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

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NOTICE

FOREWORD

This guidance describes acceptable methods 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 used fuel assemblies stored in spent fuel racks in a spent fuel pool.

Requirements for criticality controls for nuclear power plants are found in 10 CFR Part 50, Appendix A, GDC 62; which are met through 10 CFR 50.68, or 10 CFR 70.24. Guidance for performing criticality analyses in compliance with these regulations were originally developed in a 1998 Nuclear Regulatory Commission internal memorandum by L. Kopp, and supplemented by the Standard Review Plan, NUREG-0800, Sections 9.1.1 and 9.1.2. Additional guidance was issued in an Interim Staff Guidance (DSS-ISG-2010-01) in 2011. This industry guidance document is developed as a comprehensive guide that presents an acceptable approach to comply with the regulations, and upon NRC endorsement would supersede previous guidance documents.

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GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

1 INTRODUCTION

1.1 PURPOSE

This document provides guidance to the nuclear industry for performing criticality analyses for light-water nuclear reactor plant 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 (both new and 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, which will achieve durability through NRC concurrence, and at such time this guidance will supersede previous guidance on criticality analyses 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] was granted on a case-by-case basis for licensees demonstrating subcriticality through analysis. Compliance with either regulation satisfies 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, following promulgation of 10 CFR 50.68 [1], was also issued in 1998 through an NRC internal memorandum from L. Kopp to T. Collins, often referred to as the "Kopp Memorandum" [21]. 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. The guidance in the Kopp Memorandum provided regulatory clarity and stability over the next few 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 would 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," [22] 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 [22] 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, at which time NEI 12-16 would supersede previous guidance documents.

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 61, "Fuel Storage and Handling and Radioactivity Control." [5]

- 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(a)(8), "Contents of applications; technical information in final safety analysis report." [4]

It is noted that in addition to the applicable regulations, the NRC has developed associated 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. [11]

- NUREG-0800, Standard Review Plan, Section 9.1.2, "New and Spent Fuel Storage," Revision 3. [12]

1.4 DOUBLE CONTINGENCY PRINCIPLE

The double contingency principle, as described in ANSI/ANS 8.1, Section 4.2.2 [9], states that "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 and

abnormal occurrences. This will ensure that no single occurrence can lead to a criticality. The double contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or abnormal occurrences. When the licensee applies the double contingency principle, the licensee should provide an explanation why the conditions chosen/used are independent from one another (i.e. do not result from a common initiator) and are unlikely (i.e. low probability). For example, for PWRs, the loss of soluble boron below the Technical Specification limit is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the Technical Specification soluble boron may be assumed in evaluating other accident conditions.

1.5 USE OF PRECEDENCE

The use of precedence (i.e., adopting methods or conclusions previously approved in another application, but not documented in a generic regulatory document) is a well-established principle by the NRC in the process of reviewing applications. The use of precedence provides regulatory stability and efficiency. In order for a licensee to use precedence 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. Precedence should be used within the confines of the limitations of the context established when previously approved. Precedence may be used in whole or in part and should be technically justified. Any similarities and/or differences/deviations should be technically supported and demonstrated as appropriate, in order to ensure a high degree of fidelity in keeping with the context to which applicability is being sought. Consideration should also be given to any NRC guidance that has been documented 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

In some instances, use of engineering judgment in criticality analyses can result in resource efficiencies, and may be needed due to lack of conclusive information. In either situation, the use of engineering judgment should be applied in an appropriate manner. 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 should note that assumptions used in the criticality analysis should be explicitly identified and clearly stated. The licensee should also bear in mind that 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 (again specifically referring to the criticality analyst) 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, the licensee (criticality analyst) should clearly identify their assumptions. The licensee, to the extent practicable, should provide a basis supporting assumptions defined in the application. Where no basis exists for the use of engineering judgment, the licensee should modify their approach such that the criticality analyses can be performed without the use of that engineering judgment.

Use of engineering judgment and assumptions may also consist of applying 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 ANALYTICAL TECHNIQUES TO CALCULATE K_{EFF}

2.1 ACCEPTANCE LIMITS

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 accident conditions with associated limits, which should be incorporated into plant technical specifications:

a) With the new fuel storage racks loaded with fresh fuel of the maximum permissible reactivity and flooded with pure un-borated water, the maximum k_{eff} must not exceed 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

b) With the new fuel storage racks loaded with fresh fuel of the maximum permissible reactivity and filled with low-density hydrogenous fluid corresponding to optimum moderation, the maximum k_{eff} must not exceed 0.98, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other independent accident conditions need not be considered.

Spent (Used) Fuel Storage

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, which should be incorporated into the plant technical specifications:

a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with unborated water, the maximum k_{eff} must not exceed 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level, for both normal and accident conditions.

2) For pools where credit for soluble boron is taken (typically PWR pools), the criticality safety analyses must meet two independent limits, which should be incorporated into the plant technical specifications:

a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with unborated water, the maximum k_{eff} must remain below 1.0, including mechanical and calculation uncertainties, with a 95-percent probability at a 95-percent confidence level.

b. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with borated water, the maximum k_{eff} must not exceed 0.95, including mechanical and calculation uncertainties, with a 95-percent probability at a 95-percent confidence level.

2.2 K_{EFF} EQUATION

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 following formula:

$$k_{max} = k_{eff} + \sum_{i=0}^{m} Bias_i + \sqrt{\sum_{j=0}^{n} Uncertainty_j^2}$$

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 (e.g., validation biases and uncertainties are in Section 3, and mechanical uncertainties are in Section 4).

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 maximum k_{eff} must be less than the regulatory limit.

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 all 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, criticality codes and depletion codes.

The criticality codes are used to determine the eigenvalue (k_{eff}) of the analyzed system. The isotopic concentrations are determined from manufacturing data and depletion analysis. Although many criticality codes utilize Monte Carlo methods, there are some criticality codes that utilize deterministic transport methods. 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.

The depletion codes are used to calculate the nuclide density changes that occur in the nuclear reactor core when operated at power. In addition, decay changes in nuclide concentrations due to non-power cooling times are also captured in depletion calculations. In general, depletion codes utilize transport methods.

The codes to calculate depletion and criticality rely upon use of cross-section libraries. Cross-section libraries should be widely accepted and peer reviewed. Cross-section libraries that have previously been found acceptable for use include multi-group and continuous energy ENDF series.

The licensee should 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.

If the code is not capable of directly modeling the benchmark experiments, then an intermediary code may be used that is benchmarked to the experiments, and to which the primary code is benchmarked. 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 intermediary code should be validated by benchmarking to experiments that are similar to the neutronics and geometry of the criticality safety analysis. The primary code should then be validated by benchmarking to the intermediary code over a range of parameters (neutronic and geometric) that bound the range of parameters for the criticality safety analysis. The bias and uncertainty of the primary code should include the biases and uncertainties from both the primary code to intermediary code validation, and the intermediary code to benchmarks validation.

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 licensee (i.e, the analyst or organization performing the analysis) by comparison with experiments and accounting for the parameters not accounted for by the existing experiments.

This qualifies both the ability of the licensee (analyst/organization) and the computer environment. The critical experiments used for benchmarking should include configurations having neutronic and geometric characteristics comparable to those of the proposed storage facility.

The computer code validation for new fuel storage consists of validation for fresh unburned fuel, as described in Section 3.2.1. The computer code validation for used fuel storage consists of validation for fresh unburned fuel, described in Section 3.2.1, and additional validation of used depleted fuel, described in Section 3.2.2. If the validation for used depleted fuel necessitates slight modifications to the validation of the fresh unburned fuel, these modifications are explained in the applicable areas in Section 3.2.2.

3.2.1 Fresh Fuel Criticality Validation

The 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 [13].

3.2.1.1 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
    - Experiments should cover material for the rack structure (e.g., stainless steel), material for the cladding (e.g., zirconium), fissile isotopes in the applicable enrichment range (e.g., U-235 for low enriched UO_2, Pu-239 for MOX), water, 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
    - 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.

3.2.1.2 Selection of Critical Experiments

The features listed above are covered with available critical experiments, for example the OECD/NEA *International Handbook of Evaluated Criticality Safety Benchmarks Experiments* [23] and the HTC critical experiments [14] are considered an appropriate reference 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 set of experiments selected should ensure a statistically appropriate validation. Care should be taken in selecting critical experiments so that trends can be identified and addressed.

3.2.1.3 Modeling the Experiments

Section 2.3 of NUREG/CR-6698 [13] 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).

3.2.1.4 Analysis of the Critical Experiment Data

NUREG/CR-6698 [13] 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 [13] 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 [23] are sufficient for this purpose so the statistical approach defined in NUREG/CR-6698 [13] 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. In these cases, only spectral or enrichment trends need to be considered.

The equation in Section 2.2 can be used to calculate the maximum k_{eff}. Alternatively, the method in NUREG/CR-6698 [13] 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 2.2). The bias and uncertainty determined from the critical experiments may be applied as a function of the trending parameters or as conservative values that cover the desired range(s).

3.2.1.5 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.

Limitations and Conditions

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 [13] provides further guidance for extending trends and accounting for increasing uncertainty if there are insufficient critical experiments.

3.2.2 Used Fuel Criticality Validation

Additional validation is required for used fuel since it depends on depletion analysis and the reactivity worth of isotopes not found in the fresh fuel critical experiments. The computer code validation for used fuel storage consists of validation for fresh unburned fuel, described in Section 3.2.1, and additional validation of used depleted fuel, described in Section 3.2.2. If the validation of used depleted fuel necessitates slight modifications to the validation of the fresh unburned fuel, these modifications are explained in the applicable areas in Section 3.2.2.

For used fuel storage, fuel rack criticality analyses utilize lattice depletion calculations to generate isotopic concentrations which are transferred to criticality analysis tools to assess the reactivity of the storage system. The major difference between these "fresh fuel critical experiment" benchmarks and true storage configurations is the isotopic composition of used fuel. Therefore, it is the purpose of used fuel validation to assess the accuracy with which lattice depletion codes can assess the isotopic and corresponding reactivity change of fuel lattices from the initial fresh fuel condition to the used fuel condition. Previously, there were not any methods for calculating a validation bias and uncertainty that had wide-spread consensus, and the historical approach was to use engineering judgment. Depletion validation that directly calculates a bias and uncertainty is preferred but an approach that uses conservative bias and uncertainty values, which are not directly calculated, is also acceptable.

The following are three acceptable methods for calculating a validation bias and uncertainty, and are described in detail in the following sub-sections.

1. Use measured flux data from power reactors.
2. Use measured critical data from power reactors.
3. Use chemical assays and worth experiments.

Alternatives approaches may be proposed by licensees provided they are adequately justified.

3.2.2.1 Validation Using Measured Flux Data from Power Reactors

PWR depletion benchmarks were developed by EPRI [24] 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 [25]. 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. The bias and uncertainty associated with fresh fuel are also required to be included in the validation of the criticality safety evaluation. The EPRI report describes in detail how to apply the benchmarks in the criticality safety analysis.[24]

3.2.2.2 Validation Using Measured Critical Data from 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. The following are two approaches to use this bias and bias uncertainty.

Method A (Assigning the startup bias and uncertainty to isotopic content only):

In Method A, the criticality code validation, isotopic composition, and cross section uncertainties are assessed in three steps:

1) The criticality analysis code and fresh fuel isotopic cross sections are validated using fresh fuel critical experiments as described in Section 3.2.1. It should be noted that there is a set of experiments that include plutonium in concentrations consistent with used fuel known as the HTC critical experiments [14]. The inclusion of the HTC critical experiments in the fresh fuel validation can cover the major actinide worth uncertainty.
2) The measured startup core critical bias (Δk_{SUb}) and bias uncertainty (Δk_{SUu}) is applied to cover the isotopic composition uncertainty. This is appropriate since isotopic content for the criticality analysis comes directly from the same lattice physics code used for the reactor startup analysis, and the corresponding startup bias and bias uncertainty are a function of the lattice physics code's capability to calculate nodal cross sections (and isotopics) for the core simulator.
3) Actinides' and fission products' cross sections which are not explicitly represented in the critical experiments are covered by adding an uncertainty that is proportional to the reactivity worth of the isotopes not explicitly validated. One approach to do this has been developed in NUREG/CR-7109 [15].

The final validation bias in Method A is the sum of the bias from the startup data and the bias from the fresh fuel critical experiments. The uncertainty is the statistical combination of the uncertainties from the startup data, the fresh fuel critical experiments, and proportional reactivity worth assessment. These uncertainties can also be statistically combined with other independent uncertainties such as rack and fuel manufacturing tolerances.

Method B (Reactivity use of the startup bias and uncertainty):

In Method B, it is assumed that the measured core critical bias (Δk_{SUb}) and bias uncertainty (Δk_{SUu}) fully validates the lattice physics code results and therefore covers both the uncertainty in isotopic content and worth. To implement, a reactivity equivalence must be established between the lattice physics code used for the depletion analysis for the startups and the Monte Carlo code used in the criticality analysis.

In Method B the following steps are required:

1. Using the same lattice physics code as used in the core startup analyses a calculation of the k-inf at the peak reactivity condition (enrichment, burnup, and gadolinium) is performed.
2. This same peak reactivity condition is modeled in the criticality Monte Carlo code to establish a bias (Δk_{MC}) between the Monte Carlo code and the lattice physics code. Notice that the isotopic content comes from the lattice physic code depletion.
3. Since the power reactor startups do not have stainless steel and possibly other rack features, fresh fuel critical experiments have to be run to seek any bias and uncertainty from these features. Fresh fuel critical experiments also validate the criticality analysis tool's solution method, as described in Section 3.2.1 of this guidance. (Note that this step in practice is the same as step 1 of Method A but with slightly different justification.)

This analysis results in a bias (Δk_{cb}) and uncertainty (Δk_{cu}). Since the power reactor startup bias and uncertainty contain uranium and plutonium the bias due to these isotopes are counted twice. To assure no cancelation of errors negative, biases are ignored. Since the power reactor startups contain fuel with well understood fission product content, no additional bias and uncertainty for fission products is needed.

The final validation bias in Method B is the sum of the bias from the power reactor startup data (Δk_{SUb}), the bias from the benchmarking of the Monte Carlo code to the lattice physics code (Δk_{MC}), and the bias from the fresh fuel critical experiments (Δk_{cb}). The uncertainty is the statistical combination of the uncertainties from the startup data (Δk_{SUu}) and the fresh fuel critical experiments (Δk_{cu}). These uncertainties can also be statistically combined with other independent uncertainties such as rack and fuel manufacturing tolerances.

3.2.2.3 Validation Using Chemical Assays and Worth Experiments

Depletion validation using chemical assays and worth experiments, performed for the NRC by ORNL, are documented in NUREG/CR-7108 [16] and NUREG/CR-7109 [15]. The NUREG/CRs include biases and uncertainties that can be used in the validation of PWR and BWR criticality analyses if the system and method match those used to produce the bias and uncertainty. It should be noted that this method may be significantly conservative due to the large experimental uncertainties in performing chemical assays. In this method the experimental uncertainty in measuring the isotopic content increases the uncertainty in prediction of the isotopic content. Studies in the past have shown that the uncertainty changes very little with method changes, which would be expected if the uncertainty is dominated by the uncertainty in the chemical assays rather than the uncertainty from the calculational method.

The NUREG/CRs were prepared mainly for analysis of transport casks; however, some modifications can be added in order to develop an approach more appropriate for spent fuel pool applications. In NUREG/CR-7108 a direct difference method was presented [16]. The Monte Carlo approach used large burnup bins in order to get enough data to establish the distribution of data around the mean for each isotope. Although this appropriately accounts for the variation in number of isotopes included in the chemical assay samples, it loses most of the burnup dependence of the data. The direct difference approach does not lose the burnup dependence of the data and handles the missing isotopic data by using “surrogate data” for nuclides without measurements. If validation through chemical assays is selected, it is recommended that the 100 chemical assays selected for NUREG/CR-7108 [16] be analyzed and then the direct difference approach be applied to determine a bias and uncertainty as a function of burnup.

Since the chemical assay approach results in a conservative estimate of the bias and uncertainty it is recommended that the bias and uncertainty from the chemical assays be applied for all isotopes. It has been shown that the isotopes in excess of the 28 major isotopes selected, in NUREG/CR-7108, have a relatively small worth so it would be appropriate to use the bias for all isotopes. However, another method, which encompasses all isotopes such as described in Section 3.2.2.1 or Section 3.2.2.2, may be employed to justify the use of the chemical assay based bias and uncertainties for all isotopes.

In this approach the reactivity worth of actinides is shown by MOX critical experiments (including the HTC critical experiments [14]). Since the burnup of used fuel can vary from a few

GWd/T to 60 GWd/T the bias and uncertainty from the critical experiments should come from the most limiting of the fresh UO_2 and MOX critical experiment sets.

NUREG/CR-7109 [15] recommends a bias of 1.5 % (one sigma) of the reactivity worth of the isotopes not included in the critical experiments to cover the bias and uncertainty. The isotopes used in addition to the 28 isotopes are expected to behave similarly so the use of 1.5% of the reactivity worth can be extended to cover these isotopes [15].

4 RACK AND FRESH FUEL MODELING

4.1 FRESH FUEL ASSEMBLIES

The criticality analysis typically relies on a nominal representation of the fuel assembly design (i.e., nominal dimensions, materials, and isotopic concentrations), and applies manufacturing tolerances as uncertainties. If independent uncertainty items are evaluated separately, 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}.

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. Typically, an 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.

The following fresh fuel assembly tolerances should be considered for inclusion as uncertainties in the criticality analysis, unless they can be shown to be insignificant. The parameters are listed in descending order of estimated level of significance to k_{eff} uncertainty.

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

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.

4.2 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., fog or foam) should 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 large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the maximum k_{eff} is no greater than the regulatory limits. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which sub criticality depends should be explicitly identified and evaluated.

The following vault tolerances should, at a minimum, be considered when evaluating the uncertainties. The parameters are listed in descending order of estimated level of significance to k_{eff} uncertainty.

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).

4.3 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). 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, at a minimum, be considered when evaluating the uncertainties due to tolerances. The parameters are listed in descending order of estimated level of significance to k_{eff} uncertainty.

a) Flux Trap Size
b) Cell/Storage Location Pitch
c) Eccentric Fuel Positioning
d) Storage Cell Wall Thickness

4.4 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 includes modeling of the boron content (B-10 areal density) and dimensions. Of these modeling parameters, B-10 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 [26].

Typically, neutron absorbers are not used in dry new fuel vaults, where the geometry of the vault is designed to prevent criticality.

4.4.1 Dimensions

The modeling of SFP rack dimensions is described in Section 4.3. Fixed neutron absorbers are typically part of the original rack design. Rack manufacturer drawings will provide detailed dimensions for the neutron absorber including how the 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 these parameters may be used in lieu of tolerance analyses.

4.4.2 Boron Content

The boron content of the neutron absorber (B-10 areal density) is a critical 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 are guaranteed to have higher boron concentrations, since the manufacturer must take into account manufacturing tolerances. For example, the manufacturer will target a nominal boron concentration that 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 are guaranteed to have boron concentrations at or above this minimum concentration, and these are documented in QA records. In some cases, these records have been collected by the manufacturer and provided with delivery on a batch basis.

An alternative 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 with the criticality analysis prior to storing fuel in the racks.

4.4.3 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 should clearly identify the absorber assumptions and inputs. If material changes are anticipated over their intended service life, these anticipated changes should be appropriately bounded by the criticality analysis. In extreme cases, if degradation, loss of ^{10}B areal density is anticipated, then appropriate margin to account for the degradation should be accounted for 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 surveillance program (see Section 9.5). If any un-anticipated aging or change is identified through the surveillance 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 CONFIGURATION MODELING

5.1 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 assembles are in the approved storage locations. It is just as important to consider normal activities and operations in the spent fuel pool. Examples of these normal activities are movement of fuel in and around the spent fuel pool, fuel inspection and reconstitution. 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. Normal conditions also include the normal range of water temperature for pool storage. It is noted that different plants will have different normal conditions.

5.2 INTERFACES

In the event the spent fuel pool contains more than a single storage configuration, the criticality analysis should consider the interface between storage configurations, unless shown to be neutronically decoupled. An interface occurs every time two or more different storage configurations meet. In some cases, interfaces may result in a higher k_{eff} than the k_{eff}'s of the configurations evaluated individually. If necessary, interface restrictions may be applied. Evaluation of the interfaces should consider biases and uncertainties.

An interface can also occur between old and new racks. If the separation distance between the new and old racks is more than 12 in. at the interface, then there is no need to evaluate.

5.3 ABNORMAL AND ACCIDENT CONDITIONS

The licensee should 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. The separate boron dilution accident is discussed in Section 6.3.

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.

5.3.1 Temperature

Abnormal pool water temperatures (above or below those normally expected) should be evaluated to consider the effect on criticality.

5.3.2 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 typically 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. In many cases this separation prevents neutronic coupling but even if there is some coupling the other accident conditions are usually more limiting.

Also, a mislocated fresh fuel assembly outside and adjacent to the storage racks (inside the pool wall) should also be evaluated if applicable. An example of when this scenario is not applicable is the case where there is not enough room to physically fit a fuel assembly in between the racks and/or the pool wall.

5.3.3 Assembly Misload

Misloading of a single fuel assembly into an unapproved location, such as loading a fresh fuel assembly with the highest enrichment into a storage location intended for a used fuel assembly, should be evaluated as part of postulated accident scenarios. However, if the storage conditions, and analysis of these conditions, are a uniform pattern of the maximum reactivity fuel conditions (e.g. design, enrichment, burnup), then a single misload of a higher reactivity assembly would be precluded and need not be evaluated.

The effects of multiple misloads of fuel assemblies should be evaluated if a single initiation event resulting in multiple misloads is considered credible. Whereas a single event resulting in a single misload is typically a result of an error in the fuel handling selection or relocation of an assembly (i.e., picking up an assembly other than the intended assembly), a single event resulting in multiple misloads is typically the result of a planning or process error. Therefore, whether a multiple misload resulting from a single event is credible depends upon the administrative

controls and processes the licensee establishes for assuring compliance with the loading patterns. Implementing a robust administrative control program for verifying used fuel assembly configurations and addressing potential non-compliant loading conditions (see Section 9.2), may preclude common cause failure of misloads.

Some licensees may be able to demonstrate that a multiple misload from a single event is not credible, while others may determine it is credible and should analyze the consequences of a multiple misload. Again, the administrative controls and processes the licensee establishes for assuring compliance with the loading patterns will influence the potential consequences of a 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. In this example, the misloaded fuel assemblies could represent the minimum burnup for once burned fuel with the highest enrichment, since the process check would ensure that it is not credible to misload fresh fuel assemblies.

6 SOLUBLE BORON CREDIT

6.1 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.

6.2 ACCIDENT CONDITIONS

For conditions with soluble boron, the accident conditions in Section 5.3 should be evaluated at the minimum allowable normal soluble boron concentration. 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. A misload event with a boron dilution event is precluded by the double contingency principle, even though there is a low probability that the misload could persist for some time before it is identified, if it can be demonstrated not to occur from the same initiating event. 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, the maximum calculated k_{eff}, including all biases and uncertainties, must meet the regulatory k_{eff} limit of 0.95 or less. Accidents that result in a dilution of the soluble boron are addressed in Section 6.3.

6.3 BORON DILUTION

In the event the licensee is crediting soluble boron in the criticality safety calculation, a boron dilution accident should be considered. The boron dilution analysis should initiate 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 should confirm the time needed for dilution 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 it needed for actions to be taken to prevent further dilution.

A graded approach to the boron dilution analysis may be taken depending on the amount of soluble boron being credited versus the amount required to be in the spent fuel pool, or the amount of time necessary for a boron dilution to become less than the minimum required and the time necessary to prevent further dilution. For example, if 2000ppm of soluble boron is required by the technical specification, then a licensee that takes credit for nearly this entire amount (e.g. 1800ppm) in the criticality analysis, may need to provide additional justification for their assumptions than a licensee that only takes credit for a small portion of soluble boron (e.g. 200ppm). Similarly, an analysis that calculates that the credible soluble boron dilution event would not reduce levels below those required in less than 24 hours, and action would be taken to prevent further dilution in less than 8 hours, may not need to provide additional justification for the assumptions in the boron dilution analysis.

7 REACTIVITY EFFECTS OF DEPLETION

7.1 REACTIVITY EFFECTS OF DEPLETION FOR PWRS

The most important parameters that could potentially result in an increase in the reactivity of burned fuel in depletion analyses for PWRs, listed in descending order of importance, are:

a) Relative power during depletion (which impacts the moderator and fuel temperatures during depletion);
b) Soluble boron during depletion;
c) Presence of burnable absorbers;
d) Rodded operation; and
e) Axial burnup shapes.

NUREG/CR-6665 [17] provides guidance in selecting operating parameters for depletion analysis.

7.1.1 Depletion Analysis

Relative Power during Depletion

The relative power of a fuel assembly during depletion will directly impact the moderator and fuel temperature. Higher relative power will result in higher moderator and fuel temperatures. Higher moderator and fuel temperatures typically result in increased reactivity in the storage rack. The moderator and fuel temperature used during the depletion analysis should therefore be conservative and appropriately justified. A high relative power also results in a high specific power. While higher specific power will lead to a higher Sm-149 content after the decay of Pm-149, which lowers reactivity, this effect is much smaller than the impact of the moderator and

fuel temperature. Therefore, the highest relative power should be selected to maximize the net reactivity of all the effects.

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 [27]. A conservatively high burnup averaged soluble boron concentration should therefore be confirmed and used in the depletion calculations.

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 integral to the fuel (Gd, Erbium, etc), as a coating on the fuel pellet (ZrB_2 IFBA) or as inserts in the guide tubes (e.g. WABA, BPRA, Pyrex). In all cases the effect of the presence of these absorbers on the reactivity of the fuel assembly should be appropriately considered and accounted for in the depletion analysis. The maximum neutron absorber loading of the burnable absorbers for the maximum burnup should be modeled. Note that studies have shown that burnable absorbers that are integral to the fuel pellet, e.g., Gadolinium, Erbium, can be conservatively neglected [18]. If Gadolinium or Erbium is to be neglected care should be taken to assure the analysis is consistent with the NUREG/CR. ZrB_2 IFBAs should be considered explicitly.

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 of the burnable absorbers present in the fuel assembly.

It should also be mentioned that neutron absorbers are modeled with nominal dimensions in the criticality 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]. Note that most PWRs operate with all rods out. However, use of this assumption should be justified. Multiple loading criteria may be developed if multiple assumptions are used for rodded operation.

7.1.2 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 and shape are affected by the operating conditions. The burnup distribution near the top of the fuel assembly usually controls 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. Site-specific burnup shapes from the fuel cycle designs can be used as well as generic shapes, such as those in NUREG/CR-6801 [20]. In addition, the analysis should also address the usage of a distributed axial burnup profile versus a uniform profile, as a uniform profile may be conservative at low burnups.

7.2 PEAK REACTIVITY ANALYSIS FOR BWRS

It is standard practice that BWR spent fuel pool criticality analysis design basis calculations, 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 given burnup value, usually under 25 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 peak reactivity is determined by performing criticality calculations using isotopic compositions from separate depletion calculations performed over a burnup range to determine the burnup at which the peak reactivity occurs. A licensee should perform calculations in a manner that accounts for both the radial and axial pin locations. The peak reactivity method inherently bounds all axial effects by modeling the peak axial reactivity across all exposures at all axial nodes.

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

- **Reactor operating parameters**:
  - Void fraction – Higher void fractions typically increases peak reactivity, however, this is dependent upon the other reactor operating parameters and the full range of void fractions should be considered in conjunction with the other reactor parameters.
  - Control blade operation – Increased control blade operation typically increases peak reactivity, however, this is dependent upon the other reactor operating parameters and it should be considered in conjunction with the other reactor operating 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. Note that higher moderator temperatures typically result in an increase in peak reactivity in the storage racks.
  - 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.
- **Non-reactor operating parameters**:
  - Lattice specific parameters. Lattice specific parameters should each be evaluated during depletion and in the storage rack for their impact on peak reactivity. These parameters should at a minimum include:
    - Number, location and concentration of integral burnable absorber fuel rods

        - Number and location of partial length rods
    - Cooling time – Time from end of irradiation to the point of maximum reactivity resulting from isotopic changes due to fission product decays (typically 5 days).
    - SFP rack tolerances and uncertainties
    - BWR fuel lattice tolerances and uncertainties
    - Other tolerance and uncertainty calculations (e.g., fuel assembly specific parameters, methodology specific items)

A licensee should consider the following when preparing the depletion analysis for a submittal of a license application:

- All BWR criticality calculations should ensure a conservative reactivity is analyzed in the storage configuration with consideration given to possible cooling and discharge times.
- 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 and all calculations that are used to determine the maximum in-rack k_{eff}, including non-reactor parameter studies. Due to the large variation of BWR fuel designs and lattices within designs, the bounding reactor operating parameters may or may not be applicable to another design or lattices and therefore further evaluations may or may not be needed to show which parameters are bounding for other fuel designs or lattices within a design.
- The non-reactor operating parameter studies may demonstrate a peak reactivity burnup and a peak reactivity magnitude that varies from the design basis model and should be accounted for in the analysis by appropriate inclusion of the magnitude of the reactivity difference due to the change in peak reactivity.

8 OTHER CREDITS

Credit may be taken for the following reductions in reactivity, and should be adequately described and justified:

1. **Decay Time** – change of isotopic content due to radioactive decay

2. **Fresh Integral Burnable Absorbers** – in fresh fuel

3. **Used Removable Burnable Absorbers** – displacement of moderator

4. **Control Rods** – full-length rod control cluster assemblies (RCCAs)

5. **Absorber Inserts** – absorbers that are inserted into the assembly. Note that absorbers inserted between assemblies and the racks are addressed in Section 4.4.

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 technical specifications, plant procedures and programs that control storage configurations, burnup/enrichment loading curves, and ensure that the storage of fuel is bounded by the criticality analyses.

9.2 ADMINISTRATIVE CONTROLS

A licensee establishes administrative 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 are typically in the form of plant procedures and processes. These procedures implement the requirements for tracking the location of fuel assemblies in accordance with Special Nuclear Material (SNM) regulations and 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 software QA plan. The software QA program covers the use of codes for criticality analyses and software used to plan and implement fuel movements.

Administrative controls should be developed based upon the complexity of storage patterns in order to provide reasonable assurance of adequate public health and safety. The administrative controls may also affect the assumed accident conditions (see Section 5.3) For example, if storage patterns are relatively straight-forward and the administrative controls preclude a credible multiple misload event resulting from a single initiating event, then the multiple misload event would not need to be evaluated as an accident condition. The following are typical administrative controls used by licensees. Additional administrative controls should be considered for more complex storage patterns.

- Pool Assembly Storage Planning
  - Fuel Characterization
    - Fuel reactivity category determination, e.g.,
      - 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)
    - Soluble boron surveillance
    - Neutron Absorber Panel material behavior surveillance 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 approved of all database updates
    - Administrative controls ensure integrity of database prior to utilizing the data

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 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 storage rack/vault.

If an initial assessment determines that the new fuel type represents a potential change to existing criticality safety design basis/analysis of record for the storage rack/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 can be 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, shutdown decay 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. 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 analyses may be able to verify that the reactor operated within Technical Specification limits, while others may need to verify more detailed reactor parameters or assembly specific parameters, such as:

- Axial burnup shape
- Moderator temperature
- Fuel temperature

- Soluble boron
- 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 in off-normal configurations
- Changes to plant configuration that result in higher-than-expected reactor coolant temperatures

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).

9.5 NEUTRON ABSORBER SURVEILLANCE PROGRAMS[1]

Neutron absorbers serve as an important material to control reactivity in most spent fuel pool storage racks. As neutron absorbers significantly reduce reactivity, it is important to ensure that they continue to provide their criticality control function for the duration that they are in service and relied upon in the criticality analyses. Neutron absorber surveillance programs should be developed with the purpose of ensuring that the neutron absorbers continue to provide the criticality control relied upon in criticality analyses. To accomplish this, the surveillance program must be capable of identifying whether unanticipated changes are occurring, and if anticipated changes are occurring that the anticipated characteristics of change can be verified.

Coupon surveillance, in-situ measurement or a combination are acceptable approaches to a neutron absorber surveillance program; however, alternative approaches are also acceptable if adequately justified. A surveillance program should also consist of identifying material testing, R&D and operating experience at other plants, and evaluation on the relevance of outside data on the in-service material. Acceptance criteria should be developed as the basis for the comparison

[1] While these guidelines for neutron absorber surveillance programs are intended for initial license applications, and license amendment requests that install new neutron absorber materials, they may be useful for licensee's consideration in license renewal applications under 10 CFR Part 54.

of surveillance results in order to determine whether material performance is acceptable or actions are necessary to address performance issues.

9.5.1 Coupon Surveillance

Use of coupons is the preferred method for a neutron absorber surveillance program. Coupon surveillance programs should meet the following criteria:

- The number of coupons should be sufficient to provide sampling at an appropriate interval for the intended life of the neutron absorber. The intended life of the neutron absorber should be based upon the amount of time the neutron absorber will be relied upon to provide criticality control. This is typically the life of the plant plus some additional time to permit off-loading the spent fuel pool during decommissioning.

- Sampling intervals should be based upon the expected rate of material changes, which may be influenced by the qualification testing of the material. For new materials that do not have a lot of operating experience in conditions similar to the pool environment (i.e. their ability to perform is not well known), the initial interval should not exceed 5 years, with subsequent intervals up to 10 years. For materials that have been used for several years in conditions similar to the pool environment (i.e. their ability to perform is well known), and for which stability in the material condition has been documented, initial and subsequent intervals up to 10 years is acceptable.

- Coupon testing can be categorized as basic or full testing. The coupon testing is used to identify whether unanticipated changes are occurring, and if they are, the condition of the neutron absorber material. The extent to which each of these is utilized should be determined based upon the operating history of the material, as follows:

  a) Basic testing consists of visual observations, dimensional measurements, and weight. These parameters focus on identification of whether changes are occurring in the materials. Basic testing is appropriate when testing and operating experience of the material indicates that there are no degradation mechanisms that would result in loss of ^{10}B areal density.

  b) Full testing may consist of a combination of density measurements, ^{10}B areal density measurements, microscopic analysis, and characterization of changes, in addition to the basic testing parameters. These parameters focus on quantifying changes if they are occurring in the materials. Full testing should be performed for the first coupon test, but may not be necessary for subsequent test periods unless a loss of ^{10}B areal density is anticipated. Basic testing may be used in combination with full testing for materials that have degradation resulting in loss of ^{10}B areal density to extend the interval of full testing, if appropriately justified.

- Coupons should be located such that their exposure to parameters controlling change mechanisms (e.g., gamma fluence, temperature) is similar to the in-service neutron absorbers.

- Results are acceptable to confirm the continued performance of neutron absorber materials if either:

a) For materials that are not anticipated to have a loss of ^{10}B areal density; the ^{10}B areal density of the surveillance coupon is the same as its original ^{10}B areal density.

b) For materials that are anticipated to have a loss of ^{10}B areal density; the loss of ^{10}B (difference between ^{10}B areal density of the surveillance coupon and its original ^{10}B areal density) is less than the loss of ^{10}B areal density used in the criticality analysis.

9.5.2 In-situ Measurement

In-situ measurement is another acceptable method for confirming ^{10}B areal density of neutron absorber material. In-situ measurement is used to identify whether unanticipated changes are occurring, and if they are, the condition of the neutron absorber material. There are two potential uses for in-situ measurements:

1. Supplement coupon surveillance to extend the coupon testing interval or permit greater reliance on basic testing.
2. In lieu of coupon testing if coupons do not exist.

Both uses of the in-situ measurement should meet the following criteria:

- In-situ measurement campaigns should be performed on an adequate number of panels and at an acceptable interval.
- Number of panels tested should be an appropriate statistical sample.
- Sampling interval is based upon the expected rate of material change, which may be influenced based upon the qualification testing of the material. For new materials that do not have a lot of operating experience in conditions similar to the pool environment (i.e. their ability to perform is not well known), the initial interval should not exceed 5 years, with subsequent intervals up to 10 years. For materials that have been used for several years in conditions similar to the pool environment (i.e. their ability to perform is well known), and for which stability in the material condition has been documented, initial and subsequent intervals up to 10 years is acceptable.
- Note that sampling interval can be longer if used in conjunction with coupons.
- Sources of measurement uncertainty should be identified and the degree of uncertainty quantified.

Additional criteria for in-situ measurements depend upon the performance of the neutron absorber material, specifically whether material changes result in a degradation of the ^{10}B areal density.

A. For materials where potential change mechanisms do not result in a loss of ^{10}B areal density, in-situ measurements are used to confirm their presence. Results confirm the continued performance of neutron absorber materials if the measured ^{10}B areal density plus the measurement uncertainty is greater than the ^{10}B areal density assumed in the criticality analysis.

B. For materials where degradation mechanisms may result in a loss of ^{10}B areal density, in-situ measurements are used to determine the amount of ^{10}B areal density remaining. Results confirm that potential loss of ^{10}B has not resulted in the loss of the neutron absorber material's ability to perform its criticality control function if the measured ^{10}B areal density minus the measurement uncertainty is greater than the ^{10}B areal density assumed in the criticality analysis.

9.5.3 Evaluating Neutron Absorber Surveillance Results

Results from neutron absorber surveillance fall within the broad categories of 1) confirmation that no material changes are occurring, 2) confirmation that anticipated changes are occurring, and 3) identification that unanticipated changes are occurring. Processes should be established to evaluate results of the surveillances with the criticality analysis input. If no changes, or if anticipated changes are occurring, then the material condition continues to be adequately represented in the criticality analysis.

If unanticipated changes are identified (either new mechanisms or anticipated mechanisms at rates or levels beyond those anticipated), then additional actions may be necessary. In addition to relevant regulatory and licensing processes (e.g. operability determination, reporting requirements, the 10 CFR 50.59 [8] process), the following technical assessments may be necessary.

- Determine if unanticipated changes could result in a loss of ^{10}B areal density. This is considered the only major impact to criticality control (See Section 4.4), since ^{10}B areal density has a much larger impact than dimensions. Evaluation of the effects of ^{10}B areal density on the criticality analysis should be performed and addressed through appropriate licensee processes.

- Determine if unanticipated changes not resulting in loss of ^{10}B areal density have an impact on the criticality analyses. Dimensional or non-neutron absorbing material changes (e.g. formation of gaps, localized displacement of moderator, or superficial scratches) may have no or little impact on the criticality analyses. However, the potential effects of these changes on the criticality analysis should, nevertheless, be evaluated and addressed through appropriate licensee processes.

10 REFERENCES

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