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SUMMARY OF REVISION

Revision 0 (Draft 1/31/2021): Original Issue.

1.0 INTRODUCTION

For Dry Storage Systems, 10CFR72.236(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). 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 that is greater or equal to 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, and 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 calculated does 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 10CFR72.236(a), where the specific contents can be defined in separate qualification reports that are prepared and maintained outside of the FSAR and CoC. Instead of specifying BECTs, limiting dose rates are defined in the FSAR/CoC/TS, and the 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, which 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. Nevertheless, the qualification reports are submitted to NRC for information.
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 aspects that are needed to utilize this approach.

2.0 OVERVIEW OF THE APPROACH

The overall approach to define allowable contents as supported by this Topical Report has three components, the methodology defined in this report, information in the FSAR of the dry storage system, and the qualification reports. These are discussed in the following subsections.

2.1 Information in this Topical Report

This TR describes the method to define allowable contents for a cask system. Once approved, it is referenced in the TS and can be invoked instead of defining the contents in the TS.

1) Section 3.0 outlines the individual steps that need to be followed in applying this methodology.
2) Section 4.0 contains the description of the source term methodology to be used. This is an integral part of the method.

Note that a given FSAR (or TS) may already contain previously established BECT limits to satisfy 10CFR72.236(a) for some given conditions. When updating and 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 that the contents is to be qualified for. 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 recommendations and requirements for the information that needs to be defined in the FSAR. This is not an integral part of the method but needs to be consulted to ensure the appropriate information is specified in the FSAR and TS.

2.3 Qualification reports

The evaluations and analyses to qualify any given set of contents 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. Qualifications reports must be submitted to NRC, but the implementation of the limits established in a report does not require NRC approval of that report.

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.

3.0 FUEL QUALIFICATION METHODOLOGY

3.1 Inputs

Inputs are candidate loading patterns for given casks or 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.

2.2 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 the criteria are used to define and qualify the contents, it is only necessary to apply limits to one of these two casks. The corresponding FSAR will define the dose rate limits and specify if they are applicable to the storage or transfer cask. The FSAR may also specify limits for both casks.
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. However, the dose locations will be selected to be away from locations with large dose gradients, so that small changes in the selection of the dose rate tallies would not have a significant effect on the fuel qualification process.
   a) Dose locations will not include locations where peak dose rates are expected over small areas. Such dose rates will be discussed and presented in the FSAR, but such locations are not considered suitable to be used in the qualification of the contents.
4) Appropriate dose locations, and corresponding dose rate limits will be defined in the FSAR for each specific design or design variation of the storage systems. Examples of dose rate limits are provided in Appendix A for illustrative purposes.

To muster qualification, candidate patterns shall remain below the limits. The complete qualification process is illustrated in the examples in Appendix A.

3.3 Calculations

The calculational methods for source term generations are outlined in Section 3.0 of this TR. The method outlined there is to be used to determine the gamma and neutron source terms for any fuel assembly with a given burnup, enrichment, and cooling time. It is also used to determine the source terms of any Non-Fuel Hardware that is to be stored with the fuel in a specific cell location.

The calculational methods for the radiation transport calculations are outlined in the FSAR for the system to be qualified, considering the discussions in Appendix B of this TR. The dose rates are then calculated for the locations specifically defined for the respective cask and based on the source terms for the fuel to be qualified.

3.4 Additional Restrictions on Contents

This Topical Report establishes the principal Methodology to technically evaluate and qualify candidate loading patterns that satisfy given dose rate limits. Any additional restrictions or requirements that maybe necessary from a licensing perspective, for example heat load limits, will be specified in the FSAR or separate documents. Additional parameters are not addressed 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.

3.5 Implementation

The implementation of the approach outlined in this TR consist of the following steps

1) Approval of this TR

2) License Amendment for any FSAR/CoC that intends to use this TR.
   a) The description of the radiation transport analyses needs to meet the requirements discussed in Appendix B of this Topical Report.
   b) Dose rate limits need to be specified in the CoC/TS.
   c) A reference to this TR needs to be included in the CoC/TS.
   d) Such License Amendments will be submitted separately from this TR
3) Qualification reports need to be generated.

4.0 SOURCE TERM EVALUATIONS FOR QUALIFICATION OF FUEL

4.1 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.

The preferred code to calculate neutron and gamma source terms are the TRITON and ORIGAMI modules of the SCALE 6.2.1 system. It is also acceptable to use these modules from newer version of the SCALE code. In that case it must be demonstrated, for a representative set of BECTs, 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 generated libraries. In such a case, the TRITON calculations need to consider the relevant fuel characteristics, which are burnup and enrichment range, fuel rod and rod array information, and the active length of the fuel.

Source term calculations are generally performed for selected design basis assemblies that reasonably bound all assembly types in the corresponding FSAR. However, it is also acceptable to perform the source term evaluations for other selected assembly types, for example for a site-specific qualification of acceptable contents. In this case:

1) The fuel qualification is limited to this assembly type, and this limitation is clearly stated in the qualification report.
2) The modeling in the radiation transport calculations must also consider this assembly type. For example, it is important that for an assembly with a lower uranium weight than the design basis assembly used in the source term generation, this lower uranium weight is also considered in the dose analyses so that the self-shielding is not overestimated.

When performing the ORIGAMI calculations for design basis assemblies, a single full power cycle shall be used to achieve the desired burnup, together with the above-average specific powers listed in Tables 3.1

or 3.2, since this has been shown to result in conservative source terms relative to actual multicycle power operation.

When performing ORIGAMI calculations for other selected assembly types, 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.

4.2 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 needs to be used in the analyses. Table 4.5 presents an acceptable energy group structure, taken from Reference [1]. Other structures can be used, as long as they contain a similar number of groups and similar 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 can both be assumed to have the same 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 needs to 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 4.1 were taken from References [3], [4], and [5]. As stated above, a ^{59}Co

impurity level of 0.8 gm/kg was 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 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 4.3.

4.3 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 are to be used in the analyses.

As for gamma sources, neutron source terms need to be generated by energy group in a suitable group structure. Table 4.6 presents a suitable group structure, taken from Reference [1]. Other structures can be used, if they contain a similar number of groups and similar group limits.

4.4 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) can be qualified for storage as an integral part of a PWR fuel assembly.

4.4.1 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 are to be determined in the same way as the structural materials in the fuel assembly. In the calculations the ^{59}Co impurity level should conservatively be assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for Inconel. The calculations are then 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 should 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 is then scaled by the appropriate scaling factors listed in Table 4.3 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 4.4. For specific qualifications, it is also acceptable to consider assembly and/or site-specific inserts. Since inserts are managed, handled and can be qualified separate 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 must 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.

Traditionally, the mass of BPRAs or TPDs are 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.

4.4.2 CRAs and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral 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 in aluminum is used, 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.

4.4.3 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. If an evaluation is performed that shows that the neutron source term from an NSA is in fact 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 evaluated, 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 should be located near the center of that basket, consistent with the approach in Reference [1].

4.5 Choice of Design Basis Assembly

The Westinghouse 17x17 and GE 10x10 assemblies are selected as the preferred design basis assemblies since they are widely used throughout the industry. Alternatively, B&W 15x15 and GE 7x7 can be used, which contain higher fuel masses. Parameters of these assembly type are stated in Tables 3.1 and 3.2. Specific assemblies other than those may be used for site-specific qualifications. In any case, the choice of assembly parameters must be consistent between the source term and radiation transport calculations.

Table 4.1

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 4.2

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 4.3

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 4.4

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 4.5 Recommended 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 4.6 Recommended 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

5.0 CONCLUSION

This Topical Report provides 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 sections 2.0 and 3.0 of 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.

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.

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.

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.

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.3

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 2

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 2

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.

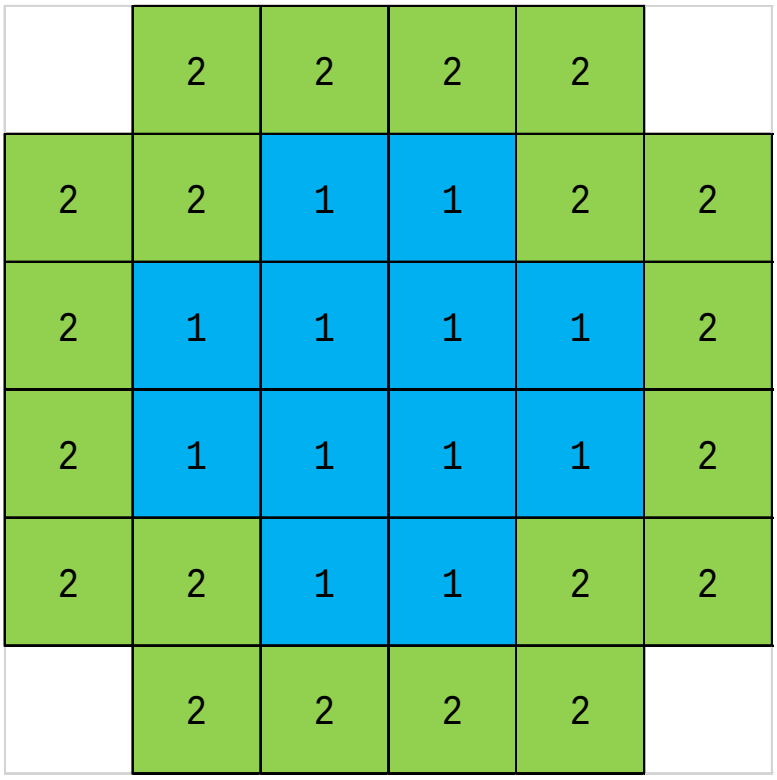

Figure A.1 32 Assembly Basket Layout with the Region Number identified in each Cell for Example 1

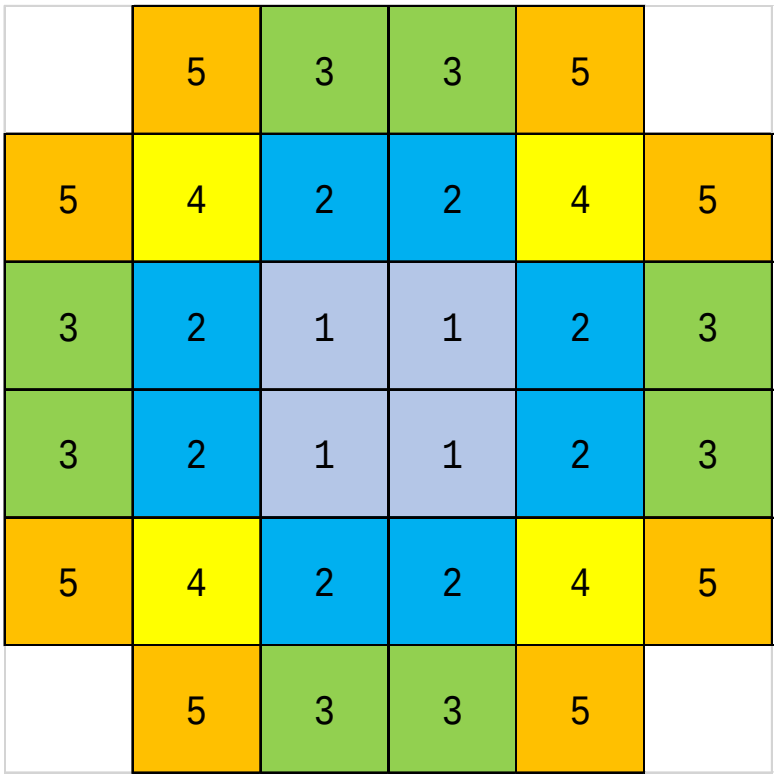

Figure A.1 32 Assembly Basket Layout with the Region Number identified in each Cell for Example 3

APPENDIX B GUIDANCE ON FSAR CONTENTS

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. Details and guidance on the information that needs to be provided in the FSAR are summarized below.

Calculational Models

1) The calculational models have to 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 the masses utilized in the source term calculations.
6) The calculations must appropriately 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) It is considered sufficient to specify dose rate limits for either transfer or storage system, not for both, since limits for a single system would be sufficient to allow the determination of allowable BECTs.
   b) 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, 4 dose rate locations on side of the cask, equally spaced around the perimeter, and one location on the top of the lid, may be sufficient. 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.

c) 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 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 need to 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 distribution 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 should be analyzed, and for each dose location the higher value should be reported or utilized.
4) For accident conditions, both source distributions should 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.