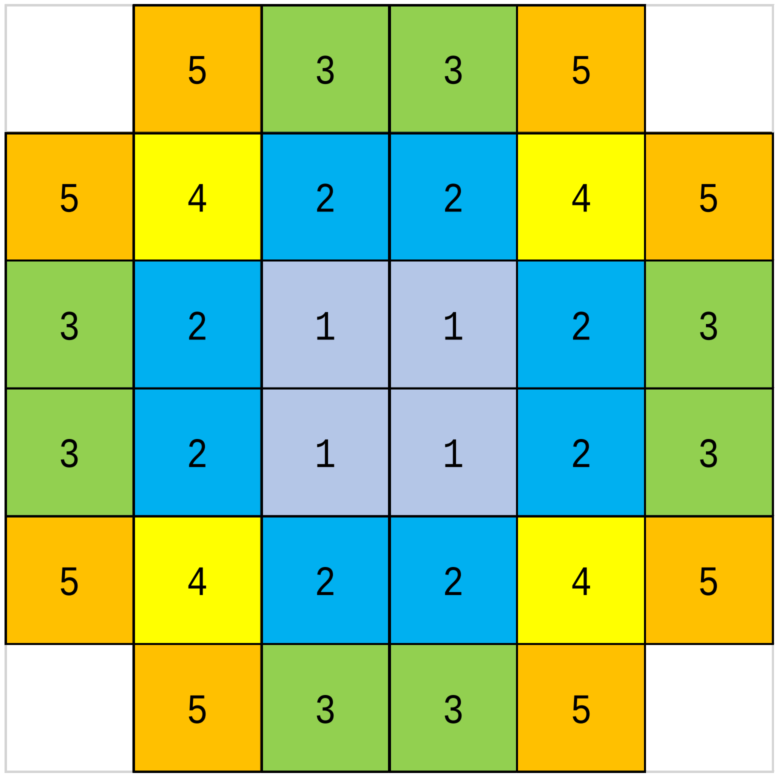

Figure A.2 32 Assembly Basket Layout with the Region Number identified in each Cell for Example 3

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.

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) 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 and possibly any non-fuel hardware must be consistent with (i.e., the same or lower than) the masses utilized in the source term calculations.
6) The calculations must consider the axial burnup distribution of the fuel assemblies.

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.

Area of Applicability

1) For fuel, the area of applicability must 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.

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 case, both conditions 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 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 must 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, two limiting values are defined for each area, an average and a maximum. The average will be more representative from an ALARA perspective for personnel working further away from a cask, the overall cask content and the site boundary, whereas maximum does rates are more relevant for personnel working in close proximity to the cask, and more representative for the most limiting content of the cask. Defining only one of those would be insufficient from both an ALARA and a content qualification perspective, hence both are important and specified for each area.

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
  - Average TBD rem/hr
  - Maximum TBD rem/hr
- Top
  - Average TBD rem/hr
  - 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
    - Average TBD mrem/hr
    - Maximum TBD mrem/hr
- Top
    - Average TBD mrem/hr
    - Maximum TBD mrem/hr

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.2 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 also validated against all other safety requirements, not just shielding.

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.

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.3 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
    - For 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.
    - For 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 surface average and maximum dose rates.

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

Results are presented in the tables at the end of this subsection.

C.4 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.2.

C.5 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

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 implementation of this topical report in the corresponding FSAR and CoC are listed as “TBD”.

Table of Contents

1. Purpose

The purpose of this report is to document the qualification of fuel for loading into the HI-STORM 100 [5]. Qualification is performed based on the process defined in the Shielding MOE Topical Report [6], together with the Section 5.TBD in [5] and the requirements outlined in Section TBD of the HI-STORM 100 CoC [1].

The qualification is performed for fuel defined in this report, and for loading into the MPC-32 and for the HI-TRAC 100 and the HI-STORM 100, with the specific fuel characteristics defined in this report. Only fuel that meets all explicit requirements outlined in this report can be loaded under this qualification in this basket and in these casks.

The purpose of this qualification is compliance with 10 CFR 72.236(a), and hence with 10 CFR 72.236 (d) as discussed in [5].

Note that compliance with 10 CFR 72.104 and 10 CFR 72.106 is not part of the scope of this document. This is demonstrated in other reports.

For the qualification, dose rates calculations with burnup, enrichment and cooling times bounding all fuel to be qualified are performed for these conditions, for the locations and conditions specified in Section 5.TBD in [5], and it is shown that the dose rates are below the applicable limits defined in Section TBD of the HI-STORM 100 CoC [1]

2. General Methodology

The HI-STORM 100 model is taken from reference [5], with the following site-specific parameter:

- TBD

The HI-TRAC 100 model is taken from reference [5], with the following site-specific parameter:

- TBD

Qualification is only for loading fuel into casks that meet these conditions.

The calculations are performed for the 17x17 design basis fuel specified in Table 3.1 of [6].

The radiation analysis performed in this report can be separated into two distinct parts. The first is the generation of the radiation source terms to represent the spent nuclear fuel at the appropriate burnup and cooling time. The second part is the radiation transport simulation to calculate the dose rates near and far from a cask.

The neutron and gamma source terms, and the decay heat values, are calculated with the TRITON / ORIGAMI module of the SCALE 6.2.1 code package [7]. This is an improved method compared to the SAS2H [2] / ORIGEN-S [3] from SCALE 5.1, using predefined libraries for a large number of standard fuel assemblies, based on updated data sets, using a 252-energy group structure. Use of this code is consistent with the requirement in [6].

The TRITON / ORIGAMI input and output files are presented in reference [10]. Calculated gamma, neutron and hardware source terms are also presented in reference [10].

The radiation transport simulation is performed with MCNP5 [4] from Los Alamos National Laboratory. This is a state of the art Monte Carlo code that offers coupled neutron-gamma transport using continuous energy cross sections in a full three-dimensional geometry. The HI-STORM 100 and HI-TRAC 100 are modeled in full three-dimensional geometries in MCNP5.

3. Acceptance Criteria

The HI-STORM 100 Certificate of Compliance (CoC) [1] describes dose rate requirements for fuel qualification in Section 5.TBD of Appendix A. Subsection 5.TBD states:

"Based on the analysis performed pursuant to Section 5.TBD, the licensee shall demonstrate that for the fuel to be loaded, dose rate for the HI-TRAC TRANSFER CASK and the HI-STORM OVERPACK shall be below the following value:

a. The top of the TRANSFER CASK TBD average, TBD maximum
b. The side of the TRANSFER CASK TBD average, TBD maximum
c. The top of the OVERPACK TBD average, TBD maximum
d. The side of the OVERPACK TBD average, TBD maximum"

4. Assumptions

The following assumptions are used:

1. The HI-TRAC 100 transfer cask is calculated with the water jacket full of water, a dry annulus between the MPC and HI-TRAC and an MPC empty of water.
2. The HI-TRAC transfer cask is assumed to be surrounded by air on all sides.
3. The enrichment of the fresh fuel assembly modeled in MCNP for this report is 3.4 wt.% ^{235}U. This is a conservative assumption since the actual spent fuel in a storage cask has fewer amounts of fissile isotopes as compared to using a ^{235}U enrichment of 3.4 wt.%. Using a higher ^{235}U concentration in the active fuel region in MCNP will result in higher secondary neutron production from subcritical multiplication and fast fission in the fuel and also higher gamma doses from n, gamma reactions within the shielding materials. Both of these quantities are calculated in MCNP and are not part of the source term calculations. Also, fission products in the burned fuel, which decrease the neutron multiplication factor, are conservatively neglected. This assumed enrichment of 3.4 wt.% ^{235}U in the MCNP models is consistent with MCNP models used in References [8].

Other assumptions are stated in the text as necessary. Since this report uses MCNP models from the HI-STORM 100 analyses, additional assumptions and discussion can be found in references [5] and [8] .

5. Input Data

The source terms are from reference [10]. The MCNP input data, material compositions, cask geometry are from references [5] and [8].

The loading pattern for this qualification analysis is shown in Table 1, with the respective regions shown in Figure 1.

6. Computer Codes

Computer codes to perform source term calculations are TRITON and ORIGAMI from the SCALE 6.2.1 package [7].

Dose rate calculations are performed with MCNP5 [4].

7. Analysis and Results

This analysis principally uses the same cask models as used in reference [5] for site boundary calculations.

This section of the report describes the calculations that are performed to determine the dose rates on the surface of the HI-TRAC 100 and HI-STORM 100. The basic development of the MCNP models is provided in references [5] and [8]. This information is appropriately referenced as needed.

The source terms and method of tallying are described in reference [6]. All results presented in this calculation package were calculated using the ANSI/ANS-6.1.1-1977 flux to dose conversion factors [9].

Appendix A describes the HI-TRAC 100 and HI-STORM 100 Tallies. This appendix summarizes the tally surfaces and cells, and segments for the dose calculations performed for the HI-TRAC 100 and the HI-STORM 100.

8. Computer Files

All computer runs listed here are made on Computer Systems at Holtec’s office. All files are stored on the Holtec computer server.

The following is a list of the MCNP runs that are used.

HI-STORM 100 Filenames	
MCNP Run	**Source**
TBD	TBD
TBD	TBD
TBD	TBD

HI-TRAC 100 Filenames (water jacket full of water, a dry annulus between the MPC and HI-TRAC, and an MPC empty of water)	
MCNP Run	**Source**
TBD	TBD
TBD	TBD
TBD	TBD

9. Summary

The qualification of content (fuel assembly burnup, enrichment and cooling times) is presented in this report.

Tables 2 and 3 present the surface dose rates for the HI-TRAC 100 and the HI-STORM 100 for the content to be qualified that is listed in Table 1. All dose rates are below the respective limits, hence the fuel as specified in Table 1 is qualified through this report.

10. References

[1] HI-STORM 100 Certificate of Compliance 1014, Amendment 8R1.

[2] I.C. Gauld, O.W. Hermann, “SAS2: A Coupled One-Dimensional Depletion and Shielding Analysis Module,” ORNL/TM-2005/39, Version 5.1, Vol. I, Book 3, Sect. S2, Oak Ridge National Laboratory, November 2006.

[3] I.C. Gauld, O.W. Hermann, R.M. Westfall, “ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms,” ORNL/TM-2005/39, Version 5.1, Vol. II, Book1, Sect. F7, Oak Ridge National Laboratory, November 2006.

[4] LA-UR-03-1987, MCNP — A General Monte Carlo N-Particle Transport Code, Version 5, April 24, 2003 (Revised 2/1/08).

[5] Final Safety Analysis Report for the HI-STORM 100 Cask System, HI-2002444 Revision 18, Holtec International (US NRC Docket No. 72-1014).

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

[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 (2018). Available from Radiation Safety Information Computational Center as CCC-834.

[8] HI-STORM 100 System Additional Shielding Calculations, HI-2012702 Revision 15, Holtec International.

[9] American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors, ANSI/ANS-6.1.1-1977.

[10] Source Terms and Loading Patterns Using Scale 6.2, HI-2167524 Revision 5, Holtec International.

Table 1 Qualified Burnup, Enrichment and Cooling Time Combinations

Region (See Figure 1)		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 2
Surface Dose Rates for the HI-TRAC 100 for Fuel Qualification

Location	Calculated Dose Rate (mrem/hr)	Dose Rate Limit (mrem/hr)	Dose Rate Limit Met
Side, Average	TBD	TBD	YES
Side, Maximum	TBD	TBD	YES
Top, Average	TBD	TBD	YES
Top, Maximum	TBD	TBD	YES

Table 3
Surface Dose Rates for the HI-STORM 100 for Fuel Qualification

Location	Calculated Dose Rate (mrem/hr)	Dose Rate Limit (mrem/hr)	Dose Rate Limit Met
Side, Average	TBD	TBD	YES
Side, Maximum	TBD	TBD	YES
Top, Average	TBD	TBD	YES
Top, Maximum	TBD	TBD	YES

	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 1: 32 Assembly Basket Layout with the Region Number identified in each Cell

Appendix A: HI-TRAC 100 and HI-STORM 100 Models and Tallies (total of 5 pages)

This appendix summarizes the tally surfaces and cells, and segments for the dose calculations performed for the HI-TRAC 100 and the HI-STORM 100.

HI-TRAC 100 Discussion:

The MCNP surfaces and cells that are used for tallying in the HI-TRAC 100 models are provided below.

Surface/Cell	Description
4100-4108	Cells – Segmented axially along outer side surface.
6000	Cell – Top of HI-TRAC 100 Lid, segmented in concentric rings.

The surfaces are segmented in axial, circumferential and radial direction so individual tallied areas are about the size of TBD ft^2.

Figures A1 through A2 provide figures which will help in understanding the modeling.

HI-STORM 100 Discussion:

The MCNP surfaces and cells that are used for tallying in the HI-STORM 100 models are provided below.

Surface/Cell	Description
4100-4108	Cell – Outer edge of overpack
6000	Surface – Top of HI-STORM lid

The surfaces are segmented in axial, circumferential and radial direction so individual tallied areas are about the size of TBD ft^2.

Figures A3 through A4 provide pictures of the MCNP model of the HI-STORM 100 overpack that will aid in understanding the modeling.

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TBD

Figure A1: MCNP model for HI-TRAC 100.

TBD

Figure A2: A cross sectional view of the HI-TRAC 100 model.

TBD

Figure A3: A cross sectional view of the side of the HI-STORM 100 overpack model.

TBD

Figure A4: A view of the model used for the HI-STORM 100 overpack.