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

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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 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, independent of previous NRC guidance, and presents an acceptable approach to comply with the regulations.

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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 guidance document is intended to provide a clear and durable framework for the preparation of criticality analyses by industry and that would be acceptable to meet NRC regulations.

10 CFR 50.68 was promulgated in 1998 to provide an analysis alternative to the criticality monitoring required by 10 CFR 70.24. Prior to the rulemaking, exemptions to the monitoring requirement in 10 CFR 70.24 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." 10 CFR Part 52 was originally promulgated in 2007, and requires compliance with 10 CFR 50.68.

The first guidance on acceptable methods for performing criticality analyses at LWR plants, following promulgation of 10 CFR 50.68, was also issued in 1998 through an NRC internal memorandum from L. Kopp to T. Collins, often referred to as the "Kopp Letter" [NRC Memorandum from L. Kopp to T. Collins]. 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 Letter 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 Letter, and would better reflect the contemporaneous positions the staff had developed for acceptable criticality analysis methods in recent interactions with licensees.

NRC Interim Staff Guidance (ISG) DSS-ISG-2010-01, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," was issued in 2011 to formalize additional staff expectations for spent fuel pool storage rack criticality analyses. The guidance in DSS-ISG-2010-01 is useful to support NRC staff review of industry criticality analysis until more permanent and durable guidance is available.

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 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 62, "Prevention of Criticality in Fuel Storage and Handling."

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

- 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."

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.

1.3 DOUBLE CONTINGENCY PRINCIPLE

The double contingency principle, as described in ANSI/ANS 8.1, Section 4.4.2 [ANSI/ANS-8.1-1998; R2007], 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 postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

1.4 USE OF PRECEDENCE

The use of precedence (i.e., adopting methods 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. Similarities and any differences or 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 original use to the time of the intended use as precedence.

1.5 Assumptions and Engineering Judgment

In some instances, use of engineering judgment in criticality analyses can result in resource efficiencies, and in less common cases may be needed due to lack of conclusive information. In either situation, the use of engineering judgment should be applied in an appropriate manner. To use engineering judgment as a basis for a statement is acceptable as long as the applicant can demonstrate that the rationale behind such a statement is sound and justify that the engineering judgment would not lead to non-conservative results with respect to the regulatory requirements. The licensee, to the extent practicable, should provide a detailed technical basis supporting any and all assumptions defined in the application. Where no technical basis exists for the engineering judgment, the licensee should modify their approach such that, to the extent practicable, the criticality analyses can be performed without the use of that engineering judgment.

Engineering judgment 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 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

<u>Fresh (New) Fuel Storage</u>

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 fuel of the maximum permissible reactivity and flooded with pure un-borated water, the maximum k_{eff} shall be no greater than 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 fuel of the maximum permissible reactivity and flooded with moderator at the (low) density corresponding to optimum moderation, the maximum k_{eff} shall be no greater than 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.

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

Criticality safety analyses for pool storage of new and spent 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} shall be less than or equal to 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} shall be less than 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} shall be no greater than 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).

Applying a bias that reduces the calculated value of k_{eff} is typically not performed, but may be permissible if technically justified. 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 tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in 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 problem-dependent probabilities of interaction and the Monte Carlo code then track 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 is 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 well 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.

3.2 Computer Code Validation

The licensee should describe all computer codes that are used in the criticality safety analyses, including the validation of the codes. Validation of the codes includes benchmarking by 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 analyst/organization and the computer environment. The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible. The computer code validation for new fuel storage consists of validation for fresh, unburned fuel. The computer code validation for spent fuel storage consists of validation for used, depleted fuel analysis as well as validation of fresh, unburned, fuel.

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. [NUREG/CR-6698]

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 controlled by the following 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 credit is used (BWRs) or if credit for gadolinium in fresh fuel is used, and/or Ag/In/Cd if control rods are used in the criticality analysis.
- Spectrum
  - The spectrum can be affected by geometry and storage rack design (e.g., Region I flux design, Region 2 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 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* [OECD] and the HTC critical experiments [D.E. Mueller, K.R. Elam, and P.B. Fox] are considered an appropriate reference for criticality safety benchmarks. The handbook has reviewed the available 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.

In order to ensure a statistically appropriate validation, the selected set of experiments should be as large as possible. For example, a set of experiments that exceeds 100 experiments and comes from multiple independent sources, would likely minimize the impact of experimental biases.

3.2.1.3 Modeling the Experiments

Section 2.3 of NUREG/CR-6698 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.

3.2.1.4 Analysis of the Critical Experiment Results

NUREG/CR-6698 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 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 accurate or this approach can create the appearance of trends that are not statistically significant. The uncertainties provided in the OECD criticality benchmark handbook are sufficient for this purpose so the statistical approach defined in NUREG/CR-6698 and recommended by the NRC 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. If trends on other parameters are not analyzed, a justification should be made as to why they are insignificant, or are actually the result of spectral or enrichment trends embedded in the data.

The equation in Section 2.2 should be used to calculate the maximum k_{eff}. Alternatively, the method in NUREG/CR-6698 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 analysis covers an area of applicability, or also known as the "benchmark applicability". [ANSI/ANS-8.24] The criticality safety analysis should identify 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 that should be considered include [ANSI/ANS-8.24]:

- Nuclides, form, and composition 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 or process under consideration. [ANSI/ANS-8.24]

If a trend exists in a bias or uncertainty 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. However, all extrapolations should be justified. NUREG/CR-6698 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. Depletion validation that calculates a bias and uncertainty is preferred but is not the only acceptable approach, and in some cases may not be possible. If depletion validation does not calculate a bias and uncertainty, then the conservative bias and uncertainties that are used should be technically justified to the extent practicable based on available data. 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. The following are three acceptable methods for calculating a validation bias and uncertainty. In some cases, use of any of these may be impractical or would result in overly-conservative bias and uncertainty due to limitations in data. Therefore, alternatives approaches may be proposed by licensees, including approaches that do not directly calculate the bias and uncertainty, as stated earlier, provided they are adequately justified.

1. Use measured flux data to infer the depletion reactivity.
2. Use measured critical data from power plants,
3. Use chemical assays and worth experiments.

These three approaches will be described the next subsections.

3.2.2.1 Validation Using Depletion Reactivity Inferred from Measured Flux Distribution

PWR depletion benchmarks were developed by EPRI 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. [Kord Smith, Shaun Tarves, et al.] The depletion reactivity has been used to create 11 benchmark cases for 10, 20, 30, 40, 50, and 60 GWd/T and 3 cooling times 100 hour, 5 years, and 15 years. All of these benchmark cases should be analyzed with the depletion code 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 so the bias and uncertainty associated with fresh fuel should also 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. [D. Lancaster, EPRI Report]

3.2.2.2 Validation Using Measured Critical Data from Power Plants

For BWR fuel storage, an analysis for 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 criticality analysis tools may be benchmarked thoroughly against fresh fuel critical experiments, as described in Section 3.2.1. The major difference between these "fresh fuel critical experiment" benchmarks and true storage configurations is the isotopic composition of spent fuel. Therefore, it is the purpose of used fuel validation for BWR applications 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 peak reactivity condition.

One method to assess the accuracy of depletion codes in calculating spent fuel isotopes and their corresponding reactivity is to compare critical conditions from power plant startups with predicted eigenvalues. For example, 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. 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 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 typical control rod withdrawal sequence involves banked rod movements of blades 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.

By comparing the measured data to calculated results over a large range of startup experience, a bias and bias uncertainty can be assessed that is primarily a function of the lattice physics code's capability to assess nodal cross sections for the core simulator. These cross sections are directly a function of the nodal isotopics calculated by the lattice depletion code. As these nodal isotopics are then used with previously benchmarked criticality tools in storage rack analyses, the measured core critical bias and bias uncertainty is appropriate for use in the similar cold, controlled, unvoided in-rack configurations typically found in spent fuel pools. Although this bias and uncertainty contains contributions from both the isotopic content and cross sections it should be conservatively assumed to represent only the isotopic composition uncertainty.

The uncertainty in cross sections for isotopes not validated in the fresh fuel experiments but subsequently credited in criticality analysis tools can also be assessed. This can be broken into two components: actinides worth uncertainty and fission product worth uncertainty.

There is a set of experiments that include plutonium in concentrations consistent with used fuel known as the HTC critical experiments. [D.E. Mueller, K.R. Elam, and P.B. Fox] The inclusion of the HTC critical experiments in the fresh fuel validation can cover the major actinide worth uncertainty.

An acceptable means of assessing minor actinides and fission product worth uncertainty which are not explicitly represented in the critical experiments used would be to increase the uncertainty by an amount proportional to the reactivity worth of the isotopes not explicitly validated.

An alternative approach for the minor actinide and fission product worth uncertainty is to utilize the reactor measured data to cover both the isotopic and worth uncertainties but add a calculation that corrects for reactivity differences between the Monte Carlo code results and the lattice physics code results for the peak reactivity rack conditions.

3.2.2.3 Validation Using Chemical Assays and Worth Experiments

Depletion validation using chemical assays and worth experiments through PWR and BWR experiments were performed by ORNL. The validation method includes biases and uncertainties that can be used if the system and method match those used to produce the bias and uncertainty. It should be noted that this method may be overly conservative due to the large experimental errors in performing chemical assays. In this method the experimental error in measuring the

isotopic content is interpreted as uncertainty in prediction of the isotopic content. Studies in the past have shown that the criticality uncertainty changes very little with massive method changes which would be expected if the uncertainty is based on the uncertainty in the chemical assays rather than the uncertainty from the calculational method. A second major conservatism is due to limiting the number of fission product isotopes to 16 isotopes.

There are some modifications to the method that can be added for spent fuel pools to remove some of the conservatism. In NUREG/CR-7108 a direct difference method was presented [G. Radulescu and I. C. Gauld]. 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 cleanly accounts for the variation in isotopic content of the chemical assays, 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 to analyze the 100 chemical assays selected for NUREG/CR-7108 and then apply the direct difference approach 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 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 analyzed to justify the use of the chemical isotope 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 [D. E. Mueller, K. R. Elam, and P. B. Fox]). 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 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 [D. E. Mueller, J. M. Scaglione, J. C. Wagner, and W. J. Marshall]. The isotopes used in addition to the 28 NUREG/CR-7109 isotopes are expected to behave similarly so the use of 1.5% of the reactivity worth can be extended to cover these isotopes.

3.3 CODE-TO-CODE COMPARISON

If the use of a particular code is necessary, but the validation of that code by benchmarking to critical experiments is not feasible, then a code-to-code comparison may be considered. There is no accepted standard for performing a code-to-code comparison, therefore it would be expected that, at a minimum, the following conditions would be met for a code-to-code comparison.

A code-to-code comparison may be necessary if the primary code is not capable of modeling the benchmark experiments. In this type of code-to-code comparison, 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 (code used for comparison) should be validated by benchmarking to experiments that are similar to the neutronics and geometry of the criticality safety analysis. The primary code should 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. The bias and uncertainty of the primary code should include the biases and uncertainties from both the primary code validation and the secondary code validation.

A code-to-code comparison may also be considered if the area of applicability cannot be reasonably extrapolated to envelope the range of parameters for the criticality safety analysis. In this type of code-to-code comparison, both the primary and secondary code should be validated by benchmarking to critical experiments that are as similar as possible to the neutronics and geometry of the criticality safety analysis. A sufficient number of configurations that model the range of parameters of the criticality safety analyses should be performed by both the primary and secondary codes in order to establish a statistically appropriate bias and uncertainty of the comparison of the results. The bias and uncertainty of the primary code should include the biases and uncertainties from both the primary code validation and the code-to-code validation.

4 RACK AND FRESH FUEL MODELING

4.1 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. Alternatively, the analysis could calculated k_{eff} at the extremes of the manufacturing tolerances. To ensure that the maximum reactivity is being calculated per the requirement of 10CFR50.68, each parameter that may contribute to a significant positive reactivity effect should be perturbed. These parameters include assembly placement and manufacturing tolerances. The following fresh fuel assembly tolerances, at a minimum, should be considered:

- Fuel Pellet Density
- Fuel Enrichment
- Fuel Rod Cladding Outside Diameter
- Fuel Pellet Outside Diameter
- Fuel Rod Pitch
- Water Channels (BWR)

The fuel assembly mechanical tolerances should be evaluated in the appropriate rack model.

4.2 NEW FUEL VAULT

While the New Fuel Vault is normally 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 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 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, at a minimum, should be considered when evaluating the uncertainties due to tolerances:

- Storage Cell Wall Thickness (if present)
- Cell/Storage Location Pitch

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 capture any reactivity increases due to uncertainties associated with manufacturing tolerances, each parameter that may contribute to a positive reactivity effect should be evaluated. The following spent fuel pool rack tolerances, at a minimum, should be considered when evaluating the uncertainties due to tolerances:

- Cell/Storage Location Pitch
- Flux Trap Size
- Storage Cell Wall Thickness
- Eccentric Fuel Positioning
- Neutron Absorber Thickness, Width and Height
- Neutron Absorber Loading

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 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. 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 [EPRI Report 1019110].

Typically, neutron absorbers are not used in dry new fuel racks, where geometry alone prevents 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. The rack absorber and any supporting material should be modeled consistent with the guidance provided in Section 4.3.

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 fixed 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 is a critical parameter in the SFP criticality analysis. The most conservative modeling of boron content is to use the minimum boron concentration (typically described in terms of areal density in g/cm^2 ^{10}B). For existing absorbers, the minimum as-built areal density can be used if detailed manufacturing records are available. In some cases, these records have been collected by the manufacturer and provided with delivery on a batch basis.

If these records are not available, or if the application is submitted prior to completion of all manufacturing, then the minimum certified areal density can be used. This is based on the original material specification, and will likely result in an overly conservative calculation.

If the minimum as-built or minimum certified areal density is used, no tolerance calculation on boron content is required, since this value already represents the most conservative representation of the boron content.

Alternatively, the average areal density can be used based on as-built data. If this approach is used, the boron concentration tolerance should be included in the uncertainty calculations. The tolerance should be based on the difference between the average areal density and either the minimum as-built areal density or minimum certified areal density, depending on the availability of the data as described above.

4.4.3 Neutron Absorber Degradation

Certain neutron absorbers may be subject to degradation and aging effects of the neutron absorbing material. The mechanics and impact of this degradation is specific to the absorber

material and rack design. The criticality analysis should clearly identify the absorber assumptions and appropriate margin should be available to cover anticipated degradation (the appropriate margin can be zero if no boron loss over the life of the plant is anticipated).

Neutron absorber performance and aging characteristics are monitored through a surveillance program (see Section 9.5). If any un-anticipated aging or degradation is identified through the surveillance program, then it should be evaluated to determine if there is any impact on the criticality analyses and whether other licensee programs should be utilized (e.g., 10 CFR 50.59 process, operability evaluation).

5 CONFIGURATION MODELING

5.1 NORMAL CONDITIONS

The criticality analysis should consider all normal conditions and operations that occur in the spent fuel pool. That is, it is not sufficient to consider only just 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 all normal conditions, generally calculations are only required for the static condition. Normal conditions also include the normal range of water temperature for pool storage.

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. 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. In these cases, the maximum k_{eff} for the interface of more than one configuration must be less than the regulatory limit.

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 condition does 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.

5.3.1 Temperature

Abnormal pool water temperature (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated to consider the effect on criticality of loss of all cooling systems or coolant flow, unless the cooling system meets the single-failure criterion.

5.3.2 Dropped 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 misplacement of a 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 would be if there is not enough room to physically fit a fuel assembly in between the rack and the pool wall.

5.3.3 Assembly Misload

Misloading of a fuel assembly into an unapproved location, such as loading a PWR fresh fuel assembly with the highest enrichment into a storage location intended for a spent fuel assembly, should be evaluated as part of postulated accident scenarios.

Multiple misloads of fuel assemblies have the potential of occurring in spent fuel pools. 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 spent 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 allows soluble boron credit of up to 5%. 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

If soluble boron credit is credited, then 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. 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, which should include performing a boron dilution analysis. The boron dilution analysis should initiate at the minimum allowable, normal soluble boron concentration, should consider the concentration necessary to ensure k_{eff} does not exceed 0.95 under normal conditions in Section 6.1, and the time before action is 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 (such as a licensee taking credit for only a small percentage of the amount of boron actually expected to be present).

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 are the following:

- Relative power during depletion (which impacts the moderator and fuel temperatures during depletion);
- Soluble boron during depletion;
- Presence of burnable absorbers;
- Rodded operation; and
- Axial burnup shapes that maximize reactivity.

The nuclear criticality safety analysis should consider depleted fuel at the highest reactivity. NUREG/CR-6665 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 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. The higher the specific power the lower the reactivity of spent fuel due to a higher Sm-149 content after the decay of Pm-149. This effect is much smaller than the impact of the moderator and fuel temperature. The relative power should be selected to maximize the net reactivity of all the effects so the highest relative power should be used.

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. Typically, the higher the concentration during depletion, the greater the impact. The concentration used during the depletion calculations should therefore be justified. Since the soluble boron concentration typically decreases over the core cycle, a graded approach may be justified for 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., Gd, Erbium, are conservative if neglected [NUREG/CR-6760]. Therefore, these absorbers may not need to be considered. This does not apply to ZrB_2 IFBAs, which 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 [NUREG/CR-6759]. Note that most PWRs operate with all rods out. However, use of this assumption should be justified. Separate loading criteria may be used for separate assumptions on 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 (see NUREG/CR-6801). 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 some burnup, 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.

A licensee should account for the dependence of the burnup of the peak reactivity 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 can be selected 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 part length rods
    - SFP rack geometry
    - Cooling time
    - 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 for storage rack geometry 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

8.1 DECAY TIME

Credit may be taken for the change of isotopic content due to radioactive decay.

8.2 Fresh Integral Burnable Absorbers

Credit may be taken for integral burnable absorber in fresh fuel.

8.3 Used Removable Burnable Absorbers

Credit may be taken for the presence of removable burnable absorbers.

8.4 Control Rods

Credit may be taken for the presence of full-length rod control cluster assemblies (RCCAs) placed in selected fuel assemblies in the spent fuel pool.

8.5 Absorber Inserts

A number of absorber inserts have been designed for use in spent fuel pools. These include rods of absorber material that go into the control rod guide tubes of PWR assemblies and plates of absorber material that slide between the fuel assembly and the cell wall. Credit may be taken for these inserts.

There are absorber plate inserts that are placed in the cell and are not intended to be moved. These inserts should be handled as a normal part of the rack.

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 could range from technical specifications to control storage configurations and burnup/enrichment loading curves to procedures.

9.2 Administrative Controls

Under current regulations, a licensee has the responsibility to have administrative controls in place addressing the movement and storage of nuclear fuel assemblies, usually in the form of "plant procedures". These procedures implement the requirements for tracking the location of fuel assemblies due to Special Nuclear Material (SNM) regulations, criticality analysis requirements, to help ensure proper assembly selection for core loading activities, thermal management, gamma flux, etc. These administrative controls and "plant procedures" generally cover the following areas with those relating to minimizing multiple misloaded assemblies highlighted in italics:

- Pool Assembly Storage Planning
    - Fuel Characterization
        - Fuel reactivity category determination
            - Burnup
              (plots of burnup v enrichment to identify outliers, possible errors)
            - Enrichment
              (plots of burnup v enrichment to identify outliers, possible errors)
            - Decay time
        - Absorber inserts
    - Development of desired pool fuel assembly storage configurations
        - *Use of verified software application to confirm pool configuration is in accordance with the criticality analysis*
        - Independent verification of desired pool configuration
    - Development of Fuel Transfer Forms (FTF) to implement desired pool plan
        - *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 and surveillance
      (use of a larger soluble boron concentration to provide more reactivity hold-down to minimize the effect of assembly misloadings)
    - Absorber Insert material behavior surveillance program

In addition to the above, programs and procedures are in place to establish and enforce a software QA plan. Generally speaking, the following program and procedures should be in place to ensure that the licensee is following the established software QA program. This QA program should be in writing and regularly updated by the licensee. The requirements include:

- Software Requirements:
    - initial implementation or software revision, testing and documentation is performed by an independent reviewer
    - configuration controls ensure integrity of executable files and data files
    - cyber security controls prevent tampering / inadvertent changes
- Database Requirements:
    - All database updates are independently reviewed and approved

- o Administrative controls provide method for users to ensure integrity of database prior to utilizing the data

9.3 FUTURE FUEL TYPES

It is common for licensees to continually use newer fuel types that have better in reactor performance characteristics. However, it is impractical and extremely difficult 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 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 represent 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. The full criticality analysis of the future fuel should include all credible configuration 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 process should be used to determine whether NRC review and approval is necessary prior to storing the new fuel design.

9.4 PRE- AND POST-IRRADIATION FUEL CHARACTERIZATION

Fuel characterization is the process of ensuring that nuclear fuel assemblies are acceptable for insertion into the spent fuel racks. With regard to criticality analyses, this process includes ensuring that the fuel assemblies in question are adequately bounded by the assumptions concerning fuel characteristics in the criticality analysis itself.

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. This process is performed, in general, prior to delivery of the fuel in question to the plant site, and, in any case, 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, 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, some check should be performed to ensure that the fuel assemblies in

question were depleted in a manner consistent with the assumptions in the criticality analysis. Note that this is separate from the typical board 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.

Post-irradiation characterization will be 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 in this process include:

- Axial burnup shape
- Moderator temperature history
- Fuel temperature history
- Soluble boron history
- Control rod history
- Burnable absorber content and history (particularly if discrete, removable burnable absorbers are used)

Ideally, the process of post-irradiation characterization is performed 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 re-visited, however, if, during the fuel cycle, actual reactor operation differs significantly 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 perturb 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 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. In addition to the evaluation of the criticality effects of the differences between the actual and modeled characteristics, the licensee may need to utilize other processes to determine if any regulatory actions are necessary (i.e., determine operability or whether there is a non-conformance to the technical specification, and/or perform a 10 CFR 50.59 review to determine whether NRC review and approval is necessary)

9.5 NEUTRON ABSORBER SURVEILLANCE PROGRAMS

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 providing on-going confirmation that the neutron absorber

material is not undergoing any unanticipated degradation, that any material aging affects are accounted for in the criticality analyses, and that the condition of the neutron absorber material provides the criticality control relied upon in those analyses.

A neutron absorber surveillance program should be based upon either coupon surveillance or in-situ measurement. 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.

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:

- Number of coupons should be sufficient to provide sampling at an appropriate frequency 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 frequency is based upon the expected material degradation rate, which may be influenced based upon the qualification testing of the material. For materials without a long-term in-service life in conditions similar to the pool environment, the initial frequency should not exceed 5 years. For materials with long term in-service life in conditions similar to the pool environment, and for which stability in the material condition has been documented, a frequency up to 10 years is acceptable.
- Coupon testing can be categorized as basic or full testing. The extent to which each of these is utilized should be determined based upon the operating history of the material, as follows:
    - Basic testing consists of visual observations, dimensional measurements, and weight. 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 B-10 areal density.
    - Full testing may consist of a combination of density measurements, B-10 areal density measurements, microscopic analysis, and characterization of degradation. Full testing should be performed for the first coupon test. Basic testing may be used in combination with full testing for materials that have degradation resulting in loss of B-10 areal density to extend the frequency of full testing, if appropriately justified.
- Coupons should be located such that their exposure to parameters controlling degradation mechanisms (e.g., gamma fluence, heat) are similar to the in-service neutron absorbers. To the extent practicable, the coupon exposure should bound 95% of the in-service neutron absorber material.

9.5.2 In-situ Measurement

In-situ measurement is another acceptable method for confirming B-10 areal density of neutron absorber material. There are two potential uses for in-situ measurements:

1. Supplement coupon surveillance to extend the coupon testing frequency 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 frequency.
- Number of panels tested should be an appropriate statistical sample.
- Sampling frequency is based upon the expected material degradation rate, which may be influenced based upon the qualification testing of the material. For materials without a long-term in-service life in conditions similar to the pool environment, the initial frequency should not exceed 5 years. For materials with long term in-service life in conditions similar to the pool environment, and for which stability in the material condition has been documented, a frequency up to 10 years is acceptable. Note that sampling frequency can be longer if used in conjunction with coupons.
- The ability of the in-situ measurement method to measure B-10 areal density should be appropriately justified. This includes identifying uncertainties and explaining how they are addressed.
- For materials where potential degradation mechanisms do not result in a loss of B-10 areal density, in-situ measurements are used to confirm their presence. For materials where degradation mechanisms may result in a loss of B-10 areal density, in-situ measurements are used to determine the amount of B-10 areal density remaining.

9.5.3 Evaluating Neutron Absorber Surveillance Results

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

If unanticipated degradation is identified (either new degradation mechanisms or anticipated degradation 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 process), the following technical assessments may be necessary.

- Determine if unanticipated degradation could result in a loss of B-10 areal density (this is considered the only major impact to criticality control).
- Determine if unanticipated degradation not resulting in loss of B-10 areal density has any impact on the criticality analyses. Material degradation not resulting in B-10 areal density is expected to have little (e.g. formation of gaps) impact on the criticality analyses, or may not have any impact on the criticality analyses (e.g. localized displacement of moderator, or superficial scratches).

10 REFERENCES

10.1 REGULATIONS

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.

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

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

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.109 “Backfitting”.

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

10.2 STANDARDS

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