NEI 12-16, Revision 4

Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants

September 2019

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Nuclear Energy Institute

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ACKNOWLEDGEMENTS

This guidance was developed by the NEI Criticality Task Force. We also recognize the direct participation of the licensees and vendors who contributed to the development of the guidance. The dedicated and timely effort of the many participants, including management support of the effort, is greatly appreciated. Finally, we would like to thank the U.S. Nuclear Regulatory Commission for providing feedback during the series of meetings. This guidance has been updated to incorporate NRC feedback.

Nuclear Energy Institute, 1201 F Street N. W., Suite 1100, Washington D.C. 20004 (202.739.8000)

FOREWORD

This guidance describes acceptable approaches that may be used by industry to perform criticality analyses for the storage of new and spent fuel at light-water reactor power plants, in compliance with 10 CFR Part 50. The guidance provided herein is applicable to new fuel assemblies stored in a new fuel vault, and to new and spent fuel assemblies stored in a spent fuel pool.

Criticality requirements for the spent fuel pool of nuclear power plants are found in 10 CFR 50.68 or 10 CFR 70.24. Guidance for performing criticality analyses in compliance with these regulations was previously developed in a 1998 Nuclear Regulatory Commission internal memorandum by L. Kopp, further supplemented by the Standard Review Plan, NUREG-0800, Sections 9.1.1 and 9.1.2. More recent guidance was issued by the NRC in an Interim Staff Guidance (DSS-ISG-2010-01) in 2011. This industry document is developed as a comprehensive guide that presents an acceptable approach to comply with the regulations upon NRC endorsement. Individual vendors or licensees can deviate from the method supplied herein, with appropriate justification and approval by the NRC.

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ABBREVIATIONS AND ACRONYMS

AEG	Average Energy Group Causing Fission
APSR	Axial Power Shaping Rod
B&W	Babcock & Wilcox
BMU	Burnup Measurement Uncertainty
BPRA	Burnable Poison Rod Assembly
BWR	Boiling Water Reactor
CE	Combustion Engineering
CFR	Code of Federal Regulations
EALF	Energy of the Average Lethargy Causing Fission
ENDF	Evaluated Nuclear Data File
EPRI	Electric Power Research Institute
FTF	Fuel Transfer Form
GWD	Giga-Watt Days
IFBA	Integral Fuel Burnable Absorber
ISG	Interim Staff Guidance
LAR	License Amendment Request
MOX	Mixed-Oxide
MTU	Metric Ton Uranium
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Co-Operation and Development
ORNL	Oak Ridge National Laboratory
PWR	Pressurized Water Reactor
QA	Quality Assurance
RCCA	Rod Cluster Control Assembly
RSS	Root Sum Square
SCCG	Standard Cold-Core Geometry
SFP	Spent Fuel Pool
SNM	Special Nuclear Material
WABA	Wet Annular Burnable Absorber

1 INTRODUCTION

1.1 PURPOSE

This document provides acceptable approaches for performing criticality analyses for light-water nuclear reactor spent fuel pool storage racks and new fuel vaults. This guidance is applicable to both Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) facilities. These analyses are integral to the technical foundation for the design of nuclear fuel storage structures, systems and components, and the associated Technical Specifications in applications (i.e., License Amendment Requests (LARs)) submitted to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

This document is developed to provide comprehensive and durable guidance to improve consistency and clarity for performing criticality analyses that assure criticality safety and regulatory compliance. It is envisioned that this guidance will be endorsed by the NRC through a Regulatory Guide, and provide durable guidance for preparation of criticality analysis for LWR facilities.

1.2 BACKGROUND

10 CFR 50.68 [1] was promulgated in 1998 to provide an analysis based alternative to the criticality monitoring required by 10 CFR 70.24 [2]. Prior to the rulemaking, exemptions to the monitoring requirement in 10 CFR 70.24 [2] were granted on a case-by-case basis for licensees demonstrating subcriticality through analysis. Compliance with either regulation is consistent with 10 CFR 50, Appendix A, General Design Criteria 62, “Prevention of Criticality in Fuel Storage and Handling.” [3] 10 CFR Part 52 [4] was originally promulgated in 2007, and requires compliance with 10 CFR 50.68 [1].

The first guidance on acceptable methods for performing criticality analyses at LWR plants, was issued in 1978 in Generic Letter 78-11 [42] and further modified in 1979 with Generic Letter 79-04 [43]. More extensive guidance was developed in1998 in conjunction with the promulgation of 10CFR 50.68 [1] through an NRC internal memorandum from L. Kopp to T. Collins, often referred to as the “Kopp Memorandum” [24]. Although this was an internal NRC memorandum, it was quickly adopted by industry for use in performing criticality analyses, referenced in LARs, and referred to by NRC staff in the Safety Evaluation Reports for the associated license amendments due to the lack of formal guidance. The guidance in the Kopp Memorandum provided regulatory clarity and stability for many years. In 2010, the NRC issued an action plan to develop new interim staff review guidance followed by a durable Regulatory Guide that would replace the Kopp Memorandum and better reflect the staff positions on acceptable criticality analysis methods that evolved through interactions with licensees since 2005.

NRC Interim Staff Guidance (ISG) DSS-ISG-2010-01, “Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools,” [25] was issued in 2011 to provide additional guidance to staff for the review of spent fuel pool storage rack criticality analyses. The guidance in DSS-ISG-2010-01 [25] is useful to support NRC staff review of industry criticality analyses until the more permanent and durable guidance in NEI 12-16 is endorsed by the NRC.

1.3 APPLICABLE REGULATIONS

The following regulations are applicable to criticality analyses for nuclear fuel storage at LWR facilities:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 1, "Quality Standards and Records." [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 2, "Design Bases for Protection Against Natural Phenomena." [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 3, "Fire Protection." [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 4, "Environmental and Dynamic Effects Design Bases." [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 5, "Sharing of Structures, Systems and Components." [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 61, "Fuel Storage and Handling and Radioactivity Control." [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 62, "Prevention of Criticality in Fuel Storage and Handling." [3]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." [6]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.68, "Criticality Accident Requirements." [1]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical Specifications." [7]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(17), "Contents of applications; technical information."; 52.79(a)(43), "Contents of applications; technical information in final safety analysis report."; 52.137(a)(17), "Contents of applications; technical information."; and 52.157(f)(8), "Contents of applications; technical information in final safety analysis report." [4]

- Title 10 of the *Code of Federal Regulations* (10 CFR) 70.24, "Criticality Accident

Requirements." [2]

It is noted that in addition to the applicable regulations, the NRC developed the following staff review guidance associated with the criticality analyses for nuclear fuel storage at LWR facilities:

- NUREG-0800, Standard Review Plan, Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3. [12]
- NUREG-0800, Standard Review Plan, Section 9.1.2, "New and Spent Fuel Storage," Revision 4. [13]
- GL 78-11, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", [42]
- GL 79-04, "Modifications to NRC Guidance, 'Review and Acceptance of Spent Fuel Storage and Handling Applications'" [43]
- NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998. [24]
- DSS-ISG-2010-001, Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools, [25]

1.4 Double Contingency Principle

The double contingency principle [9] states, "process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." In other words, the nuclear criticality analysis is required to demonstrate that criticality cannot occur without at least two unlikely, independent and concurrent incidents or abnormal occurrences. This will ensure that no single occurrence can lead to an inadvertent criticality event. The double contingency principle means that a realistic condition may be assumed for the criticality analysis when calculating the effects of incidents or abnormal occurrences. When applying the double contingency principle, the chosen conditions need to be independent from one another (i.e. do not result from a common initiator) and are unlikely (i.e. low probability) to occur. For example, for PWRs, the loss of soluble boron below the minimum Technical Specification requirement is considered as one accident condition and a second concurrent accident need not be assumed (e.g., such as a fuel assembly misloading or misplacement). Therefore, compliance with the Technical Specifications minimum required soluble boron concentration may be credited when evaluating other accident conditions.

1.5 Use of Precedents

The use of precedents (i.e., adopting methods or conclusions previously approved in another application, but not necessarily documented in a generic regulatory document) is a well-

established principle by the NRC in the process of reviewing applications. The use of precedents provides regulatory stability and efficiency. In order for a licensee to use precedents in an application, the licensee should demonstrate the applicability to its site specific analysis reflecting an evaluation of the similarities and differences from the original use. Precedents should be used within the confines of the limitations of the context established when previously approved. Precedents may be used in whole or in part with technical justification. Any similarities or differences should be technically supported and demonstrated as appropriate. Consideration should also be given to any relevant NRC guidance that has been issued in the form of Interim Staff Guidance, Information Notices, etc., from the time of the approval of the original occurrence to the time of the application that uses it as precedence.

1.6 ASSUMPTIONS AND ENGINEERING JUDGMENT

Use of engineering judgment in criticality analyses can result in resource efficiencies. The use of engineering judgment as a basis for an element of the methodology is acceptable as long as the applicant can demonstrate that the rationale behind such determination is sound and can justify that the engineering judgment would not lead to non-conservative results with respect to the regulatory requirements.

The licensee assumptions used in the criticality analysis should be explicitly identified and clearly stated. Assumptions can be listed under two categories: explicit and implicit. Explicit assumptions are those the licensee (in this case more specifically the criticality analyst) consciously chooses in preparing the analysis. Implicit assumptions are those the licensee uses that are inherent [i.e., involved in the constitution or essential character of something] to the method. To ensure completeness, and provide clarity to the regulator for the application review, it is important that the licensee clearly identify their assumptions. The licensee, to the extent practicable, should provide a basis supporting assumptions defined in the application.

Use of engineering judgment and assumptions may incorporate risk insights as part of a "graded" licensing approach and is acceptable as long as the assessments consider relevant safety margins and defense-in-depth attributes. For example, a criticality analysis that demonstrates a maximum k_{eff} with a relatively large margin to the regulatory k_{eff} limit, may be permitted to make more assumptions about results or uncertainties than a criticality analysis that demonstrates a maximum k_{eff} with a relatively small margin to the regulatory k_{eff} limit.

2 ACCEPTANCE CRITERIA

Fresh (New) Fuel Storage

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core. However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire-fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation). Normal conditions (i.e., dry) need not be addressed in criticality safety analyses since there is no moderator. However, criticality safety analyses must address the following two independent events with associated limits:

a) With the new fuel storage racks assumed to be loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. The evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used (10CFR 50.68(b)(2)).

b) With the new fuel storage racks loaded with fresh fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid corresponding to optimum moderation, the k_{eff} must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. The evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used (10CFR 50.68(b)(3)).

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if: (1) it can be clearly demonstrated that design features and/or administrative controls prevent such flooding; (2) criticality monitors in accordance with requirements of 10 CFR 70.24 are provided, or (3) an exemption to the criticality monitoring requirements of 10 CFR 70.24 has been granted.

<u>Spent (Used) Fuel Storage</u>

Criticality safety analyses for pool storage of new and used fuel may utilize one of two available approaches.

1) For pools where no credit for soluble boron is taken (typically BWR pools), the criticality safety analyses must meet the following limit:

    a. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must not exceed 0.95, at a 95 percent probability, 95 percent confidence level (10CFR 50.68(b)(4)).

2) For pools where credit for soluble boron is taken (typically PWR pools), the criticality safety analyses must meet two independent limits:

    a. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must remain below 1.0 (subcritical), at a 95-percent probability, 95 percent confidence level (10CFR 50.68(b)(4)).

    b. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with borated water, the k_{eff} must not exceed 0.95, at a 95-percent probability, 95-percent confidence level (10CFR 50.68(b)(4)).

3 COMPUTER CODES

3.1 TYPES AND USES OF COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent the important neutronic and geometrical aspects of the storage racks. In spent fuel pool criticality safety analyses, there are two general types of computer codes that are used. These are criticality codes and depletion codes. The criticality codes are used to determine the eigenvalue (k_{eff}) of the analyzed system. The isotopic concentrations of the materials in the system are determined from manufacturing data and depletion analysis.

The codes to perform depletion and criticality calculations rely upon the use of cross-section libraries. Cross-section libraries used in the criticality analysis should be widely accepted and peer reviewed. Cross-section libraries recommended for use include the multi-group and continuous energy ENDF/B-V and later series.

The licensee needs to state which codes were utilized along with the type/version of cross section libraries. The use of the term computer code in this document means the combination of the computer code and cross-section library. The code version and cross section set used in the analysis needs to be the same as those used in the validation of the codes for simplicity and reduction of calculational burden.

3.1.1 Criticality Codes

Typically, a criticality code uses a Monte Carlo method to estimate the system k_{eff}. The Monte Carlo method relies on repeated random sampling to compute the answer. Cross sections are used as probabilities of interaction and the Monte Carlo code then calculates and tracks individual neutron lifecycles. Although many criticality codes utilize Monte Carlo methods, there are other criticality codes that provide acceptable results utilizing deterministic transport methods.

A description of the criteria for determining acceptability of convergence should be included as part of the application. The convergence and uncertainty of the Monte-Carlo criticality code result is sensitive to various input parameters, including, for example:

- the number of neutrons per generation,
- number of generations skipped prior to averaging,
- the total number of generations, and
- the initial source distribution, especially in loosely coupled systems.

The choice of input parameters is intended to optimize calculational accuracy and computer processing time. The initial source distribution should be specified appropriately for the type of code being utilized and geometric configuration(s) being analyzed. The resultant Monte Carlo uncertainty is dependent upon the parameters selected above. There is no stipulation or requirement specified on the magnitude of the resultant Monte Carlo uncertainty, as it is incorporated into the overall calculational result with other uncertainties.

3.1.2 Depletion Codes

The depletion codes are used to calculate the nuclide density changes that occur in the fuel during operation in the reactor core. In addition, decay changes in nuclide concentrations due to non-power cooling times are also captured in depletion calculations. In general, depletion codes utilize deterministic transport methods. However, Monte Carlo methods may also be used for depletion calculations, provided that spatial convergence of the neutron flux is achieved.
The depletion code needs to be used in accordance with the topical report, user manual guidance and/or other documentation associated with the use of the code. Specific attention needs to be paid to any limitations and/or conditions of the depletion code. For example, the burnup step size in the depletion calculations needs to be sufficiently small to ensure proper calculation of the isotopic inventory from burnup step to burnup step.

3.1.3 Nuclides Credited

The number of nuclides that are credited in the depletion and criticality analysis is dependent upon a number of factors: 1) available cross-sections in the depletion and criticality code, 2) methodology choice in the analysis (i.e., full burnup credit, actinide only), and 3) ability of the nuclides to remain within the fuel matrix and fuel rod. Credit for all actinides and fission products is based upon appropriate modeling in the depletion and criticality code and prior NRC approval of this approach on previous submittals over the last two decades.

One important consideration is that certain nuclides, such as fission gases and short-lived nuclides will no longer be present in the fuel matrix to the extent predicted by the depletion code. The short-lived nuclides are addressed by using the isotopic inventory after several days to allow the short-lived nuclides to decay to a negligible amount in the calculations.

Conservative gas release fractions that have been reviewed and approved by the NRC are available from the current Regulatory Guide 1.183 [44]. However, it is expected that Reg. Guide 1.183 will be updated to reflect higher linear powers than were used in developing the current limits. PNNL-18212, Revision 1 [45] provides both limiting release rates (Table 2.9) and a method for determining the release rates when the linear power is known (Appendix C).

3.2 COMPUTER CODE VALIDATION

The licensee should describe all computer codes that are used in the criticality safety analysis, including the validation of the codes. Validation of the codes includes benchmarking by the applicant (i.e., the analyst or organization performing the analysis) by comparison with experiments and accounting for the parameters not covered by the existing experiments. This qualifies both the ability of the applicant (analyst/organization) and the computer environment. The critical benchmark experiments used for validation should include configurations having neutronic and geometric characteristics comparable to those of the proposed storage facility.

The computer code validation consists of validating both the computer code used in the depletion calculations and the computer code used for calculating the reactivity of the system (i.e., the criticality code). Appendix A contains a discussion of acceptable methods of performing validation of the criticality (Section A.1) and depletion codes (Section A.2).

4 REACTIVITY EFFECTS OF DEPLETION

This section described appropriate methods for performing the depletion analysis for PWR and BWR fuel.

4.1 DEPLETION MODELS

Historically, depletion models consisted of a model to produce one-group cross sections followed by a solution of the isotopic production and loss equations. Although this approach produced good results, modern nodal methods used in core reload design use a two-dimensional lattice model which determines the one group fluxes used in the isotopic production and loss analysis. Separate lattice models are developed for each unique axial plane, such as low enrichment blankets, control rods insertion, and burnable absorbers.

Depletion analysis is performed using nominal fuel geometric dimensions, with the grid modeled as water.

4.2 REACTIVITY EFFECTS OF DEPLETION FOR PWRS

Spectral hardening results in an increased production rate of plutonium from increased fast neutron capture in ^{238}U. Enhanced plutonium production and the concurrent diminished fission of ^{235}U due to increased plutonium fission has the effect of increasing the reactivity of the fuel at discharge and beyond.

Significant parameters that could impact reactivity of used fuel in depletion analyses for PWRs are:

a) Power, Moderator Temperature and Fuel Temperature during Depletion
b) Soluble boron during depletion
c) Presence of burnable absorbers
d) Rodded operation

Additional guidance in selecting operating parameters for depletion analysis is provided in NUREG/CR-6665 [17]. While this list generally identifies operating parameters and components that are known to have an impact on the reactivity of the fuel, the applicant also needs to address any site-specific items (e.g., tritium production rods, axial power shaping rods) that are not explicitly identified here.

4.2.1 Depletion Analysis

Power, Moderator Temperature and Fuel Temperature during Depletion

The power density, fuel temperature and moderator temperature (and associated moderator density) are grouped together because of the unique inter-relationship between these three values during in-reactor fuel depletion. The power density and moderator flow rate of a fuel assembly during depletion will directly impact the moderator and fuel temperature with a higher power (and/or lower moderator flow) resulting in higher moderator and fuel temperatures. Higher moderator and fuel temperatures during depletion result in increased reactivity of used fuel in the

storage rack. While a higher power will lead to a higher ^{149}Sm content after the decay of ^{149}Pm, which lowers reactivity, this effect is much smaller than the impact of the moderator and fuel temperature. Therefore, depletion at high power, moderator temperature, and fuel temperature is typically conservative. Previous studies [17] have also identified a small reactivity impact due to power history, with a low power coast down providing a conservative end of life reactivity. If load follow (variation of reactor power to adjust to demand) is exercised, this should be evaluated against the high constant power assumption.

The power density of an individual fuel assembly tends to slightly increase with burnup to a maximum value (associated with the burnup near where the integral or burnable absorbers become fully depleted) at which point it drops off with additional burnup. The analyst may use either a single power density value chosen to bound the power density over the life of the fuel assembly in the reactor or use a bounding power density as a function of burnup. Further, assembly power density may be a function of fuel management strategy (e.g., cycle fuel management techniques, enrichment, presence of absorbers).

A conservative (and computationally simpler) approach to the choice of depletion moderator and fuel temperatures would be to use a maximum value along the entire axial length of the fuel assembly. A more realistic approach could use the moderator and fuel temperature as a function of axial position. Licensed fuel management tools use models that predict fuel temperature as a function of the linear heat rate and burnup. It is acceptable to use these fuel temperatures based on a maximum power density to determine a conservative fuel temperature (applied either uniformly or as a function of axial height and burnup). If the approach taken is to use an axially distributed moderator temperature, justification for its appropriateness is needed.

Soluble Boron during Depletion

The soluble boron concentration during depletion can have a significant impact on the reactivity of the fuel in the storage rack. The higher the concentration during depletion, the higher the reactivity of the fuel at a given burnup. It has been shown that treatment of the soluble boron as a burnup averaged value results in the same effect on the fuel reactivity as modeling the actual boron concentration changes as a function of time [30], for complete cycles.

A conservatively high burnup-weighted cycle-averaged soluble boron concentration (to bound future cycle-average soluble boron contents that increase with time due to increased fuel enrichment and fuel utilization) should therefore be confirmed and used in the depletion calculations. The licensee will confirm the actual cycle-average soluble boron for the purposes of confirming the individual cycles meet the inputs of the approved analysis.

A licensee would evaluate a mid-cycle offload in accordance with the licensee's corrective action program and current NRC guidance for identifying and resolving potential non-conservatisms or unanalyzed conditions in a design basis analysis. If an issue is identified, the licensee would make an initial operability determination, and subsequently evaluate in accordance with 50.59 to determine whether NRC approval is required. As a default, any fuel assembly could be conservatively treated as a fresh fuel assembly with no burnable absorbers.

Burnable Absorbers

PWR reactors use a variety of burnable absorbers during operation for the purposes of reactivity control and power distribution control. These absorbers can be mixed into the fuel pellet (e.g., Gadolinium, Erbium), added as a coating on the fuel pellet (e.g., ZrB_2 IFBA) or be included as inserts in the guide tubes (e.g., WABA, BPRA, Pyrex).

In all cases, the depletion analysis should appropriately consider and account for the effect associated with the presence of these absorbers on the reactivity of the fuel. The bounding neutron absorber loading of the burnable absorbers for the maximum burnup should be modeled.

Burnable absorbers harden the energy spectrum during operation due to the presence of the neutron absorber (i.e., absorption of thermal neutrons) and the displacement of water from the guide tubes. The reactivity effect on the fuel assembly is a function of the duration of the removable absorber in the fuel assembly (determined through the amount of burnup the fuel assembly experiences while the burnable absorber is present). Therefore, the maximum burnup that a fuel assembly receives while containing a burnable absorber must be determined and used in the analysis.

Studies have shown that Gadolinium and Erbium burnable absorbers can be conservatively neglected [18]. While spectral hardening does occur in fuel bearing Gadolinium or Erbium, the positive reactivity impact of this effect is never larger than the negative reactivity impact due to displacement of fissile material (UO_2) and residual Gadolinium/Erbium. Therefore, it is conservative to model fuel bearing Gadolinium or Erbium as though the integral absorber was not present. Note that when Gadolinium and/or Erbium are excluded from the analysis, the models cannot credit the reduced UO_2 and fuel density caused by Gadolinium and Erbium; a fuel density based on fuel without integral absorber must be used. Recent analysis has confirmed that neglecting Gadolinium and Erbium burnable absorbers is a conservative approach [31].

It is also important to note that multiple absorbers, such as WABAs and IFBAs, can be present in a fuel assembly undergoing depletion in any given cycle. In the event of multiple absorbers, the depletion analysis should take into account all burnable absorbers present in the fuel assembly, over the entire burnup. For instance, if one burnable absorber type is assumed to be present in only the first cycle, then it should be confirmed that assemblies exposed to burnable absorbers or other inserts (e.g., detector thimbles, hafnium suppressor rods, primary and secondary sources) in subsequent cycles are appropriately bounded by the assumptions of the depletion analysis. Normally, primary and secondary sources will be covered by the conservatism in the burnable absorber assumptions, but confirmation is necessary.

For part-length absorbers, it is conservative to model the absorber as full length, as the hardening of the spectrum is applied to axial sections that do not contain absorbers. An inherent assumption behind this conservative approach is that any residual absorber is not credited. However, it is acceptable for an applicant to perform separate depletion calculations with and without absorbers, with the appropriate isotopic concentrations applied to each axial section in the criticality analysis. For burnable absorbers that are inserted into the guide tube and modelled as part length, separate depletion calculations for the regions above/below the burnable absorber

should be modelled with water displaced in the guide tubes and the appropriate isotopic inventory applied to these nodes in the criticality models.

In all cases the burnable absorbers are modeled with nominal dimensions in the depletion analysis.

Rodded Operation

The criticality safety analysis should include the impact of exposure to fully or partially inserted control rods (and/or part length rods) since rodded operation typically increases the fuel assembly reactivity at a given burnup [19]. Control rod insertion has a similar effect as burnable absorber by affecting the energy spectrum in the core. While most PWRs operate with all rods out (i.e., no partial insertion in the core), use of this assumption should be justified. Separate loading criteria may be developed if different assumptions are used for addressing rodded operation.

Cooling Time

The standard practice is to perform the depletion analysis at a very short cooling time (hours or days) with no ^{135}Xe to determine the spent fuel isotopic inventory after discharge. This is commonly referred to a zero cooling time and is intended to represent freshly discharged fuel. However, as the short lived fission products decay and ^{241}Pu decays to ^{241}Am, the fuel assembly reactivity continues to decline to a minimum at approximately 100 years, as demonstrated in NUREG/CR-6781 [16]. This additional reduction of reactivity with cooling time can be credited to allow for greater flexibility in managing the spent fuel inventory. Many of the modern depletion codes can perform the change in isotopic inventory with additional time automatically. The applicant needs to include a description of the code and approach used to perform the cooling time calculations. It is recommended to limit the cooling time credit to 30 years.

Other Depletion Parameters

The modeling of down time or part power operation during depletion has been shown to have only a small effect on the assembly reactivity [32]. As discussed above, the use of conservative moderator and fuel temperature based on the highest assembly power for the duration of depletion produces a conservative isotopic concentration.

Flux suppression inserts have been used at a number of plants. Flux suppression inserts are composed of a strong neutron absorber, such as Hafnium, to reduce the flux on the core vessel. Being composed of a neutron absorber, they harden the spectrum and displace water from the guide tubes, similar to the effect associated with control rods and burnable absorbers. Typically, these inserts are placed in fuel assemblies in the periphery of the core, where little additional burnup accumulates while these inserts are present. These inserts require analysis to show that the burnable absorber assumptions cover the reactivity effects associated with flux suppression inserts.

It is recommended that the applicant include a summary of the core depletion parameters (operating parameters, presence of burnable absorber, etc.) used in the analysis in sufficient

detail to support performance of confirmatory calculations. The summary should include sketches or figures and a table with dimensions and material properties. This information can also serve the applicant as a guide to the inputs used in the analysis for evaluating future changes in operation.

4.2.2 Fuel Assembly Physical Changes with Depletion

During reactor operation, the fuel rods and fuel assembly undergo physical changes. For the fuel rod, these changes are driven by the behavior of the ceramic uranium dioxide fuel pellets as they generate energy. These may have an impact on the reactivity of the fuel in the SFP environment. The specific physical changes of concern are changes to fuel density, clad outer diameter (OD), and clad thickness. It should be noted data for fuel pellet diameter is also captured because fuel pellet diameter changes are directly correlated to fuel density changes. Additionally, the fuel assembly geometry changes as a result of exposure to irradiation and temperature that result in growth of the grid spacers (and corresponding increase in pitch between fuel rods)

Applicants need to address the potential reactivity impact of the following changes that occur during depletion:

a. Fuel rod changes (clad creep, fuel densification/swelling)
b. Material dependent grid growth

While this list generally identifies known changes to the fuel rod or fuel assembly that have a potential impact on the reactivity of the fuel, the applicant also needs to address any potentially site-specific changes (e.g., crud induced power shift (CIPS)), that are not explicitly identified here.

An example of the reactivity impact of changes of fuel geometry with irradiation was analyzed in a proprietary Westinghouse study which is summarized in Appendix B.

4.2.3 PWR Depletion Bias and Uncertainty

Historically, engineering judgment was used to estimate the uncertainty associated with fuel depletion calculations as a percentage of the change in reactivity associated with depletion [24]. An independent evaluation [27, 33] has been conducted by EPRI to provide a basis for this approach. The evaluation determined that both a small bias and an uncertainty is appropriate to be applied, as further described in Appendix A. When calculating the depletion bias and uncertainty, the reactivity decrement is defined as the change in reactivity between the zero burnup, fresh fuel condition and the burnup of interest without burnable absorbers.

In lieu of a formal lattice depletion validation, the licensee may apply an uncertainty equal to 5% of the reactivity decrement, if the licensee uses the lattice depletion code in a manner that is consistent with nuclear design calculations previously performed for commercial power reactor licensing. This ensures that the depletion code will produce reliable and predictable results for the intended application.

Because these methods are an integral benchmark of the entire system modeled by the depletion codes, it covers all uncertainties associated with depletion, such as uncertainty in computation of the isotopic inventory by the depletion code, uncertainty in cross-sections (both actinides and fission products), etc.

4.3 Peak Reactivity Analysis for BWRs

It is standard practice that BWR spent fuel pool criticality analyses are performed at the burnup that produces the lattice peak reactivity. BWR fuel lattices that contain an integral burnable absorber typically result in a lattice peak reactivity at a specific burnup value, usually under 20 GWD/MTU, due to the positive reactivity from the depletion of the integral burnable absorber competing with the negative reactivity from the depletion of the fissile material.

The general methodology for BWR spent fuel pool criticality analyses is to perform in-core depletion calculation for the various assembly designs in use, then to restart the calculations with the assemblies in the standard cold core geometry (SCCG) and then in the storage rack geometry. The SCCG is defined as an infinite array of fuel assemblies on a 6-inch lattice spacing at 20ºC, without any control rods or voids. The burnup at the limiting k_{inf} in the SCCG is determined and then the k_{inf} in the storage rack geometry is calculated at this burnup. A reactivity allowance for applicable biases and uncertainties is added to the calculated k_{inf} in the rack geometry and the resulting k_{eff} is compared to the regulatory limit of 0.95.

BWR depletion analyses are performed with 2D calculations with each unique lattice modeled independently. Given that axial blankets are significantly lower enrichment than the other lattices in the bundle, axial blankets may be considered bounded by the other lattices. The peak reactivity method inherently bounds all axial effects by modeling the peak axial reactivity across all exposures for the entire length of the bundle. Additionally because the most reactive fuel lattice at its most reactive point in life is modeled, fresh fuel stored in the SFP is covered by the peak reactivity criticality analysis that meets the in-core k_{inf} limit.

4.3.1 Depletion Parameters

A licensee should account for the dependence of the peak reactivity burnup and the magnitude of the peak reactivity for all storage rack calculations that are used to determine the maximum in-rack k_{eff} in the analysis. The reactivity effects of the reactor operating parameters can be applied either as separate biases or included in the design basis models. When limiting reactor operating parameters are included in the design basis models, the analysis should determine and use the combination of reactor operating parameters that result in the bounding peak reactivity in the SFP rack geometry.

The following parameters can have a significant impact on reactivity in the storage rack and therefore should be considered:

- **Reactor operating parameters**:
    - Void fraction –The full range of void fractions should be considered in conjunction with the other reactor parameters.

  - Moderator temperature – The moderator temperature is typically a fixed value in a BWR and should be considered in conjunction with the values appropriate to the reactor operation at power.
  - Fuel temperature – Higher fuel temperatures typically results in an increase in peak reactivity in the storage racks.
  - Power density – The power density typically has a lower impact on peak reactivity than the other reactor parameters and the value used can be chosen based on its relationship to the fuel temperature.
  - Control Rod Usage - The SCCG is calculated uncontrolled (i.e. no control rods insertion). However, the reactivity impact of control rod usage should be accounted for separately in the criticality analysis.

When considering what types of lattices to evaluate in the criticality analysis, the licensee should account for the different aspects of varying bundle designs as described below:

- **Lattice specific parameters:**
  - Enrichment – Typically highest planar average ^{235}U enrichment of all the bundle types being evaluated is bounding.
  - Part Length Rods – Each unique axial plane in the bundle designs being evaluated including number and location of partial length rods should be evaluated.
  - Integral Burnable Absorber Fuel Rods - Number, location and nominal concentration of integral burnable absorber fuel rods should be evaluated appropriately for the given application.
  - Nuclides modeled – Appropriate nuclides used in PWR depletion analyses or those nuclides used in BWR core design and core monitoring analyses are acceptable as described in Section 3.1.3.

All BWR criticality calculations should ensure a conservative reactivity is analyzed in the storage configuration with consideration given to possible cooling and discharge times. Nominal values for lattice parameters like fuel pellet density, fuel rod diameter, etc. should be used and the tolerances on these parameters should be evaluated in the tolerance analysis described in Section 5.1.3.

4.3.2 BWR Depletion Uncertainty

The BWR lattice physics/depletion codes used for SFP criticality analyses are the same depletion codes used and validated for BWR core design and core monitoring applications. In these applications, the integral burnable neutron absorber burnout is very important, so there is high confidence that the integral burnable neutron absorber depletion is accurate within 5%. It is additionally noted that 5% of the reactivity decrement to burnup of interest is reasonable for BWRs given that PWR depletion uncertainty validation with measured power flux data has demonstrated the 5% of the reactivity decrement is conservative and they are very similar, both being thermal, light water reactors with low enriched UO_2 fuel.

The reactivity decrement to the burnup of interest is, specifically, the cold, beginning-of life (BOL) k_{eff} of the spent fuel rack analyzed bundle with no integral burnable neutron absorber present

compared to the maximum k_{eff} of the cold, analyzed bundle at the exposure statepoint (with Gadolinium) used in the analysis as shown in Figure 4-5. Both k_{eff} values are calculated for comparison in the rack system. Five percent of the difference in k_{eff} values between these two cases is included as an uncertainty to the spent fuel pool criticality analysis to cover the depletion isotopic benchmarking gap. Figure 4-5 illustrates determination of the reactivity decrement for BWR criticality analysis where the burnup of interest is the peak reactivity.

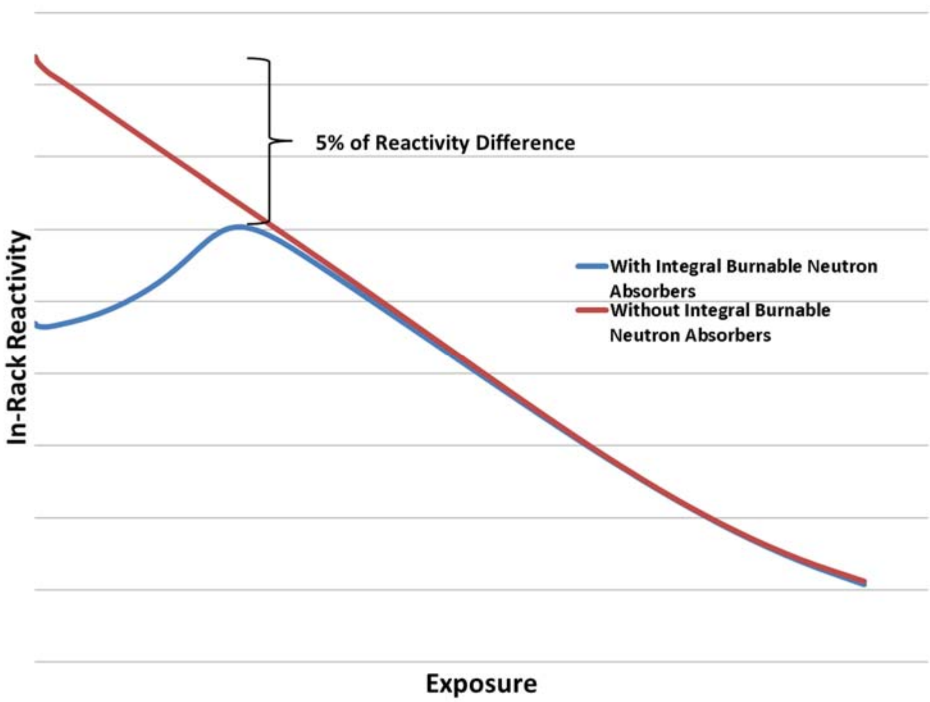

Figure 4-1: BWR Peak Reactivity Depletion Uncertainty

The licensee may use 5% of the reactivity decrement, if the licensee uses the lattice depletion code in a manner that is consistent with nuclear design calculations previously performed for commercial power reactor licensing. This ensures that the depletion code will produce reliable and predictable results for the intended application.

5 FUEL ASSEMBLY AND STORAGE RACK MODELING

5.1 Fuel Assembly Modeling

5.1.1 Fuel Assembly Modeling Considerations

The fuel assembly modeling in the criticality code includes an explicit representation of the fuel rods. For 3D analysis, the fuel rods are modeled with a length equal to the active fuel length. Hardware above and below the active fuel length is ignored and modelled as a water reflector (with no soluble boron) of the same temperature and density as the moderator in the active fuel region. For simplification, grids can be neglected in the fuel assembly model (for both borated

and non-borated conditions). However, an additional 50 ppm[1] of soluble boron needs to be reserved when soluble boron is credited to offset the reactivity impact of the fuel assembly grids [31]. For BWR fuel, the fuel assembly can be stored in the rack either with or without the channel and thus the impact of the channel presence (or absence) should be investigated. The fuel assembly is modeled using nominal dimensions with manufacturing tolerances addressed separately as described in Section 5.1.3.

5.1.2 Design Basis Fuel Assembly

Most, if not all, spent fuel pools contain multiple fuel assembly designs. In the case of PWR pools, this is typically limited to two or three different designs that are geometrically very similar with only minor changes that have a relatively small effect on reactivity (grid spacer, mixing vane modifications). BWR pools, however, typically have many more fuel assembly designs with significant geometric differences (e.g., different array sizes, differences in the number, location and shape of water rods, presence of partial length fuel rods). Regardless of the differences, it is convenient to establish a single fuel assembly design as the limiting design used in all depletion and criticality calculations for simplicity and consistency.

The design basis fuel assembly is determined at the temperature and water density that results in the maximum reactivity in the storage rack (See Section 5.2.2.1)

In the determination of the design basis assembly, it is acceptable to use a hybrid set of parameters from multiple assemblies that result in a bounding, more limiting design basis assembly. A prime example of this approach is the use of the maximum nominal fuel density that bounds all fuel designs in the spent fuel pool. This approach also provides additional conservatism in the analysis.

When significant differences occur between designs, it is acceptable to have more than one design basis fuel assembly.

Modified, reconstituted, damaged or consolidated fuel are not considered as part of the determination of the design basis assembly, but if they are present, need to be considered in the analysis separately to determine whether they are bounded by the design basis assembly or additional restrictions are necessary.

PWR Considerations

The applicant needs to evaluate the design-basis assembly for each unique rack design, storage configuration (e.g., two of four storage, absorber inserts) and fuel assembly type, using nominal dimensions to establish which fuel assembly type is most limiting. It is also important to address the change in reactivity with depletion, as the bounding fuel type can change with burnup (because of differences in the fuel to moderator ratio between different fuel designs, a fuel assembly that is bounding at fresh fuel conditions, may not be limiting at other burnups) and enrichments. The design basis fuel assembly is that assembly that provides the most limiting reactivity at a given burnup and enrichment. In the case where a single fuel assembly is not bounding over all burnup and enrichment combinations, the difference between the design basis

[1] This 50 ppm is also sufficient to offset the change in reactivity effect of tolerances under borated conditions (i.e., the reactivity effect of tolerances in the unborated condition can also be used in the borated condition).

assembly and the other more bounding assembly type(s) is applied as a bias to the calculation of maximum k_{eff}.

BWR Considerations

One method of determining an appropriate BWR design basis assembly for a given rack is to model the rack fully loaded with identical fuel assemblies that are characterized by a peak reactivity which is at or just above the desired in-core k_{inf} limit. If multiple rack storage configurations are used in the analysis, a design basis bundle assembly should be determined for each unique rack storage configuration. Ranges of the following parameters for a given fuel product line need to be considered:

- Lattice Type (i.e. Dominant, Vanished, etc)
- Lattice Exposure
- Lattice Average Enrichment
- Number of Gadolinia Rods
- Gadolinia Concentration
- Void History

The resulting in-core k_{inf} and in-rack k_{eff} values from these sensitivity studies are used to define the rack efficiency (in-rack k_{eff} /in-core k_{inf}) associated with a specific lattice and rack design combination. The design basis lattice is the lattice that results in the highest rack efficiency (i.e. worst reactivity suppression capability) at its peak reactivity statepoint and meets the SCCG k_{inf} limit criterion. Additional details can be found in ANSI/ANS-8.27-2015 [11] and the PHYSOR 2010 Proceedings [41].

5.1.3 Fuel Assembly Manufacturing Tolerances

As described in Section 5.1, criticality analyses rely on a nominal representation of the fuel assembly design (i.e., nominal dimensions, materials, and isotopic concentrations). However, each individual parameter is manufactured within specified tolerances to ensure quality control, fabricability, etc.

The following fuel assembly tolerances should be considered for inclusion as uncertainties in the criticality analysis, unless they can be shown to be insignificant:

a) Enrichment
b) Channel (BWR only)
c) Pellet Density
d) Rod Pitch
e) Fuel Pellet Outside Diameter
f) Cladding Outside Diameter

While this list generally identifies manufacturing tolerances that are known to have an impact on the reactivity of the fuel, the applicant also needs to address any site-specific tolerances (e.g., IFBA loading, dishing & chamfering) that are not explicitly identified here.

The reactivity impact of individual uncertainty items are evaluated separately. For independent uncertainties, the total k_{eff} uncertainty is the root sum square (RSS) of the individual k_{eff} uncertainty values. Alternatively, the analysis could calculate k_{eff} with all tolerance values selected to maximize k_{eff}. It is also acceptable to use a combination of these two approaches. For example, a maximum pellet density may be used and the other parameters are statistically combined.

To ensure that the maximum reactivity is being calculated per the requirement of 10CFR50.68 [1], effects of tolerances should be considered for each parameter that may contribute to a significant positive reactivity effect. Significance is determined based upon the overall effect on the total uncertainty, and on the margin to the regulatory limit. Because the total uncertainty term is typically dominated by a few large uncertainties, an individual uncertainty that is less than 10% of the total uncertainty may be considered insignificant. For example, suppose the total uncertainty (defined to be the square root of the sum of the squares of independent uncertainties or RSS) is 0.01 Δk. Using RSS, the effect of an additional independent uncertainty equal to 10% of the total uncertainty (0.001 Δk) can be calculated to increase the total uncertainty from 0.01Δk to only 0.01005 Δk. Unless the margin to the regulatory limit is very small, the 0.001 Δk uncertainty is not significant compared to the total uncertainty. An applicant can assess those uncertainties that do not need to be specifically analyzed for a given application based on previous calculations of similar systems (fuel assemblies and/or rack designs) along with engineering judgement.

The significance of some uncertainty values may vary with storage conditions (e.g. soluble boron and rack design). Fuel assembly tolerances should be evaluated in the appropriate rack model. The criticality analysis should demonstrate that the uncertainty values used are appropriate to the storage conditions by using either condition-specific values, bounding values, or application of additional k_{eff} margin to the regulatory limit.

Tolerances on the fuel clad thickness and guide and instrument tube thickness have been shown in a generic study to be insignificant and do not require analysis [31]. The clad thickness tolerance is insignificant since zirconium has a small absorption cross section. Since the inside of the clad is a gap filled with helium, the substitution of gas for zirconium has very little reactivity effect. While changing the guide and instrumentation tube thickness does affect the amount of water, the number of guide/instrument tubes is less than 10% of the number of rods in an assembly and this low volume makes the reactivity of the tolerance negligible.

It is recommended that the applicant include a summary of the fuel assembly parameters used in the analysis in sufficient detail to support performance of confirmatory calculations. The summary should include sketches or figures and a table with dimensions and material properties. This includes a figure of each unique guide tube/water rod pattern for the fuel assemblies in the spent fuel pool. This information can also serve the applicant as a guide to the inputs used in the analysis for evaluating future changes.

5.1.4 Axial Burnup Distribution

When modeling the fuel assembly in the criticality analysis, the reactivity is affected by the distribution of burnup along the axial length of the fuel assembly. The burnup distribution is affected by the operating conditions (temperatures, flux, presence of inserts, etc.). The axial burnup distribution starts out flat, quickly becomes cosine shaped and then gradually flattens in the middle of the assembly. Additionally, the neutron flux and power shifts to the ends of the assembly at the end of the fuel assembly life in the reactor. The lower burnup near the ends of the assembly combined with the lower moderator density at the top of the assembly, causes the region near the top of the fuel assembly to control the reactivity of the entire assembly. Therefore, the nuclear criticality analysis should consider an appropriate representation and nodalization of the burnup profile that encompasses a bounding shape of the licensee's inventory. Three options are provided for licensees to choose from for modeling of the axial burnup distribution, depending on the amount of information available to support the analysis and the level of verification for future fuel assemblies to meet the axial burnup distribution used in the analysis. In all three options, the results with an explicit axial burnup distribution are compared to the axially uniform profile, which assumes the same burnup along the entire axial length. This includes all storage configurations, including those with different loading requirements in different storage cells (e.g., checkerboard of fresh and spent fuel, mixing of high and low burnup fuel).

Option 1: Use of Generic Axial Burnup Distributions

NUREG/CR-6801 [20] evaluated 3169 axial burnup profiles to determine the most reactive representatives in each burnup range. Included in the population are B&W 15x15, CE 14x14, CE 16x16, Westinghouse 15x15 and Westinghouse 17x17 profiles. The profiles in the database include fuel designs that contain burnable absorbers that have been and continue to be used, including borosilicate glass, zirconium diboride (IFBA), WABAs, Gadolinium and Erbium. Additionally, the profiles include assemblies exposed to control rod insertion, including axial power shaping rods (APSRs). Given the broad range and applicability of the database, along with the selection of the axial burnup profile in each burnup range that produces the limiting reactivity, it is appropriate and conservative to use the NUREG/CR-6801 profiles for PWR reactors.

In NUREG/CR-6801, plant specific burnup profiles are used to determine the bounding axial burnup profile in individual burnup bins, and includes determination of the limiting axial burnup profile. If the plant that sets the limiting profile decides to use Option 1 for their criticality analysis, that plant needs to provide a site-specific verification that the limiting profile in NUREG/CR-6801 is still bounding for their plant (i.e., that there is not a subsequent profile that produces a more limiting result).

The database does not include axial burnup profiles associated with fuel assemblies containing lower enriched axial blankets at the top/bottom of the fuel assembly. However, it is acceptable to consider axially blanketed fuel assemblies bounded by fuel assemblies with no axial blankets.

Because of the broad range of applicability and the conservative nature of using the most reactive axial burnup profile for each identified burnup range, there is reasonable assurance that

axial burnup profiles from future discharged fuel assemblies will also be bounded by the database of profiles contained in NUREG/CR-6801. If significant changes are made to the core operation (e.g., load following, significant low-power operation, gray rods, flux suppression assemblies), it should be verified that the new axial burnup distributions still behave in a similar manner as the axial burnup distribution before the core design change.

The NUREG/CR-6801 limiting shapes were selected assuming the rack is uniform axially. If the rack has reduced-length absorber panels that leave a significant portion of the active fuel outside of the absorber panels, new limiting axial burnup distributions must be determined.

Option 2: Use of Plant Specific Bounding Profile(s)

Core management tools and advanced nodal codes have the ability to calculate the axial burnup distribution for each fuel assembly as a function of burnup throughout the cycle of operation. These axial burnup distributions are used to ensure the core operates within the limits specified for the reactor. These axial burnup profiles can also be used in the spent fuel pool criticality calculations. One conservative approach is to take the plant-specific population of axial burnup distributions and determine a bounding axial burnup profile specific to the fuel assemblies being stored in the spent fuel pool. A simple approach to create this bounding axial burnup profile is to take the minimum relative burnup of each node (there are typically between 10 and 25 nodes along the entire axial length) from all assemblies on-site at the specific licensee plant. To ensure that the composite axial burnup profile is conservative, no renormalization is performed. This typically provides a weighted relative burnup between 0.95 and 0.98. Use of Option 2 includes a need for the licensee to evaluate changes in core operations/fuel designs that might have a significant impact on the identified bounding profile (e.g., load following, changes in numbers/combinations of fuel assembly inserts).

Option 3: Use of Most Reactive Plant Specific Profile (s)

The third option also uses plant specific axial burnup profiles through the use of the most limiting profile(s) from the current population of fuel assemblies at the site. This approach involves determining which profile(s) are limiting, such as identifying those profiles with the lowest relative burnup in the nodes closest to the ends of the assembly. This approach ensures that all past discharged fuel is bounded and provides a level of reasonable assurance that future profiles will also be bounded, provided the reactor is operated in a similar manner (e.g., no increase in rodded operation, or new burnable absorber materials are introduced). Use of Option 3 includes verification that the axial burnup distribution of future fuel assemblies continue to be bounded by the limiting axial burnup profile(s) used in the analysis using the same method as was used to determine the most limiting profile. This verification would be controlled by the licensee through administrative procedures as part of the reload verification process.

Nodalization

The number and size of the nodes in the axial burnup distribution are an important consideration in ensuring the effect of the low burnup ends of the assembly are properly modeled. Previous studies have investigated the sensitivity of k_{eff} to the nodalization structure of the axial burnup distribution. NUREG/CR-6801, Appendix A [20] concludes:

> *"Results of variations in the size of axial zones in fuel assembly models indicated that for the most part, use of 18 uniform-height axial zones is sufficient to capture burnup distribution effects"*

Additionally, ORNL/TM-1999/99 [36] also found burnup distributions with even fewer nodes to be sufficient under the following circumstances:

> *"Calculations with as few as seven axial zones (three 1/18th-length zones at either end and one large central zone) were shown to converge to the same solution as an 18-uniform-zone model."*

These two references are consistent in recognizing the importance of the size of the nodes at the ends of the assembly (approximately 8 inches or less) and the non-importance of the nodal structure at the center of an assembly modeled with a distributed axial burnup profile. Therefore, the analyst should confirm that the nodes of the axial burnup distribution are appropriately sized, especially at the ends of the assembly. The recommended approach is to utilize equally sized nodes along the length of the active fuel no larger than eight inches. It may be necessary to evaluate this conclusion if changes in axial fuel enrichment and/or burnable absorber zoning occur in the future.

5.1.5 Reactor Record Burnup Uncertainty

The reactor record burnup uncertainty, also referred to as burnup uncertainty, (BU) is an uncertainty representing the maximum potential reactivity impact of deviations between an assembly's "true" burnup and the burnup based on reactor records. There are a number of ways to calculate BU, with each method assuming some value which represents the percent deviation between true and reactor record burnup. This value is typically assumed to be 5% and the effect is statistically combined with other uncertainties. Alternatively, utilities can reduce burnup of assemblies by 5% instead of incorporating the uncertainty. Reducing the burnup of each assembly is effectively the same as treating the BU as a bias instead of an uncertainty.

Both EPRI and ORNL have performed studies to evaluate the accuracy of reactor records [21, 35]. The EPRI and ORNL reports agree that burnup estimations based on the flux measurements followed by time integration are within 5% of the true assembly burnup, and as such using 5% as the BU is conservative. It should be noted that both studies indicate that when using properly calibrated core follow software which is updated with in-core measurements the uncertainty is less than 2%, however this would need to be justified on an application-specific basis for NRC approval.

Therefore, the burnup uncertainty should be accounted for by either 1) including a stand-alone uncertainty for inclusion in the statistical combination of all uncertainties, or 2) directly reducing the burnup of assemblies before storing them in the SFP. The criticality analysis should clearly identify whether the burnup uncertainty is included in the analysis or is applied to the reactor record burnup during verification that a fuel assembly can be placed in a designated storage location.

5.1.6 Assembly Inserts and Integral Absorber Credit

In addition to the modeling of the fuel assembly as described above, in some cases the burnable absorber inserts and/or integral absorbers contained in the fuel assembly are also modeled and/or credited in the criticality analysis. This is separate from the effects of these devices during depletion as described in Section 4.2.1.

Control rods are considered "used" when they meet their mechanical or nuclear design limits. This occurs before there is any significant reduction in their neutron absorbing properties for most of the control rod. These used control rods can be credited in the criticality analysis to hold down reactivity in assemblies and allow lower burnup requirements. Although neutron absorbing properties are not significantly diminished for used control rods, a conservative reduction should be considered based on the in-reactor usage of control rods.

Non-irradiated removable burnable absorbers (i.e., WABA's, BPRA's, borated SS rods) can be credited to provide additional reduction in the required burnup for storage. The primary effect is associated with crediting the neutron absorption capabilities of the insert, with a secondary effect associated with moderator displacement from the guide tube. A conservative approach is to model the insert with nominal geometrical dimensions in conjunction with a minimum absorber loading. However, uncertainties associated with the axial length and axial position of the absorber relative to the fuel need to be considered.

Irradiated removable burnable absorbers (i.e., WABA's, BPRA's) can also be credited to provide additional reduction in the required burnup for storage. Since the strong neutron absorber is no longer present the primary effect is associated with moderator displacement from the guide tube and can provide some small benefit. Any residual absorber that may remain after irradiation is not credited.

Fresh fuel often has fuel rods containing burnable absorbers inside the clad as a pellet coating (i.e., IFBA) or mixed in with the fuel (i.e., Gadolinium or Erbium). A conservative approach is to model the integral absorber with nominal geometrical dimensions in conjunction with a minimum absorber loading.

5.2 STORAGE RACK MODELING

5.2.1 New Fuel Vault

While the New Fuel Vault is a dry environment for unirradiated fuel assemblies, both full (100% density) moderator condition as well as optimum low density moderator condition (i.e., mist or foam) need to be considered to ensure the maximum reactivity condition is represented, per 10CFR50.68 [1] requirements.

Usually, the storage racks in the new fuel vault are designed with no neutron absorbers, but maintain a large lattice spacing sufficient to ensure a low reactivity under the accident condition of flooding. If used for storage of new fuel, specific calculations are necessary to assure the maximum k_{eff} is no greater than the regulatory limits. In the evaluation of the new fuel vaults, characteristics of the fuel assemblies, rack, vault construction, and any materials or equipment stored in the new fuel vault should be explicitly identified and evaluated, as applicable. If the

new fuel vault is modeled as a single system, there are no interfaces that need to be evaluated explicitly.

Given the open nature of the rack design for the new fuel vault, with limited rack structure, the model for the new fuel storage rack typically consists of just the fuel rods in the fuel assembly at the appropriate nominal pitch of the storage rack. The active fuel length of the fuel assembly is modelled at the maximum allowable enrichment, with moderator above and below. An important consideration in the optimum moderation condition, is the modelling of surrounding concrete (walls and floor), structures and equipment stored (if applicable) in the new fuel vault. Additionally, the concrete composition can have a considerable impact on the reactivity. The applicant should justify the use of the concrete composition and modeled vault geometry.

The maximum reactivity under optimum moderation conditions can typically occur between 6-15% of the fully flooded water density. A sufficiently small density variation (i.e., every 1%) is needed in this range to ensure that the maximum reactivity condition is identified. Credible temperature variations within the NFV need to be identified and analyzed.

The following vault tolerances should be, at a minimum, considered when evaluating the uncertainties:

a) Cell/Storage Location Pitch
b) Storage Cell Wall Thickness (if present)

Tolerance calculations should be performed for both moderator conditions (i.e., full and optimum).

It is recommended that the applicant include a summary of the new fuel storage vault parameters used in the analysis in sufficient detail to support performance of confirmatory calculation. The summary should include sketches or figures and a table with dimensions and material properties. This information can also serve the applicant as a guide to the inputs used in the analysis for evaluating future changes.

5.2.2 Spent Fuel Pool Racks

The spent fuel pool rack criticality model consists of a representation of the dimensions and materials of construction, including any installed neutron absorber as well as flux traps (if present). The rack structure should be modeled using nominal dimensions with an axial length equal to the active fuel region. If the neutron absorber does not extend the entire length of the active fuel region, it should be appropriately modelled (including uncertainty), depending on the location of the active fuel region in relation to the neutron absorber. The rack structure above and below the active fuel region are neglected and replaced with unborated water (even when borated water is used in the active fuel region). It is acceptable for minor parts of the rack construction (i.e., welds) to be neglected and replaced by water. Credit can be taken for radial leakage near the walls of the spent fuel pool for the purposes of allowing lower burnup fuel requirements on the periphery of the spent fuel pool.

To ensure the model captures any reactivity increases due to uncertainties associated with manufacturing tolerances, each parameter that may contribute to a significant positive reactivity

effect should be evaluated. The following spent fuel pool rack tolerances should be, at a minimum, considered when evaluating the uncertainties due to tolerances:

a) Flux Trap Size
b) Cell Inner Dimension/Storage Location Pitch
c) Storage Cell Wall Thickness
d) Rack and Insert Neutron Absorber Dimensions (length, width, thickness, axial location)
e) Neutron Absorber Sheathing Thickness

While this list generally identifies manufacturing tolerances that are known to have an impact on the reactivity, the applicant also needs to address any site-specific tolerances (e.g., rack structure cross members) that are not explicitly identified here.

It is recommended that the applicant include a summary of the storage rack parameters used in the analysis in sufficient detail to support performance of confirmatory calculations. The summary should include sketches or figures and a table with dimensions and material properties. This information can also serve the applicant as a guide to the inputs used in the analysis for evaluating future changes.

5.2.2.1 Spent Fuel Pool Temperature

The spent fuel pool temperature affects the reactivity of the storage racks through changes in the cross-sections (i.e., Doppler broadening and changes in the moderator density). The criticality analysis should include calculations at the maximum water density (4^{o} C) and the maximum temperature allowed for normal operation. The temperature producing the maximum reactivity should be used when comparing against the acceptance criteria. Typically, the most limiting condition will be found at either the highest and lowest temperature allowed. However calculations are recommended at intermediate temperatures to confirm a monotonically increasing/decreasing reactivity with temperature for each rack design and storage configuration (i.e., determination of the temperature and density of maximum reactivity).

5.2.2.2 Dimensions

Rack manufacturer drawings will provide detailed dimensions for the fuel storage racks, including dimensions for any neutron absorber panels, if present and how the neutron absorber is held in place.

For neutron absorbers that are installed after the original rack construction (i.e., rack inserts), the dimensions are also provided by the manufacturer through drawings or design specifications. The modeling of these absorbers should be consistent with these dimensions and with how they are installed in the SFP.

Manufacturing dimensional tolerances of the neutron absorbers should be included in the uncertainty analysis. Tolerances for absorber length (if shorter than active fuel length), width and thickness should be considered in the analysis. Minimum values for the length and width may be used in lieu of tolerance analyses.

5.2.2.3 Rack Neutron Absorbers

In order to increase the capacity of SFPs, many utilities performed re-racks with high density spent fuel racks. These racks incorporated neutron absorbers (typically containing boron) into the design to allow for higher density fuel storage. Additional absorbing capability may be added to the racks through the use of neutron absorbing rack inserts. The criticality analysis should include a detailed model of these neutron absorbers in order to ensure that they are effective in their intended function to prevent criticality in the SFP. Criticality analyses involving neutron absorber materials include modeling of the boron content (^{10}B areal density) and dimensions. Of these modeling parameters, ^{10}B areal density has by far the largest effect on k_{eff} (as compared with neutron absorber dimensions and non-neutron absorbing material compositions).

There are many different neutron absorbers in use in SFPs. For a detailed description of different neutron absorber materials, see the Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications [29].

5.2.2.3.1 Boron Content

The boron content of the neutron absorber (^{10}B areal density) is an important parameter in the SFP criticality analysis. A conservative approach to modeling the boron content is to assume the minimum boron concentration (typically described in terms of areal density in g/cm^2 ^{10}B) for every neutron absorber panel. This is conservative because all panels actually placed in service have higher boron concentrations, since the manufacturer must take into account manufacturing tolerances. For example, the manufacturer will target a nominal boron concentration that they can assure an acceptable minimum concentration accounting for manufacturing tolerances. In addition, the manufacturer will fabricate to an as-built minimum that is higher than the certified minimum to further account for manufacturing tolerances.

One approach is to use the minimum as-built areal density that is documented in the manufacturing records. The minimum as-built areal density is the lowest boron concentration measured from all of the panels. Thus all panels actually placed in service have boron concentrations at or above this minimum concentration, and these are documented in Quality Assurance (QA) records. In some cases, these records have been collected by the manufacturer and provided with delivery on a batch basis.

The recommended approach is to use the minimum certified areal density. This is based on the material purchase specification, and the manufacturing process must confirm that the boron content of all panels are above the minimum certified areal density in order to be acceptable for use. The minimum certified areal density is typically less than, and never greater than, the as-built minimum areal density, since QA records will document that all panels have boron concentrations at or above the minimum certified areal density. These QA records are verified prior to storing fuel in the spent fuel pool or new fuel vault racks.

5.2.2.3.2 Neutron Absorber Aging Effects

Certain neutron absorbers may undergo aging effects (i.e. changes in material dimensions or composition over the service life of the neutron absorber). The mechanisms for undergoing changes and the potential impact on their ability to perform their criticality control function are

typically specific to the absorber material and rack design. The criticality analysis needs to clearly identify the absorber assumptions and inputs. If material changes are anticipated over their intended service life, these changes should be appropriately bounded by the criticality analysis. In extreme cases, if degradation is anticipated to result in loss of ^{10}B areal density or absorber effectiveness, then appropriate margin to account for the degradation needs to be included in the criticality analysis sufficient to ensure the analysis is bounding for the intended service life of the pool.

Neutron absorber performance and aging characteristics are monitored through a monitoring program. If any unanticipated aging or change is identified through the monitoring program, then it should be evaluated to determine if there is any impact on the criticality analysis and whether other licensee programs should be utilized (e.g., 10 CFR 50.59 [8] process, operability evaluation).

5.2.2.4 Eccentric Positioning

Storage racks are designed to allow the fuel assembly to be easily moved into the storage cells with minimal interference between the fuel assembly and the storage cell walls. Based on common fuel handling techniques, equipment and procedures the fuel assembly is randomly located within the storage cells. Therefore, the common approach is to model the fuel assembly in the center of the storage cell (i.e., an equal distance from the fuel assembly face to the storage cell wall on all four sides). However, the possibility exists for fuel assemblies to be located in the corner of the storage cell, called eccentric positioning.

Studies [31] have been performed to determine the reactivity impact associated with eccentric positioning of many assemblies, with the following conclusions:

- When neutron absorber panels with an areal density above 0.01 g $^{10}B/cm^2$ are present on all four sides of the fuel assembly, a centrally located positioning of the fuel assembly in the storage cell is the most reactive configuration.
- When the neutron absorber is not present (or not credited), an eccentrically located positioning of the fuel assembly in the storage cell can be the most reactive configuration:
  - As the size of the model increases (and therefore more assemblies are eccentrically located) the reactivity increases. However, the likelihood of an increasing number of fuel assemblies being eccentrically located in the most reactive configuration also decreases.

To ensure that the reactivity effect of eccentric positioning is captured, it is recommended to determine the reactivity effect associated with a 4x4 model (16 storage locations) with eccentrically located fuel assemblies, including reflecting or periodic boundary conditions. This reactivity effect would be applied as a bias to the design basis, centrally located results. Alternatively, the applicant can incorporate eccentric positioning into the design basis calculation models, so that the reactivity impact is already captured in the calculation of k_{eff}.

In all cases, the effect of eccentric positioning would be determined for the design basis assembly at the moderator temperature and density of maximum reactivity.

6 CONFIGURATION MODELING

A storage configuration is any unique combination of requirements for fuel, inserts (either fixed neutron absorbers or reactivity hold-down devices) and/or empty cells for a rack design. The applicant needs to include a description of each unique storage configuration proposed as part of the application.

6.1 NORMAL CONDITIONS

The criticality analysis should consider normal conditions and operations that occur in the spent fuel pool. It is not sufficient to consider only the static condition where all fuel assemblies are in the approved storage locations. It is just as important to consider normal activities and operations in the spent fuel pool, including transient operations. Examples of these normal activities are movement of fuel in and around the spent fuel pool, fuel located in an inspection station or fuel elevator, fuel on pedestals in the storage racks and fuel reconstitution/repair. Normally the limiting condition is the static condition. Fuel inspections and reconstitution operations are generally separated from the rest of the pool by empty cells. Although the criticality analysis should consider normal conditions, generally calculations are only required for the static condition. Each different normal condition at a plant should be evaluated and if it is potentially more limiting than the static condition, then it should either be considered as a potential starting point for accidents or restricted to make it less limiting than static storage. It is noted that different plants will have different normal conditions.

6.2 INTERFACES

In the event the spent fuel pool contains more than a single storage configuration or storage rack design, the criticality analysis should consider the interface between adjacent storage configurations. An interface occurs every time two or more different storage configurations can be adjacent to one another. In some cases, interfaces may result in a higher k_{eff} than the k_{eff} of the individual configurations. If necessary, interface restrictions may need to be applied to provide conditions for certain storage configurations to be placed next to one another.

When an interface calculation is performed, essentially two semi-infinite arrays of each storage configuration are placed in the same model, possibly with a small gap between them in the case of rack-to-rack interfaces (i.e., no leakage is credited). If the model is sufficiently large enough (4 or more rows of storage cells of each configuration), the resulting k_{eff} of the interface can determine if the interface results in a more limiting condition than the individual storage configurations.

Acceptability of the storage configuration interfaces can be based on any of the following approaches:

1) Separate storage configurations are neutronically decoupled by a separation distance of at least 12 inches,
2) For multiple storage configurations within a single storage rack design, each individual storage cell simultaneously meets the storage requirements for all of the storage configurations of which it is a part.

3) Use of the maximum biases and uncertainties from the individual storage configurations.
4) Determine biases and uncertainties specific to the interface configuration.

In practice, interfaces show a higher reactivity than the individual storage configurations when high reactivity fuel is placed adjacent to one another across the interface. Care should be taken with interfaces to ensure that high reactivity fuel adjacent to one another across the interface is explicitly modeled and determined to be acceptable or not (if not, then restrictions should be specified to prevent these interfaces from occurring).

6.3 ABNORMAL AND ACCIDENT CONDITIONS

The licensee needs to consider all credible abnormal and accident conditions. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormal and accident conditions, as long as the conditions do not also result in a dilution of soluble boron. For PWR spent fuel pools that credit soluble boron, the limiting misload will be the accident which requires the highest soluble boron to ensure that the maximum k_{eff} does not exceed 0.95. The separate boron dilution accident is discussed in Section 7.3.

The following scenarios should be considered as part of postulated abnormal and accident conditions. Note that if a single accident scenario is clearly limiting, then other less limiting scenarios need not be explicitly calculated, but should be justified as being bounded. If the licensee determines that based on site specific rationale an accident condition is not credible, the submittal should include justification. If a design basis accident affects the inputs to the criticality analysis (e.g. if an earthquake results in physical changes to the neutron absorber material), then they should be considered.

6.3.1 Temperatures Beyond Normal Operating Range

The spent fuel pool has a normal operating range for the bulk temperature of the spent fuel pool water. Under accident conditions (loss of cooling) this temperature could be elevated beyond the normal operating range. Because the pool temperature is not a major contributor to reactivity and soluble boron credit can be taken for accident conditions, analysis should be performed for each storage configuration at temperatures between the maximum normal condition temperature (from Section 5.2.2.1) and boiling conditions in the pool at the fuel depth (typically around 124°C) with a void fraction up to 20% to confirm that higher temperature conditions are not limiting for the spent fuel pool.

6.3.2 Dropped and Mislocated Assembly

A dropped fresh fuel assembly on top of the spent fuel rack can either land horizontally on top of the rack or vertically outside the rack. The horizontal drop is not the most limiting accident condition due to the separation between the dropped assembly and the active fuel provided by the structure above the active fuel. This separation prevents neutronic coupling but even if there is some coupling the other accident conditions are more limiting. Therefore, provided the physical separation between active fuel regions is at least 12 inches to preclude neutronic coupling, no analysis of a horizontal fuel assembly on the top of the rack is necessary.

A mislocated fresh fuel assembly outside and adjacent to the storage racks (inside the pool wall) should also be evaluated if applicable, unless there is not enough room to physically fit a fuel assembly in between the racks and/or the pool wall.

6.3.3 Neutron Absorber Insert Misload

Some storage configurations may credit the neutron absorption capabilities of neutron absorber inserts, RCCAs, WABAs, BPRAs, etc. The potential exists for these devices to be inadvertently or accidentally removed and therefore should be investigated as part of the accident analysis. In most cases, this scenario will be bounded by the fresh fuel misload described in Section 6.3.4, but is nonetheless to be evaluated or justified as being bounded by other scenarios.

6.3.4 Assembly Misload

Misloading of a single fresh fuel assembly into an unapproved location is to be evaluated as a postulated accident scenario in PWR spent fuel pools. This accident scenario is postulated as an error on the part of the fuel crane operator to properly locate a fuel assembly in the correct storage location during fuel movement. For all storage configurations, an evaluation of a fresh fuel assembly of the maximum allowable enrichment, with no burnable absorbers should be evaluated in the storage location that provides the largest positive reactivity increase. For PWR spent fuel pools that credit soluble boron, the limiting misload will be the accident which requires the highest soluble boron to ensure that the maximum k_{eff} does not exceed 0.95.

For BWRs spent fuel pools that contain a homogeneous loading of the spent fuel storage rack with fuel with a limiting peak reactivity in each storage location (i.e., uniform loading), the misload event does not need to be considered. If a BWR spent fuel pool has multiple regions with different peak reactivity limits and/or storage configurations, then a misloaded bundle with the highest peak reactivity limit needs to be evaluated in the lower peak reactivity regions.

6.3.5 Multiple Assembly Misload

Additionally, there is the credible possibility of an error occurring in the selection of appropriate storage configurations such that a single initiation event can result in multiple fuel assemblies being misloaded, as evidenced by previous examples of multiple misloads. Whereas a single misload is typically a result of an error in the fuel handling selection or relocation of an assembly (i.e., picking up and moving an assembly other than the intended assembly), a single event resulting in multiple misloaded assemblies is typically the result of a planning or process error. Implementing a robust administrative control program for verifying used fuel assembly configurations and addressing potential non-compliant loading conditions therefore becomes vital to reduce the likelihood of occurrence a common cause failure resulting in misloading multiple assemblies.

It is important to have a multi-tier defense-in-depth program in place to prevent or mitigate the severity of a scenario where multiple assemblies are located into the wrong storage locations. Specific aspects of this defense-in-depth program include the following:

- Licensee controlled procedures, programs

- Event tree analysis
- Post-movement fuel assembly verification
- Storage cell blocking devices
- Analysis of multiple misload scenarios

Additional details of each of these elements are provided in the following sub-sections.

6.3.5.1 Licensee Controlled Administrative Programs

The spent fuel pool criticality analysis specifies the acceptable storage configurations and limits on the type and characteristics of fuel (i.e., burnup, enrichment, cooling time, etc.) to ensure compliance with the acceptance criteria. Adherence to these requirements is accomplished by the licensee prior to any fuel movement to ensure that the fuel assembly is placed in an acceptable location. There are many commercial software packages available that can assist the licensee in determining the acceptability of a fuel assembly to be placed in a location in accordance with the Technical Specification and the spent fuel pool criticality analysis.

The use of a QA validated software package provides an additional barrier to prevent a common-fault error of selecting the wrong location for multiple fuel assemblies. Additionally, the following features should be implemented to reduce the risk associated with the incorrect placement of multiple fuel assemblies in the spent fuel storage racks:

- Production of reports that show acceptability of fuel assembly locations
- Graphical representation (fuel assembly burnup, enrichment, cooling time against the limits for the storage configuration) to augment manual verification
- Visual, color-coded spent fuel pool maps showing acceptability of fuel assembly locations
- Pre-verification of planned fuel moves
- Detailed administrative procedures for implementation
- Training and qualification of engineers responsible for spent fuel assembly selection and verification
- Independent verification of the validated software output, such as Fuel Transfer Logs (FTLs)
- Training of responsible engineers prior to implementation of new storage configurations or Technical Specification loading curves

6.3.5.2 Event Tree Analysis

An event tree graphically represents the various accident scenarios that can occur as a result of an initiating event (i.e., a challenge to plant operation). Toward that end, an event tree starts with an initiating event and develops scenarios, or sequences, based on whether a plant system succeeds or fails in performing its function. The event tree then considers all of the related

systems that could respond to an initiating event, until the sequence ends in either a safe recovery or an accident event.

While an event tree analysis has not been historically applied to the credibility of an inadvertent criticality event in the spent fuel pool, there are several studies that have looked at the probability of a misloaded fuel assembly in a transport or storage cask [37, 40]. These studies can be used as guidance for creating an event tree analysis specific to a particular spent fuel pool configuration.

6.3.5.3 Post-Movement Assembly Verification

Verification of proper placement of fuel assemblies into approved storage locations after fuel movement can provide an independent confirmation of the acceptable storage configurations in the spent fuel pool. There are several potential processes that are suggested here that allow for additional defense-in-depth barriers to be implemented for ensuring proper placement of fuel assemblies:

- Visual verification of fresh versus spent fuel by fuel handling operators during fuel movement
- Administrative verification of high reactivity fuel assemblies prior to and after fuel movement.
- Post movement verification of fuel assembly locations

6.3.5.4 Storage Cell Blocking Devices

One simple approach to allow higher reactivity fuel to be placed in high-density racks is to designate specific storage cells to remain empty. However, placing either a fresh or spent fuel assembly in these storage locations under a multiple misload scenario would provide a significant reactivity addition. To prevent the misloading of multiple fuel assemblies into storage locations intended to be empty, blocking devices can be employed. Blocking devices are physical hardware installed into storage cells for the purposes of preventing the inadvertent placement of a fuel assembly into these locations. To ensure that maximum benefit and flexibility of these devices, the following criteria are recommended for blocking devices:

- Physically configured to prevent insertion of a fuel assembly in a fuel storage location,
- Requires specialized tools to install or remove the blocking device from a storage location,
- Designed to preclude falling inside a storage location or being dislodged from its position during normal operation,
- Contain a lock-in-place feature to prevent inadvertent movement,
- Support a load which will cause the underload trip sensor to activate. This is typically the load of one fuel assembly plus the handling tool,

- Allow for continued water flow through the storage cell.

Blocking devices do not need to be designed to prevent a dropped fuel assembly from entering the storage cell. However, the accident analysis must consider a single dropped fuel assembly in the storage cell with the blocking device.

6.3.5.5 Multiple Misload Analysis

The administrative controls and processes identified in the previous subsections can influence the credible scenarios that need to be evaluated via analysis to address the multiple misload from a single event. For example, a process check to ensure that a fresh fuel assembly is not selected when a used fuel assembly is intended to be selected (perhaps by confirming the physical appearance of the assembly) could eliminate the need to assume a multiple misload of fresh fuel for some scenarios. In this example, the most reactive misloaded fuel assemblies could be represented by fuel assemblies irradiated for a single cycle with the highest enrichment at a minimum burnup, since the process check would reduce the likelihood of misloading multiple fresh fuel assemblies.

6.3.6 Seismic Events

The spent fuel racks are designed to withstand the ground motions associated with the design basis seismic event. However, the spent fuel racks may sway or slide slightly in the spent fuel pool. These motions are typically small and do not result in a significant effect on reactivity. Typically, the spent fuel rack baseplate is designed to prevent the racks from coming too close together or from being damaged during seismic events. A straightforward approach for addressing seismic shifting is to assume that the racks are moved as close together as possible as allowed by the baseplate. For BWR spent fuel pools where the analysis is performed assuming as infinite array, seismic shifting does not require additional analysis or justification.

Additionally, the criticality analyst needs to consider movement or shifting of non-structural components (e.g., neutron absorber, inserts).

7 SOLUBLE BORON CREDIT

7.1 NORMAL CONDITIONS

10CFR50.68 [1] allows soluble boron credit of up to 5% Δk. That is, if credit is taken for soluble boron, k_{eff} of the spent fuel pool must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. Analyses performed in accordance with the guidance in Sections 5.1 and 5.2, including unborated water, must ensure that the maximum calculated k_{eff}, including all biases and uncertainties meet the k_{eff} limit of less than 1.0.

The criticality safety analysis must also demonstrate that if the spent fuel pool is flooded with borated water, k_{eff} must not exceed 0.95, at a 95% probability, 95% confidence level.

7.2 Accident Conditions

For conditions with soluble boron, the accident conditions in Section 6.3 should be evaluated at the minimum soluble boron concentration allowable under the site's Technical Specification. In other words the accident condition does not need to consider a simultaneous boron dilution event, per the double-contingency principle, if the accident does not also result in boron dilution. This is justified through application of risk insights, in that the probability of a significant boron dilution event (violating the minimum pool soluble boron concentration) is remote, and that there have not been any known cases of its occurrence in the history of nuclear power operations.

For the accident conditions that do not result in a corresponding boron dilution event, the analysis needs to determine the soluble boron necessary to ensure that the maximum calculated k_{eff}, including all biases and uncertainties, remains less than the regulatory k_{eff} limit of 0.95.

7.3 Boron Dilution

In the event the licensee is crediting soluble boron in the criticality safety calculation, a boron dilution accident needs to be evaluated. The boron dilution analysis initiates at the minimum allowable normal soluble boron concentration as described in the plant Technical Specifications and is consistent with the boron concentration assumed in the criticality analysis to maintain subcritical conditions (0.95) for normal conditions. The boron dilution analysis confirms the time needed for the dilution event to reduce the soluble boron concentration (from the plant technical specification concentration to the boron concentration assumed in the criticality analysis which shows that for normal operation the k_{eff} is less than 0.95) is greater than the time needed for actions to be taken to prevent further dilution. The boron dilution accident analysis must confirm that the operators have sufficient time to identify, diagnose and correct the cause of the inadvertent dilution, thereby preventing the maximum reactivity from exceeding the regulatory limit.

8 CALCULATION OF MAXIMUM K_{EFF}

The maximum k_{eff} must be determined for the spent fuel pools and new fuel vaults including uncertainties and biases. The maximum k_{eff} is determined by adding to the nominal calculated k_{eff} any biases that may exist in the methodology and the applicable uncertainties using the formula described in Equation 1:

$$k_{max} = k_{eff} + \sum_{i=0}^{m} Bias_i + \sqrt{\sum_{j=0}^{n} Uncertainty_j^2} \quad \text{(Equation 1)}$$

As can be seen from the above expression, uncertainties are statistically combined (assuming that such uncertainties are mutually independent) while biases are summed up. The biases and uncertainties that should be included are discussed within applicable sections of this document. These are summarized here for completeness:

Biases
Criticality Code Validation Bias
Moderator Temperature Bias
Design Basis Fuel Assembly Bias

Eccentric Positioning Bias
Depletion Code Bias (Applicant Depletion Code Bias)

Bias for Validation Gaps

Uncertainties
Fuel Manufacturing Tolerances
Rack Manufacturing Tolerances
Depletion Code Uncertainty
Burnup Uncertainty (BU)
Criticality Code Validation Uncertainty
Facility Structural and Material Uncertainties
Uncertainties for Validation Gaps
Monte Carlo Calculational Uncertainty

While this list generally identifies biases and uncertainties that are known to have an impact on the reactivity, the applicant also needs to include any site-specific biases and uncertainties that are not explicitly identified here.

Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of variations of parameters within the tolerance limits. If used, a sensitivity study should include all possible significant allowable variations within the material and mechanical specifications of the fuel and racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used. The recommended approach is to vary the parameter of interest to the maximum/minimum value allowed by the tolerance specification that maximizes reactivity. The reactivity effect of all tolerances are then combined statistically as indicated in Equation 1.

9 LICENSEE CONTROLS

9.1 LICENSEE CONTROLS

A licensee should establish controls that help to ensure that the conditions evaluated in the nuclear criticality safety analysis are and remain bounding to the current plant operating parameters. Appropriate licensee controls include plant procedures and programs that control storage configurations, and burnup/enrichment loading curves, and ensure that the storage of fuel is bounded by the criticality analyses.

9.2 PROCEDURAL CONTROLS

A licensee establishes procedural controls in order to ensure that used fuel is stored in accordance with the Technical Specifications, and to govern the planning and performance of fuel movements. These procedures implement the requirements for tracking the location of fuel assemblies in accordance with Special Nuclear Material (SNM) regulations and the spent fuel

pool criticality analysis. They also ensure proper assembly selection for core loading activities, thermal management, gamma flux, etc. In addition, programs and procedures are established to ensure that the licensee is following their QA plan. The QA program covers the use of codes for criticality analyses and software used to plan and implement fuel movements.

Procedural controls should be developed based upon the complexity of storage patterns in order to provide reasonable assurance of adequate public health and safety. The procedures may also affect the assumed accident conditions (see Section 6.3). The following are typical procedures and QA Program practices used by licensees. Additional procedures should be considered for more complex storage patterns.

- Pool Assembly Storage Planning
    - Fuel Characterization
        - Fuel reactivity category determination, for example,
            - Burnup
              (e.g., plots of burnup v enrichment to identify outliers, possible errors)
            - Enrichment
              (e.g., plots of burnup v enrichment to identify outliers, possible errors)
            - Decay time
        - Component inserts
    - Development of planned pool fuel assembly storage configurations
        - Use of verified software application to confirm planned pool configuration is in accordance with the criticality analysis
        - Independent verification of desired pool configuration
    - Development of Fuel Transfer Forms (FTF) to implement planned storage configuration
        - Use of verified software application to generate FTFs
        - Independent verification of FTFs

- Fuel Movement
    - Use of only approved FTFs
    - Activities of the Fuel Mover
    - Independent verification
      (the verifier should have no concurrent duties)
    - Independent FTF Step Verifier
      (the step verifier should have no concurrent duties)
    - Continuous communications between fuel mover, verifier, and step verifier
    - Personnel Training
    - Pre-job briefs

- Spent Fuel Pool
    - Bounding soluble boron requirement
      (use of a larger soluble boron concentration to provide more reactivity hold-down to minimize the effect of assembly misloadings)
    - Technical Specification for soluble boron surveillance

    - Neutron Absorber Panel material behavior monitoring program
- Software Requirements:
    - Independent review of software implementation and revision, testing and documentation is performed by an independent reviewer
    - Configuration controls to ensure integrity of executable files and data files
    - Cyber security controls prevent tampering / inadvertent changes
- Database Requirements:
    - Independent review and approval of all database updates
    - Procedures to ensure integrity of database prior to utilizing the data
- Confirmation of the applicability of the analysis of record for criticality safety

9.3 NEW (FUTURE) FUEL TYPES

It is common for licensees to periodically use newer fuel types that have more desirable in-reactor performance characteristics. However, it is impossible to predict the characteristics of fuel types that may be used in the distant future at the time of developing an application involving fuel criticality analyses. Therefore, the licensee should implement a process to assess (or check) newer fuel designs as they are implemented to ensure they are bounded by the existing design basis/analysis of record for the spent fuel storage rack or new fuel vault.

If an initial assessment determines that the new fuel type represents a potential change to the existing criticality safety design basis/analysis of record for the storage rack or vault, then a full criticality analysis should be performed. In accordance with 10 CFR 50.59, the full criticality analysis of the new fuel should include all credible configurations that have previously been analyzed for existing fuel types (e.g. normal, off-normal, and accident conditions) and interfaces with other fuel types.

The 10 CFR 50.59 [8] process is used to determine whether NRC review and approval is necessary prior to implementing the new fuel design.

9.4 PRE- AND POST-IRRADIATION FUEL CHARACTERIZATION

Fuel characterization is the process of ensuring that the actual nuclear fuel assemblies to be stored are bounded by the criticality analysis assumptions. This process should involve comparing actual fuel assembly and operating parameters to key assumptions utilized in the criticality analysis, and require further evaluation if the assumptions are not met. The intent is to ensure that changes in fuel design, core design, or cycle operation (both anticipated and unanticipated) are properly evaluated prior to storing the fuel.

Note that fuel characterization as discussed in this section is separate from the typical categorization of fuel assemblies according to initial enrichment, assembly-average burnup, and, in some cases, cooling time, that is used to determine where fuel assemblies may be placed in the spent fuel pool.

For any given fuel assembly, fuel characterization consists of two processes. The first process is pre-irradiation characterization, and its purpose is to review the design of the fuel assembly against the parameters assumed in the criticality analysis. Ideally, this is performed as part of the core design process. In any case, it is performed before the fuel in question is placed, for the first

time, in the new or spent fuel racks. For pressurized water reactors, the key inputs pertain to the fuel loading (fuel pellet mass, diameter, density, enrichment, etc.) and to the fuel-to-moderator ratio (fuel rod diameter, fuel rod pitch, etc.). Boiling water reactors should also consider the lattice itself (8x8, 9x9, 10x10, etc.), as well as the characteristics of the fuel channel. One acceptable method for BWR fuel characterization is the in-core k_∞ methodology. This method establishes infinite-lattice reactivity limits for each fuel storage region as part of the criticality safety analysis. Each unique fuel design is then validated against this reactivity limit to establish its acceptability for storage. For new BWR fuel designs, this includes an evaluation of whether there is an impact to the in-rack k_{eff} associated with the SCCG k_∞. Other characteristics to be considered will depend upon the nature of the criticality analysis itself. For example, if the analysis took credit for the initial presence of burnable absorbers in the fuel, then the characteristics of the burnable absorber (type, loading, and configuration) should also be considered.

The second process, called post-irradiation characterization, is only applicable if the criticality analysis in some way credits the in-reactor depletion of the fuel assemblies (i.e., burnup credit). If burnup is credited, a process should be implemented to ensure that the fuel was depleted in a manner consistent with the assumptions in the criticality analysis.

Post-irradiation characterization is concerned with ensuring that certain parameters assumed in the criticality analysis do, in fact, bound the actual operating history of the fuel assemblies. Parameters to be considered will depend on the methods and assumptions of the analysis. Some licensees may be able to verify that the reactor operated within the bounds of the analysis based on comparison to past operation, while others may need to verify more detailed reactor parameters or assembly specific parameters, such as:

- Axial burnup shape (if using Option 2 or 3 in Section 5.1.3)
- Moderator temperature
- Fuel temperature
- Soluble boron (burnup averaged)
- Control rod insertion
- Burnable absorber presence (particularly if discrete, removable burnable absorbers are used)

Ideally, the process of post-irradiation characterization is initiated as part of the core reload design process, so that potential non-compliances with the criticality analysis can be identified early on, and possible changes to the fuel or core design can be made to mitigate the concerns. Post-irradiation characterization should be finalized following actual reactor operation, to ensure that there were no significant operating differences from that assumed during the core reload design process. In particular, a re-evaluation of the post-irradiation characterization should be considered if such differences result in a significant hardening of the neutron spectrum experienced by fuel assemblies or alter the axial power shape in the fuel assemblies long enough

to significantly impact the axial burnup shape of the fuel at discharge. Specifically, this could include:

- Operation for a significant period of time at reduced power or with control rods inserted more than during normal operations
- Changes to plant configuration that result in higher-than-expected reactor coolant temperatures
- Early cycle shutdown impacting cycle average quantities, such as soluble boron

For both pre- and post-irradiation characterization, any differences that are not explicitly bounded by the criticality analysis should be evaluated to determine if there is any impact on the criticality analysis, in accordance with other licensee programs (e.g., 10 CFR 50.59 [8] process, operability evaluation).

10 REFERENCES

10.1 REGULATIONS

1. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.68, Criticality Accident Requirements.

2. Title 10 of the *Code of Federal Regulations* (10 CFR) 70.24, Criticality Accident Requirements.

3. Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants

4. Title 10 of the *Code of Federal Regulations* (10 CFR) 52, Licenses, Certifications, and Approvals for Nuclear Power Plants.

5. Not Used

6. Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

7. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, Technical Specifications.

8. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, Changes, Tests and Experiments.

10.2 STANDARDS

9. ANSI/ANS-8.1-1998; R2007, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors”.

10. ANSI/ANS-8.24-2007, "Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations" (reaffirmed 2012).

11. ANSI/ANS-8.27-2015, "Burnup Credit for LWR Fuel."

10.3 NUREGs and NUREG/CRs

12. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3, March 2007.

13. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.1.2, "New and Spent Fuel Storage," Revision 4, March 2007.

14. J.C. Dean and R.W. Tayloe, Jr, *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*, NUREG/CR-6698, Science Applications International Corporation, U.S. Nuclear Regulatory Commission, January 2001.

15. D. E. Mueller, K. R. Elam, and P. B. Fox, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*, NUREG/CR-6979 (ORNL/TM-2007/083), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 2008. (ADAMS Accession No. ML082880452)

16. J. C. Wagner and C. V. Parks, *Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses,* NUREG/CR-6781 (ORNL/TM-2001/272), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, January 2003 (ADAMS Accession No. ML030290585)

17. C.V. Parks, M. D. DeHart , and J. C. Wagner, Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel, NUREG/CR-6665 (ORNL/TM-1999/303), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2000.

18. C. E. Sanders and J. C. Wagner, *Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit*, NUREG/CR-6760 (ORNL/TM-2000/321), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2002.

19. C. E. Sanders and J. C. Wagner, *Parametric Study of the Effect of Control Rods for PWR Burnup Credit*, U.S. Nuclear Regulatory Commission, NUREG/CR-6759 (ORNL/TM 2001/69), Oak Ridge National Laboratory, February 2002.

20. J. C. Wagner, M. D. DeHart, and C. V. Parks, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*, U.S. Nuclear Regulatory Commission, NUREG/CR-6801 (ORNL/TM-2001/273), Oak Ridge National Laboratory, March 2003.

21. B.B. Bevard, J.C. Wagner, and C.V. Parks, *Review of Information for Spent Nuclear Fuel Burnup Confirmation,* U.S. Nuclear Regulatory Commission, NUREG/CR-6998, (ORNL/TM-2007-229), Oak Ridge National Lab, December 2009.

22. J.C. Wagner, *Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask*, U.S. Nuclear Regulatory Commission, NUREG/CR-6955 (ORNL/TM-2004/52), Oak Ridge National Lab, January 2008

23. J.D. Brewer, P.J. Amico, S.E. Cooper, S.M.L. Hendrickson, *Preliminary Qualitative Human Reliability Analysis for Spent Fuel Handling,* U.S. Nuclear Regulatory Commission, NUREG/CR-7017 (SAND2010-8464P), Sandia National Laboratories, February 2012

10.4 OTHER

24. NRC Memorandum from L. Kopp to T. Collins, Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," August 19, 1998.

25. DSS-ISG-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools.

26. "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC(95)3, Volume IV, Nuclear Energy Agency, OECD, Paris, Updated every year.

27. *Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation – Revision 2.* EPRI, Palo Alto, CA: 2019. 3002016888

28. *Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty – Revision 1.* EPRI, Palo Alto, CA: 2017. 3002010613

29. *Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications*: 2009 Edition. EPRI, Palo Alto, CA: 2009. 1019110

30. John Wagner, "Impact of Soluble Boron for PWR Burnup Credit Criticality Safety Analysis," *Trans. Am. Nucl. Soc.*, **89**, November 2003.

31. *Sensitivity Analysis for Spent Fuel Pool Criticality – Revision 1*, EPRI, Palo Alto, CA: 2017, 30022008197.

32. Mark M. D. DeHart, *Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages*, ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996.

33. Dale Lancaster and Albert Machiels, "Use of EPRI Depletion Benchmarks For Transport Criticality Burnup Credit," Proceedings of the International Symposium on the Packaging and Transportation of Radioactive Materials, PATRAM 2013, San Francisco, August 18-23, 2013, Institute of Nuclear Materials Management (INMM), Deerfield, IL.

34. *Millstone Unit 2 Spent Fuel Pool Criticality Analysis with No Credit for Boraflex,* Dominion Resources Services, Inc., November 2012, NRC Accession #ML12362A392. (proprietary)

35. R. Cacciapouti, et. al., *Determination of the Accuracy of Utility Spent-Fuel Burnup Records,* EPRI, Palo Alto, CA: 1999, TR-112054

36. M.D. DeHart, *Parametric Analysis of PWR Spent Fuel Depletion Parameters for Long-Term-Disposal Criticality Safety*, ORNL/TM-1999/99, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, August 1999.

37. A. Dykes, *Criticality Risks During Transportation of Spent Nuclear Fuel: Revision 1.* EPRI, Palo Alto, CA: 2008. 1016635

38. S. Sidener, J. P. Foster, et al, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)", WCAP-15063-P-A, Westinghouse Electric Company LLC., Pittsburgh, PA, July 2000 (proprietary)

39. M. Ouisloumen, H. Huria, et al, "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A, Westinghouse Electric Company, Pittsburgh, PA, August 2004. (proprietary)

40. *Estimating the Probability of Misload in a Spent Fuel Cask,* U.S. Nuclear Regulatory Commission, November 2011, NRC Accession # ML113191144

41. J. Hannah, W. Metwally, V. Mills, "Uncertainty Contribution to Final In-Rack K(95/95) from the In-Core K_∞ Criterion Methodology for Spent Fuel Storage Rack Criticality Safety Analyses," Procedings of the Advances in Reactor Physics to Power the Nuclear Renaissance, PHYSOR 2010, Pittsburgh, Pennsylvania, USA, May 9-14, 2010, American Nuclear Society (ANS), LaGrange Park, IL

42. GL 78-11, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications"

43. GL 79-04, "Modifications to NRC Guidance, 'Review and Acceptance of Spent Fuel Storage and Handling Applications'"

44. Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants,* U.S. Nuclear Regulatory Commission, July 2000.

45. C.E. Beyer and P.M. Clifford, *Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard,* PNNL-18212, Revision 1, June 2011

APPENDIX A: COMPUTER CODE VALIDATION

A.1 CRITICALITY CODE VALIDATION USING FRESH FUEL EXPERIMENTS

The criticality computer codes used for the criticality safety analysis should be validated using measured data. This validation should consist of five steps:

1. Identify range of parameters to be validated
2. Select critical experiment data
3. Model the experiments
4. Analyze the data
5. Define the area of applicability of the validation and limitations

NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," provides guidance on one approach for performing the validation [14].

A.1.1 Identify Range of Parameters

The first step is to identify the range of parameters to be validated. Examples of parameters that should be selected include type of fissile isotope, enrichment of the fissile isotope, fuel chemical form, etc. These selected parameters will lay the foundation for determining the area of applicability of the validation. Specifically the neutronic behavior is influenced by the following parameters, which should be covered by the selected experiments:

- Isotopic Content
    - o Experiments should cover materials representative of the rack structure (e.g., stainless steel), materials in the surrounding geometry (e.g., water/concrete), materials representative of the cladding (e.g., zirconium), fissile isotopes in the applicable enrichment range (e.g., ^{235}U for low enriched UO_2, ^{239}Pu for MOX), water and material temperatures, and others if applicable: boron for the soluble boron and absorber plates, gadolinium if peak reactivity is used (BWRs) or if credit for gadolinium in fresh fuel is used, and/or silver/indium/cadmium if control rods are used in the criticality analysis.
- Spectrum
    - o The spectrum can be affected by geometry and storage rack design (e.g., a region with a flux trap design or a region with no flux traps), therefore, the critical experiments should cover a range of spectra. The spectrum range can be quantified by an index such as the energy of the average lethargy of neutrons causing fission (EALF) or average energy group causing fission (AEG). Historical indices used include hydrogen-to-fissile atoms ratio (H/X), and fuel-to-moderator ratio.

- Geometry
  - Key geometric features are the fuel pin pitch, pellet or clad diameter, assembly separation, and boron areal density.

A.1.2 Selection of Critical Experiments

The features listed above can be covered in the validation by selection of an adequate number and range of critical experiments. The OECD/NEA *International Handbook of Evaluated Criticality Safety Benchmarks Experiments* [26] and the HTC critical experiments [15] are considered appropriate references for criticality safety benchmarks. The handbook has reviewed the benchmarks and carefully evaluated the uncertainties in the experiments. Other sources for critical experiments may also be acceptable and should include an estimate of the uncertainty in the critical experiments. The selection of critical experiments to be included in the validation should include benchmark experiments from multiple facilities and experiment series to minimize the possibility of a facility-specific or experiment series-specific bias.

The set of experiments selected should support determination of a statistically appropriate validation. Care should be taken in selecting critical experiments so that trends can be identified and addressed.

The applicant needs to include in the validation fresh UO_2, MOX and HTC experiments. The code bias and uncertainty needs to be determined both with UO_2 experiments alone and with HTC and MOX experiments included. The appropriate bias and uncertainty from each of these cases are included for fresh and spent fuel, respectively.

A.1.3 Modeling the Experiments

Section 2.3 of NUREG/CR-6698 [14] states that it is acceptable to "choose to use input files generated elsewhere to expedite the validation process". It should be emphasized, however, that although the input files may initially come from somewhere else, the modeling of the critical experiments should match, as closely as possible, the modeling used in the criticality safety analysis (e.g. comparable level of geometric modeling detail). Additionally, the analyst must verify and ensure the accuracy of the critical experiment models used in the validation, even if provided from a third party reputable source.

A.1.4 Analysis of the Critical Experiment Data

NUREG/CR-6698 [14] defines the steps of "Analyze the data" as:

1. Determine the Bias and Bias Uncertainty
2. Identify Trends in Data, Including Discussion of Methods for Establishing Bias Trends
3. Test for Normal or Other Distribution
4. Select Statistical Method for Treatment of Data
5. Identify and Support Subcritical Margin
6. Calculate the Upper Safety Limit

NUREG/CR-6698 [14] provides equations for the determination of the bias and bias uncertainty. These equations weight the experiments by the experimental uncertainty. It is important that the experimental uncertainty is reasonable to ensure meaningful trend analysis. It is noted that inaccurate experimental uncertainties may result in inaccurate trend results. The uncertainties provided in the OECD criticality benchmark handbook [26] are sufficient for this purpose so the statistical approach defined in NUREG/CR-6698 [14] should be used.

It is important to look over the calculated biases for trends in the data. At a minimum statistical analysis should be performed to check for a trend in the bias due to differences in spectrum and enrichment. Seeking more trends is recommended. However, it is noted that trends in some parameters may actually be the result of trends in spectrum or enrichment, i.e. dependent parameters that are embedded in the data. Trends on the following parameters need to be considered:

- Energy spectrum (e.g., EALF, AEG)
- Enrichment
- Soluble boron
- Absorber content

While this list generally identifies the important trends to evaluate, the applicant also needs to address any potentially site-specific features (e.g., AgInCd content, temperature) that are not explicitly identified here.

The equation in Section 8 can be used to calculate the maximum k_{eff}. Alternatively, the method in NUREG/CR-6698 [14] for determining an upper safety limit on k_{eff} which includes the uncertainty determined from the critical experiments may be used. The uncertainties from the critical benchmark analysis can be statistically combined with other uncertainties such as manufacturing tolerances (see Section 8). The bias and uncertainty determined from the critical experiments are applied either as a function of the trending parameters or as conservative values that cover the desired range(s).

A.1.5 Area of Applicability

The validation of the calculational methodology for nuclear criticality safety analyses covers an area of applicability, or also known as the “benchmark applicability” [10]. The criticality safety analysis should define and document this area of applicability.

The following subsection provides further detail and guidance of how to apply and use the area of applicability in the nuclear criticality analysis.

<u>Limitations and Conditions</u>

In the validation, a range of parameters should be established that are important to criticality and that reflects the range of conditions, normal and abnormal, that the fuel assemblies could experience in the new fuel vault and the spent fuel pool. Parameters, per ANSI/ANS-8.24, that should be considered include [10]:

- Nuclide composition and chemical form of all associated materials;
- Geometry (e.g., lattice pattern, spacing, reflector location, size, shape, and homogeneity or heterogeneity of the system); and
- Characterization of the neutron energy spectrum.

Again, the selection of the range of these parameters should be determined based on both normal and credible abnormal new fuel vault and spent fuel rack conditions.

Trend Evaluation

Part of the validation is to identify whether the bias has a dependency on any of the parameters in the area of applicability. The parameters selected for trending evaluation should be based on the characteristics of the system under consideration. [10]

If a significant trend exists in a bias of an important parameter in the validation of the code, then the criticality safety analyses should appropriately address the trend when determining the appropriate bias and uncertainty to utilize.

Extrapolation

If the experiments do not fully cover the analyzed system, then it may be possible to extrapolate the validation. The area of applicability may be extended beyond the range of experimental conditions by employing the trends in the bias. NUREG/CR-6698 [14] provides further guidance for extending trends and accounting for increasing uncertainty if there are insufficient critical experiments.

For the new fuel vault analysis, the fresh fuel validation is applicable in the fully flooded condition. There are limited critical benchmark experiments to cover the optimum moderation condition for the new fuel vault. New fuel vault racks are typically designed as part of an open rack structure (storage cell walls do not extend the length of the fuel assembly), but have the same materials, fuel geometry and general structure as the spent fuel pool racks. Despite this limitation, it is recommended to apply the criticality code validation using fresh fuel experiments to the optimum moderation condition. The validation (for the new fuel vault) needs to include benchmark experiments that cover the energy spectrum (i.e., EALF) of the optimum moderation condition.

A.2 Depletion Code Validation

Validation of used fuel depends on determining the accuracy of the depletion code in predicting the reactivity of a fuel assembly during operation. This section provides several approaches for both PWR and BWR racks to explicitly quantify a depletion uncertainty.

A.2.1 Validation Using Measured Flux Data from PWR Power Reactors

PWR depletion benchmarks were developed by EPRI [27, 28] using a large set of power distribution measurements to ascertain reactivity biases. The predicted reactivity of the fuel assemblies was adjusted to find the best match between the predicted and measured power

distribution. EPRI used 680 flux maps from 44 cycles of PWR operation at 4 PWRs to infer the depletion reactivity [28]. The depletion reactivity has been used to create 11 benchmark cases for various burnups up to 60 GWd/T and 3 cooling times 100 hour, 5 years, and 15 years. All of these benchmark cases should be analyzed with the code set (depletion and criticality codes) to be used in the criticality analysis to establish a bias for the depletion reactivity. The uncertainty in the benchmarks should be used as the depletion reactivity uncertainty. These biases and uncertainties cover both the isotopic content uncertainty and the worth uncertainty associated with depletion. They account for all the changes from the initial fresh fuel condition. A companion EPRI report [27] provides an example of performing the validation of the depletion and criticality code using the EPRI Depletion Benchmarks [28].

The 95/95 tolerance limit of the depletion reactivity decrement (i.e., uncertainty), as a percent of the reactivity decrement and the additional bias for NRC safety margin, are provided in the EPRI Benchmark Report [28], and reproduced here for completeness[2]. As evident from the table, the uncertainty increases as a function of the burnup.

Tolerance as Percentage of Absolute Value of Depletion Reactivity Decrement

Burnup (GWd/MTU)	10	20	30	40	50	60
95/95 Tolerance Limit (pcm)	348	537	654	752	831	888
95/95 Tolerance Limit (% of depletion)	3.05	2.66	2.33	2.12	1.95	1.81
Additional Bias (%)	0.0	0.0	0.0	0.15	0.35	0.54

The steps for validation include:

1) Perform analysis of EPRI Depletion Benchmarks (11 Benchmark Experiments as outlined in [28]) using applicant's depletion and criticality codes, at 100 hours, 5 year and 15 years cooling times,

2) Calculate the difference between the applicants calculated reactivity decrements and the measured reactivity decrements (calculated minus measured) contained in the 11 Benchmark Experiments, tabulated in Tables C-3 to C-5 in Reference 28. Then, determine the maximum positive difference to be applied as an additional bias, defined as the Applicant Depletion Code Bias,

3) Include the Applicant Depletion Code Bias (from Step 2) in the overall calculation of maximum k_{eff} and apply the Additional Bias values as a function of burnup to ensure NRC safety margins,

4) Evaluate the EPRI depletion uncertainty to be statistically combined with other uncertainties for inclusion in the overall calculation of the maximum k_{eff}.

[2] It should be noted that Table 10-2 in Reference 28, EPRI Benchmark report contains both bias and tolerances (uncertainty). The experimental biases for EPRI benchmarks are already added to the measured reactivity values tabulated in Tables C-3 to C-5 in Reference 28.

Note that linear interpolation between the burnup values is acceptable to calculate the corresponding EPRI uncertainty and additional NRC bias for specific fuel assembly burnups.

A.2.2 Validation Using Measured Critical Data from BWR Power Reactors

Each time a BWR is loaded with fresh fuel during an outage, a cold critical control rod configuration is predicted using a lattice physics and core simulator code package. To assess the accuracy of depletion codes in calculating used fuel isotopes and their corresponding reactivity, the criticality analyst can compare critical conditions from power plant startups with predicted eigenvalues. Control rods are then withdrawn from the core using the prescribed sequence until the core reaches a critical state. The core period, temperature, and control rod positions are then fed back into the lattice physics/core simulator package to obtain the calculated eigenvalue for the measured critical configuration.

The use of such measured critical data is applicable because the cold critical conditions are very similar to the rack conditions in that:

1. The moderator temperature and density is about the same as the rack,
2. The control rods which are being removed during the startup are similar (e.g. in their neutronic effects) to absorber plates in rack,
3. The fuel itself is the same (pellet diameter, pin diameter, rod pitch, etc), and
4. The average burnup is similar to the peak reactivity burnup used in the criticality analysis.

As the core is in a cold, unvoided, mostly controlled state for these measurements, the variability of the measured eigenvalue to factors other than isotopic variations in the fuel (such as fuel temperature, moderator temperature, power density, instantaneous void fraction, etc.) is minimized. Additionally, as the cold critical measurements either involve a small local subset of control rods and their adjacent bundles or typical control rod withdrawal sequences involve banked rod movements to significantly extracted positions at several distinct and spatially separate locations in the core, the results of the corresponding calculation will be sensitive to the fidelity of the lattice physics code in assessing local isotopic compositions and reactivities. Thus, measured critical conditions are capable of providing benchmark experiments for spent fuel pool conditions.

By comparing the measured data to calculated results over a large range of startup experience, a bias (Δk_{SUb}) and bias uncertainty (Δk_{SUu}) can be assessed for the lattice physics/core simulator package.

A.4 ALTERNATE CODE VALIDATION

If a code (the primary code) is not capable of directly modeling the benchmark experiments, then an intermediary code (i.e., a secondary code) may be used that is validated to the benchmark experiments, and to which the primary code is validated. The primary code (code used for the criticality safety analyses) should still be capable of accurately modeling all the important neutronic and geometric aspects of storage. The secondary code should be validated against benchmark experiments that are similar to the neutronics and geometry of the criticality safety

analysis in accordance with Section A.1. The primary code can then be validated by benchmarking to the secondary code over a range of parameters (neutronic and geometric) that bound the range of parameters for the criticality safety analysis. Those parameters that are important to be validated between the primary and secondary code include:

- Enrichment,
- Burnup,
- Energy Spectrum,
- Absorber areal density,
- Soluble boron content, and
- Storage rack geometry.

The total biases and uncertainties of the maximum k_{eff} needs to include the biases and uncertainties from both the primary code to secondary code validation, and the secondary code validation to benchmark experiments. An additional bias or uncertainty may need to be applied for any gaps between the primary and secondary code validation or capabilities.

APPENDIX B: EXAMPLE OF THE REACTIVITY IMPACT OF FUEL ROD CHANGES WITH DEPLETION

As nuclear fuel is irradiated in the reactor, the fuel rods undergo small physical changes. These changes are driven by the behavior of the ceramic uranium dioxide fuel pellets as they generate energy. This appendix addresses whether the small physical changes fuel rods undergo during operation in the reactor core have an impact on the reactivity of the fuel in the SFP environment. The specific physical changes of interest are changes to fuel density, clad outer diameter (OD), and clad thickness. It should be noted that changes in the fuel pellet diameter is also captured in this analysis because the fuel pellet diameter is directly correlated to fuel density.

Calculations were performed with NRC-approved fuel performance and fuel depletion codes [38, 39] for a Westinghouse 3-loop PWR core operating with a 15x15 fuel lattice. The analysis included both IFBA and non-IFBA fuel, modeled fuel pellets at both the center and top of the assembly, and covered a burnup range from 0 – 62 GWd/MTU. The analysis was divided into three major sections:

1. Modeling the physical behavior of fuel rods during operation using the PAD code to determine the minimum and maximum values for fuel density, clad OD, and clad thickness;
2. Modeling the depletion of the fuel with the PARAGON using the minimum and maximum values calculated with PAD to determine fuel assembly isotopic inventory; and
3. Determining the reactivity impact due to the physical changes in the fuel over depletion.

The physical behavior of the fuel rod dimensions during operation is provided in Figure B-1, through Figure B-4. These figures are based on calculational results and represent plant-specific values; however, their importance is in the demonstration of the behavior of fuel rods over depletion. The behaviors exhibited by the pellet and clad are not specific to any reactor design, they are applicable to all UO_2 fueled plants. Therefore, the values on the y-axis are omitted because the general behaviors generated by depletion are applicable to all fuel rods.

Figures B-1 and B-2 show the density and diameter changes of the fuel pellet with respect to depletion. Figure B-1 shows that the pellet density initially increases and then decreases over depletion. Figure B-2 shows the pellet diameter changes over depletion. Both figures clearly demonstrate the two widely known phenomena of fuel densification and fuel swelling. Early in reactor operation the heat generated by fission causes fuel to densify and the fuel pellet diameter to correspondingly decrease. As operation continues, the fissions products produced in the pellet cause the pellet diameter to expand and the fuel density to decrease. It should be noted that while the fuel density is changing, this is solely due to changes in pellet dimensions as the mass within the fuel is unchanged.

Figures B-3 and B-4 show the changes in fuel clad thickness and outer diameter due to fuel depletion. The behaviors of these parameters align with the behavior of the fuel pellets. Initially the clad OD decreases, thickening slightly, until the clad comes into contact with the fuel pellet. Once the clad and pellet are touching the clad expands and thins as the fuel pellet swells.

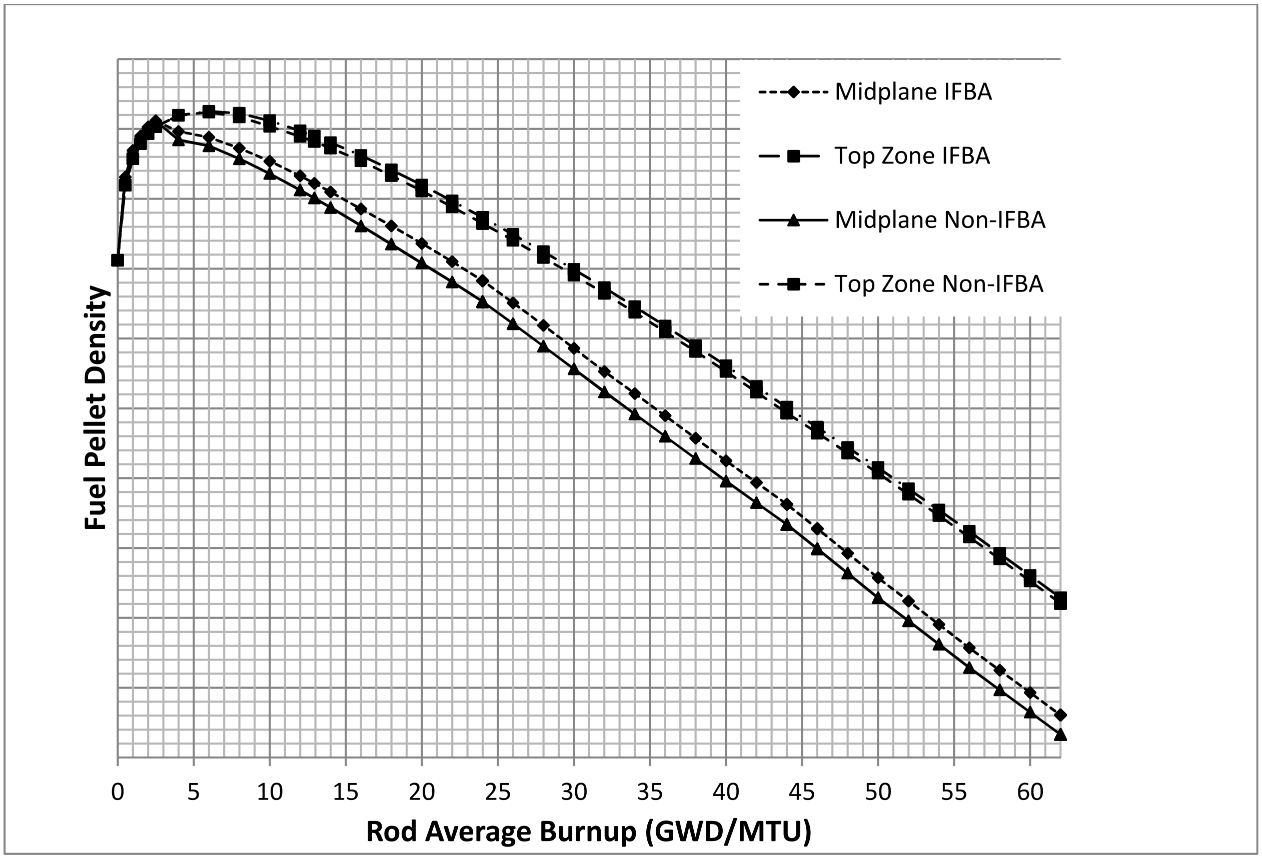

Figure B-1: Fuel Density Behavior with Depletion

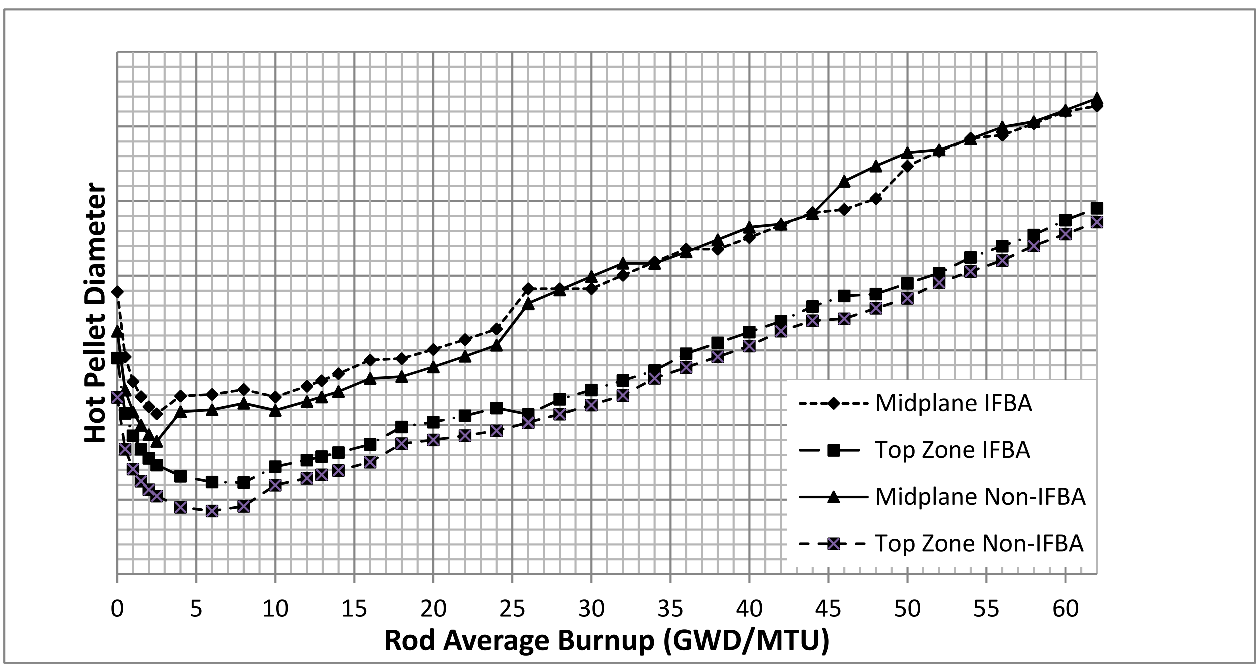

Figure B-2: Fuel Pellet Diameter Behavior with Depletion

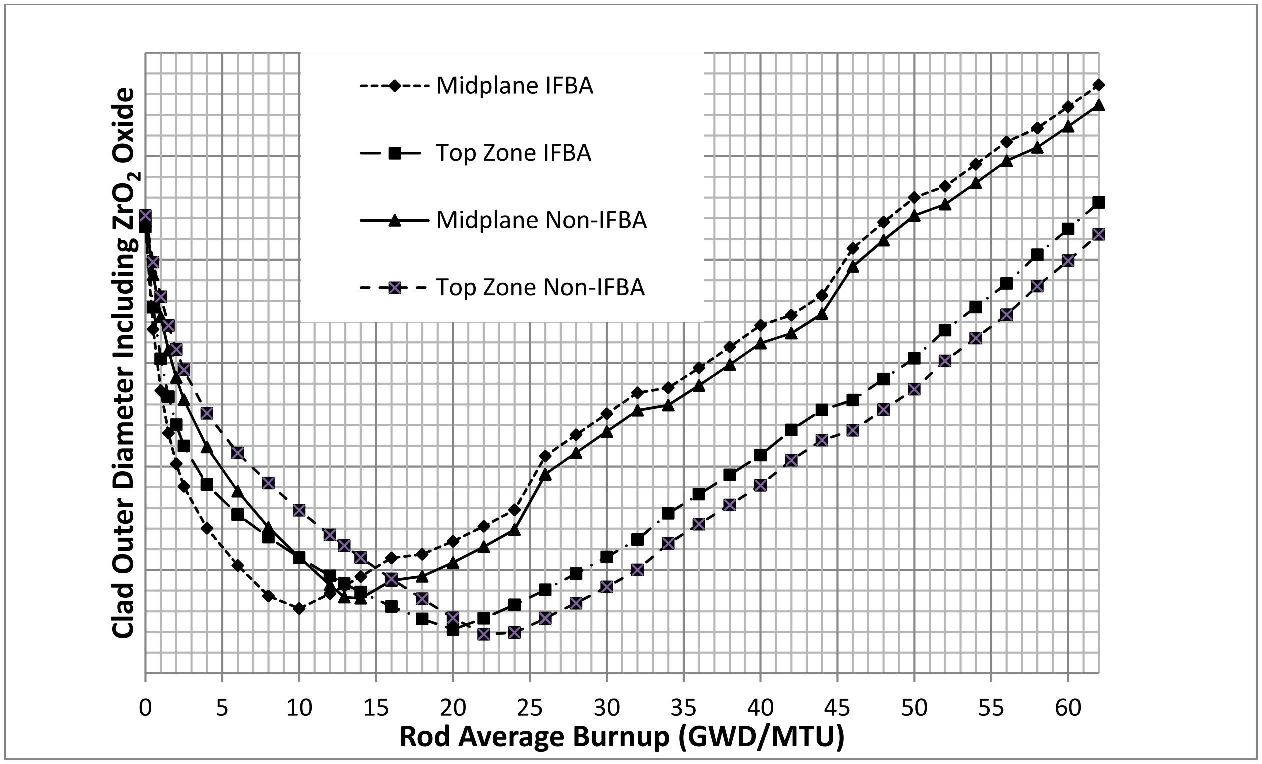

Figure B-3: Clad Outer Diameter Behavior with Depletion

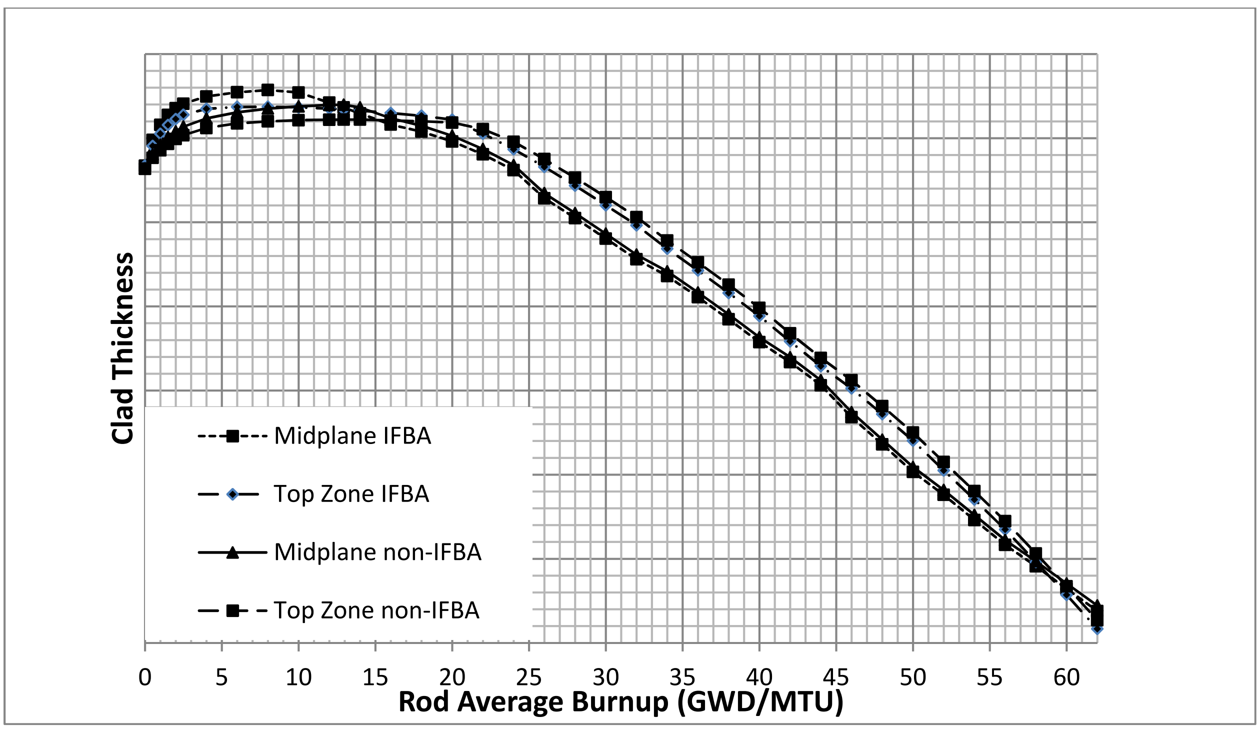

Figure B-4: Clad Thickness Behavior with Depletion

Based on the pellet and clad data developed above, depletion and reactivity calculations were performed with PARAGON and the SCALE 5.1 KENO v.A module respectively. These calculations used the minimum and maximum pellet and clad data points to develop conservative

estimates of the reactivity impact of the fuel changes during depletion. The depletion calculations assumed either the maximum or minimum value for the parameter in question throughout depletion. The parameters are each treated individually in determining the reactivity impact, although it should be noted that the fuel density and fuel pellet diameter are treated together because they are different aspects of the same parameter.

The reactivity calculations were performed at 5.0 weight percent, 62 GWd/MTU using KENO models with 26 axial nodes, an All-Cell (4-out-of-4 uniform burnup) model with a developed-cell style rack without neutron absorber. The following calculations were performed for both IFBA and Non-IFBA fuel:

- Base Case, Nominal Dimensions
- Maximum Fuel Density
- Maximum Clad Outer Diameter
- Maximum Clad Thickness
- Minimum Clad Outer Diameter Pre-Condition
- Minimum Clad Outer Diameter
- Minimum Fuel Density Pre-Condition
- Minimum Fuel Density
- Minimum Clad Thickness

The study calculated reactivity using Eq. B-1. The results of the reactivity calculations are provided below.

$$\Delta k = \left(k_{KENO} - k_{BASE}\right) + 1.645\sqrt{\sigma_{KENO}^2 + \sigma_{BASE}^2} \qquad \text{Eq. B-1}$$

<u>Fuel Pellet Density and Outer Diameter</u>
The density and outer diameter of fuel pellets change with fuel burnup as the pellet goes through densification and then swelling. The reactivity associated with conservatively modeling each phenomena was reviewed together.

- Minimum pellet diameter + maximum density; and
- Maximum pellet diameter + minimum density

Table B-1: Fuel Pellet Density Changes

Case Name	**Δk**
non-IFBA maximum pellet density	0.00223
non-IFBA minimum pellet density	-0.00375
IFBA maximum pellet density	0.00165
IFBA minimum pellet density	-0.00321

As anticipated, the results of the pellet density and diameter calculations show that increasing density slightly increases fuel reactivity.

Fuel Clad Outer Diameter

Fuel Clad Outer Diameter changes based on pellet thickness. Fuel clad OD decreases at BOC due to pellet densification and fuel clad OD increases from ~15 GWd/MTU due to pellet swelling.

Table B-2: Fuel Clad Outer Diameter Changes

Case Name	**Δk**
Non-IFBA maximum clad OD	0.00129
Non-IFBA minimum clad OD	-0.00506
IFBA maximum clad OD	0.00124
IFBA minimum clad OD	-0.00554

Table B-2 shows a large negative reactivity impact associated with fuel depletion and storage at the minimum clad OD. This is expected because a small OD softens the spectrum in both the SFP and the reactor. A softer spectrum will reduce parasitic absorption by ^{239}U, reducing Plutonium production. Lowering the amount of Plutonium produced requires that more ^{235}U be fissioned to reach the same burnup level, thus causing the isotopic inventory to be less reactive than an assembly depleted without a reduced OD. While having a smaller OD will increase the reactivity in the SFP, the SFP impact does not overcome the in-core impact, as seen from the maximum clad OD being a positive overall impact.

Clad Thickness
Clad thickness changes with depletion, at BOC clad thickens with fuel densification and oxide buildup. As shown in Figure B-4 the clad starts to 'thin' after approximately 15 GWd/MTU.

Table B-3: Clad Thickness Results

Case Name	**Δk**
Non-IFBA maximum clad thickness	0.00032
Non-IFBA minimum clad thickness	0.00223
IFBA maximum clad thickness	0.00021
IFBA minimum clad thickness	0.00237

The results of the clad thickness calculations show that minimum clad thickness is limiting. This is because the minimum clad thickness maximizes the size of the fuel rod gap. This lowers conductivity and therefore increases fuel rod temperatures.

Holistic Impact of Fuel Changes

The results of the reactivity calculations indicate that there are positive reactivity impacts from certain changes in fuel geometry when looked at in isolation. However, there are also individual fuel geometry changes which are negative reactivity impacts. Because none of these parameters are truly independent of the other parameters, an additional set of cases was performed incorporating all of the changes associated with fuel depletion together. This calculation provides a more accurate assessment of the actual neutronic importance of these changes.

To provide a better estimate of the true reactivity impact, two calculations for both IFBA and non-IFBA fuel were performed. The first case assumed the fuel pin geometry associated with peak fuel density and the second case assumed end of life conditions (62 GWd/MTU). The results of these calculations are provided in Table B-4.

Table B-4: Overall Reactivity Impact of Fuel Changes

Case Name	Δk
non-IFBA EOL Case	-0.00093
non-IFBA Maximum Density Case	-0.00123
IFBA EOL Case	-0.00040
IFBA Maximum Density Case	-0.00409

The results in Table B-4 demonstrate that each individual fuel geometry change has a small positive or negative impact on fuel reactivity. However, when all of the changes are looked at holistically, the overall impact of fuel geometry changes with depletion is small. These results are not unexpected because they align with standard procedures for performing fuel management calculations. These procedures essentially ignore fuel geometry changes, which would not be the case if they had a significant role (either positively or negatively) on fuel reactivity. This analysis for 15x15 PWR fuel and its alignment with general fuel management practices concludes that fuel geometry changes with depletion do not need to be explicitly modeled in depletion calculations. Studies of similar scope can be used to demonstrate that this conclusion is applicable to other fuel lattices for both BWRs and PWRs.

APPENDIX C: CRITICALITY ANALYSIS CHECKLIST

The criticality analysis checklist is completed by the applicant prior to submittal to the NRC. It provides a useful guide to the applicant to ensure that all the applicable subject areas are addressed in the application, or to provide justification/identification of alternative approaches.

The checklist also assists the NRC reviewer in identifying areas of the analysis that conform or do not conform to the guidance in NEI 12-16. Subsequently, the NRC review can then be more efficiently focused on those areas that deviate from NEI 12-16 and the justification for those deviations.

Subject	Included	Notes / Explanation
1.0 Introduction and Overview		
Purpose of submittal	YES/NO	
Changes requested	YES/NO	
Summary of physical changes	YES/NO	
Summary of Tech Spec changes	YES/NO	
Summary of analytical scope	YES/NO	
2.0 Acceptance Criteria and Regulatory Guidance		
Summary of requirements and guidance	YES/NO	
Requirements documents referenced	YES/NO	
Guidance documents referenced	YES/NO	
Acceptance criteria described	YES/NO	
3.0 Reactor and Fuel Design Description		
Describe reactor operating parameters	YES/NO	
Describe all fuel in pool	YES/NO	
Geometric dimensions (Nominal and Tolerances)	YES/NO	
Schematic of guide tube patterns	YES/NO	
Material compositions	YES/NO	
Describe future fuel to be covered	YES/NO	
Geometric dimensions (Nominal and Tolerances)	YES/NO	
Schematic of guide tube patterns	YES/NO	
Material compositions	YES/NO	
Describe all fuel inserts	YES/NO	
Geometric Dimensions (Nominal and Tolerances)	YES/NO	
Schematic (axial/cross-section)	YES/NO	
Material compositions	YES/NO	
Describe non-standard fuel	YES/NO	
Geometric dimensions		
Describe non-fuel items in fuel cells	YES/NO	

Subject	Included	Notes / Explanation
Nominal and tolerance dimensions	YES/NO	
4.0 Spent Fuel Pool/Storage Rack Description		
New fuel vault & Storage rack description	YES/NO	
Nominal and tolerance dimensions	YES/NO	
Schematic (axial/cross-section)	YES/NO	
Material compositions	YES/NO	
Spent fuel pool, Storage rack description	YES/NO	
Nominal and tolerance dimensions	YES/NO	
Schematic (axial/cross-section)	YES/NO	
Material compositions	YES/NO	
Other Reactivity Control Devices (Inserts)	YES/NO	
Nominal and tolerance dimensions	YES/NO	
Schematic (axial/cross-section)	YES/NO	
Material compositions	YES/NO	
5.0 Overview of the Method of Analysis		
New fuel rack analysis description	YES/NO	
Storage geometries	YES/NO	
Bounding assembly design(s)	YES/NO	
Integral absorber credit	YES/NO	
Accident analysis	YES/NO	
Spent fuel storage rack analysis description	YES/NO	
Storage geometries	YES/NO	
Bounding assembly design(s)	YES/NO	
Soluble boron credit	YES/NO	
Boron dilution analysis	YES/NO	
Burnup credit	YES/NO	
Decay/Cooling time credit	YES/NO	
Integral absorber credit	YES/NO	
Other credit	YES/NO	
Fixed neutron absorbers	YES/NO	
Aging management program	YES/NO	
Accident analysis	YES/NO	
Temperature increase	YES/NO	
Assembly drop	YES/NO	
Single assembly misload	YES/NO	
Multiple misload	YES/NO	
Boron dilution	YES/NO	
Other	YES/NO	
Fuel out of rack analysis	YES/NO	
Handling	YES/NO	
Movement	YES/NO	
Inspection	YES/NO	

Subject	Included	Notes / Explanation
6.0 Computer Codes, Cross Sections and Validation Overview		
Code/Modules Used for Calculation of k_{eff}	YES/NO	
Cross section library	YES/NO	
Description of nuclides used	YES/NO	
Convergence checks	YES/NO	
Code/Module Used for Depletion Calculation	YES/NO	
Cross section library	YES/NO	
Description of nuclides used	YES/NO	
Convergence checks	YES/NO	
Validation of Code and Library	YES/NO	
Major Actinides and Structural Materials	YES/NO	
Minor Actinides and Fission Products	YES/NO	
Absorbers Credited	YES/NO	
7.0 Criticality Safety Analysis of the New Fuel Rack		
Rack model	YES/NO	
Boundary conditions	YES/NO	
Source distribution	YES/NO	
Geometry restrictions	YES/NO	
Limiting fuel design	YES/NO	
Fuel density	YES/NO	
Burnable Poisons	YES/NO	
Fuel dimensions	YES/NO	
Axial blankets	YES/NO	
Limiting rack model	YES/NO	
Storage vault dimensions and materials	YES/NO	
Temperature	YES/NO	
Multiple regions/configurations	YES/NO	
Flooded	YES/NO	
Low density moderator	YES/NO	
Eccentric fuel placement	YES/NO	
Tolerances	YES/NO	
Fuel geometry	YES/NO	
Fuel pin pitch	YES/NO	
Fuel pellet OD	YES/NO	
Fuel clad OD	YES/NO	
Fuel content	YES/NO	
Enrichment	YES/NO	
Density	YES/NO	
Integral absorber	YES/NO	
Rack geometry	YES/NO	
Rack pitch	YES/NO	

Subject	Included	Notes / Explanation
Cell wall thickness	YES/NO	
Storage vault dimensions/materials	YES/NO	
Code uncertainty	YES/NO	
Biases	YES/NO	
Temperature	YES/NO	
Code bias	YES/NO	
Moderator Conditions	YES/NO	
Fully flooded and optimum density moderator	YES/NO	
8.0 Depletion Analysis for Spent Fuel		
Depletion Model Considerations	YES/NO	
Time step verification	YES/NO	
Convergence verification	YES/NO	
Simplifications	YES/NO	
Non-uniform enrichments	YES/NO	
Post Depletion Nuclide Adjustment	YES/NO	
Cooling Time	YES/NO	
Depletion Parameters	YES/NO	
Burnable Absorbers	YES/NO	
Integral Absorbers	YES/NO	
Soluble Boron	YES/NO	
Fuel and Moderator Temperature	YES/NO	
Power	YES/NO	
Control rod insertion	YES/NO	
Atypical Cycle Operating History	YES/NO	
9.0 Criticality Safety Analysis of Spent Fuel Pool Storage Racks		
Rack model	YES/NO	
Boundary conditions	YES/NO	
Source distribution	YES/NO	
Geometry restrictions	YES/NO	
Design Basis Fuel Description	YES/NO	
Fuel density	YES/NO	
Burnable Poisons	YES/NO	
Fuel assembly inserts	YES/NO	
Fuel dimensions	YES/NO	
Axial blankets	YES/NO	
Configurations considered	YES/NO	
Borated	YES/NO	
Unborated	YES/NO	
Multiple rack designs	YES/NO	
Alternate storage geometry	YES/NO	

Subject	Included	Notes / Explanation
Reactivity Control Devices	YES/NO	
Fuel Assembly Inserts	YES/NO	
Storage Cell Inserts	YES/NO	
Storage Cell Blocking Devices	YES/NO	
Axial burnup shapes	YES/NO	
Uniform/Distributed	YES/NO	
Nodalization	YES/NO	
Blankets modeled	YES/NO	
Tolerances/Uncertainties	YES/NO	
Fuel geometry	YES/NO	
Fuel rod pin pitch	YES/NO	
Fuel pellet OD	YES/NO	
Cladding OD	YES/NO	
Axial fuel position	YES/NO	
Fuel content	YES/NO	
Enrichment	YES/NO	
Density	YES/NO	
Assembly insert dimensions and materials	YES/NO	
Rack geometry	YES/NO	
Flux-trap size (width)	YES/NO	
Rack cell pitch	YES/NO	
Rack wall thickness	YES/NO	
Neutron Absorber Dimensions	YES/NO	
Rack insert dimensions and materials		
Code validation uncertainty	YES/NO	
Criticality case uncertainty	YES/NO	
Depletion Uncertainty	YES/NO	
Burnup Uncertainty	YES/NO	
Biases	YES/NO	
Design Basis Fuel design	YES/NO	
Code bias	YES/NO	
Temperature	YES/NO	
Eccentric fuel placement	YES/NO	
Incore thimble depletion effect	YES/NO	
NRC administrative margin	YES/NO	
Modeling simplifications	YES/NO	
Identified and described	YES/NO	
10.0 Interface Analysis		
Interface configurations analyzed	YES/NO	
Between dissimilar racks	YES/NO	
Between storage configurations within a rack	YES/NO	
Interface restrictions	YES/NO	

Subject	Included	Notes / Explanation
11.0 Normal Conditions		
Fuel handling equipment	YES/NO	
Administrative controls	YES/NO	
Fuel inspection equipment or processes	YES/NO	
Fuel reconstitution	YES/NO	
12.0 Accident Analysis		
Boron dilution	YES/NO	
Normal conditions	YES/NO	
Accident conditions	YES/NO	
Single assembly misload	YES/NO	
Fuel assembly misplacement	YES/NO	
Neutron Absorber Insert Misload	YES/NO	
Multiple fuel misload	YES/NO	
Dropped assembly	YES/NO	
Temperature	YES/NO	
Seismic event/other natural phenomena	YES/NO	
13.0 Analysis Results and Conclusions		
Summary of results	YES/NO	
Burnup curve(s)	YES/NO	
Intermediate Decay time treatment	YES/NO	
New administrative controls	YES/NO	
Technical Specification markups	YES/NO	
14.0 References		
Appendix A: Computer Code Validation:		
Code validation methodology and bases	YES/NO	
New Fuel	YES/NO	
Depleted Fuel	YES/NO	
MOX	YES/NO	
HTC	YES/NO	
Convergence	YES/NO	
Trends	YES/NO	
Bias and uncertainty	YES/NO	
Range of applicability	YES/NO	
Analysis of Area of Applicability coverage	YES/NO	