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May 31, 2016

Mr. Brian J. Benney
Project Manager, Division of Reactor Safety Systems
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Final Response to Request for Additional Information Related to NEI 12-16, *Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants,* and EPRI Report 3002003073, "Sensitivity Analysis for Spent Fuel Pool Criticality" and NEI 12-16, Revision 2 – DRAFT A

Project Number: 689

Dear Mr. Benney:

On behalf of the nuclear energy industry, the Nuclear Energy Institute (NEI)[1] is providing the remaining responses to the Request for Additional Information associated with NEI 12-16, Revision 1, and EPRI Report 3002003073 that was provided by the NRC to NEI via letter dated November 16, 2015 [Ref. 1]. NEI 12-16, Revision 1, was submitted to the NRC on April 18, 2014, [Ref. 2] after a series of four meetings between the nuclear industry and the NRC between September 2013 and February 2014. NEI 12-16, Revision 1, was extensively enhanced and expanded to be fully responsive to the comments provided by the NRC during the meeting and the action items captured in the summaries for each of the four meetings [Ref. 3-6].

Attachment 1 provides the remaining NEI responses to the NRC request for additional information (RAI) associated with NEI 12-16, Revision 1. Attachment 1, combined with the previous NEI letter [Ref. 7], provides a complete response to the NRC RAIs dated November 16, 2015. In conjunction with the final response, Attachment 2 contains a draft version of NEI 12-16, Revision 2, which has been updated to incorporate responses to the RAIs and minor editorial corrections.

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[1] The Nuclear Energy Institute (NEI) is the organization responsible for establishing unified industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations and entities involved in the nuclear energy industry.

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DESIGNATED AS ORIGINAL
BRIAN BENNEY 5/31/16

I look forward to continuing to work with the NRC staff to finalize this important guidance document by no later than September 2016. Please do not hesitate to contact me at any time with questions.

Sincerely,

Kristopher W. Cummings

References:

[1] REQUEST FOR ADDITIONAL INFORMATION RELATED TO NEI 12-16, "GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS" (ML15273A056)

[2] Submittal of NEI 12-16, *Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants*, Revision 1, dated April 2014 (ML14112A517, ML14112A516)

[3] SUMMARY OF SEPTEMBER 24, 2013, MEETING ON NEI 12-16, "GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS" (ML13268A115)

[4] SUMMARY OF OCTOBER 31, 2013, MEETING ON NEI 12-16, "GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS" (ML13309B558)

[5] SUMMARY OF JANUARY 17, 2014, MEETING ON NEI 12-16, "GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS" (ML14021A018)

[6] SUMMARY OF FEBRUARY 19, 2014, MEETING ON NEI 12-16 "GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS" (ML14051A252)

[7] Partial Response to Request for Additional Information Related to NEI 12-16, *Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants,* and EPRI Report 3002003073, "Sensitivity Analysis for Spent Fuel Pool Criticality", dated December 15th, 2016 (ML16005A216, ML16005A235)

Attachment 1: Response to Request for Additional Information Related to NEI 12-16, *Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants,* and EPRI Report 3002003073, "Sensitivity Analysis for Spent Fuel Pool Criticality"

Attachment 2: *Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants,* NEI 12-16, Revision 2 – DRAFT A

cc: Mr. William M. Dean, NRR, NRC
Ms. Michele G. Evans, NRR, NRC
Mr. Brian J. McDermott, NRR, NRC
Mr. Timothy J. McGinty, NRR/DSS, NRC
Mr. Robert M. Taylor, NRR/DSS, NRC
Mr. Eric R. Oesterle, NRR/DSS/SRXB, NRC
Mr. Jeremy L. Dean, NRR/DSS/SNPB, NRC
Mr. Kent A. L. Wood, NRR/DSS/SNPB, NRC
Mr. Amrit D. Patel, NRR/DSS/SNPB, NRC

Response to Request for Additional Information Related to NEI 12-16, *Guidance for Performing Criticality Analyses of Fuel Storage at Light Water Reactor Power Plants,* and EPRI Report 3002003073, "Sensitivity Analysis for Spent Fuel Pool Criticality"

NEI 12-16 Revision 1 was submitted to the NRC on April 18, 2014 [ML14112A516] after a series of four all-day meetings with the NRC between September 2013 and February 2014. In each of these meetings the industry and NRC developed action items and technical areas that needed further explanation in the guidance document. Additionally, the Electric Power Research Institute (EPRI) agreed to perform sensitivity analysis in support of the revised NEI guidance document. EPRI detailed the scope of the proposed work at the beginning of each meeting for NRC discussion and consensus. These discussions and identified action items are documented in each of the four meeting summaries [ML13268A115, ML13309B558, ML14021A018 and ML14051A252] that are available in ADAMs.

The following provides the remaining NEI responses to the NRC request for additional information (RAI) associated with NEI 12-16, Revision 1, dated November 16, 2015 [ML15273A056]. NEI previously provided a partial response to this RAI by letter dated December 16, 2015 [ML16005A235]. This Attachment combined with the previous NEI letter [ML16005A216] provide a complete response to the NRC RAI dated November 16, 2015. To facilitate the review, the RAI questions are shown below in italics followed by the response to each in normal type.

9. *The next to last sentence in the first paragraph discussing new fuel storage states; "Normal conditions (i.e., dry) need not be addressed in criticality safety analyses since there is no moderator." This is an inherent assumption that the new fuel storage facility is not also used for any other purpose, such as temporary storage of outage supplies. The guidance should require that inherent assumptions such as this one are captured in the nuclear criticality safety analysis.*

   Response:

   As a result of the response to RAI-31, Section 5.2.1 has been expanded to provide further details on the criticality analysis of the new fuel storage areas under flooded and optimum moderation conditions, which are not considered to be normal conditions. Modelling of the concrete and any other surrounding structures in the new fuel vault, especially under optimum moderation conditions, is important to consider because of neutron reflection in a low density moderator condition. See response to RAI-31 and updated Section 5.2.1.

14. *Revise the unnumbered subsection titled "Relative Power during Depletion" to provide a clear definition of and justification for the standard method(s). Several issues were noted with the analysis techniques described and with the justifications provided.*

    a. *The first sentence in the second paragraph of Section 4.2.1 is:*

    > *Since the effects of higher moderator temperature and higher fuel temperature can be approximated as linear [17], it is appropriate to use the maximum burnup-averaged relative power.*

    As written, the first part of the sentence is provided as the justification for the second part of the sentence. It is not clear that the intent of the referenced report was anything more than to suggest that the behavior varied in a somewhat linear behavior. In fact, in the third paragraph of Section 4.2.4 of the cited reference provides the following observation:

    > *Recent work[30] has shown that for high-burnup fuel with fission products present, the behavior of the SNF neutron multiplication as a function of specific power departs from a linear response.*

    From this text, it is not clear that the behavior is linear. It is also not clear that use of the "maximum burnup-averaged relative power is acceptable." Revise the text to provide justification for the claim that it is appropriate to use the maximum burnup-averaged relative power.

    Revise the report to carefully, objectively, and fully describe and use information extracted from other references.

    b. *Equation 1 provides a method to calculate a burnup-averaged relative assembly power. Provide text that explains how this quantity is to be used. It is clear how this equation can be used for fuel that will not be used again. It is less clear how this will be applied to fuel that will be used in the future. Describe how this equation will be applied to future fuel.*

    c. *The paragraph under Equation 1 includes the following text:*

    > *However, in criticality safety calculations, the reactivity of the system is reliant on the mass of fissile material in more than one fuel assembly. Therefore, it is appropriate to use the relative power for criticality safety calculations based on an assembly averaged parameter.*

    While it is a fact that the reactivity of the reactor is reliant on the fissile material in more than one assembly, it is not clear why this fact supports a conclusion that "it is appropriate to use the relative power for criticality safety calculations based on an assembly averaged parameter." Revise the discussion to more clearly describe and justify what was intended.

    d. *The second paragraph after Equation 1 includes the following statement:*

    > *A conservative moderator temperature would be the core outlet temperature increased by the relative power determined as stated above.*

i. *How would the relative power be used to increase the core outlet temperature?*

ii. *How would the moderator temperature (and presumably density) be determined?*

iii. *Revise the discussion on moderator temperature to clarify the recommended method for determining the moderator temperature and density to be used in depletion calculations.*

e. *The last paragraph under "Relative Power during Depletion" provides discussion on fuel temperatures for use in depletion calculations. Revise the text to clarify how the burnup averaged relative power is used to determine the fuel temperature.*

Response:

The first parameter in the list of depletion-related parameters provided in Section 4.2 has been changed from "Relative power during depletion (which impacts the moderator and fuel temperatures during depletion)" to "Power, Moderator Temperature and Fuel Temperature during Depletion"

Additionally, the subsection titled "Relative Power during Depletion" has been rewritten as follows:

"<u>Power, Moderator Temperature and Fuel Temperature during Depletion</u>

The power density, fuel temperature and moderator temperature (and associated moderator density) are grouped together because of the unique inter-relationship between these three values during in-reactor fuel depletion. The power density and moderator flow rate of a fuel assembly during depletion will directly impact the moderator and fuel temperature with a higher power (and/or lower moderator flow) resulting in higher moderator and fuel temperatures. Higher moderator and fuel temperatures during depletion result in increased reactivity of used fuel in the storage rack. While a higher power will lead to a higher ^{149}Sm content after the decay of ^{149}Pm, which lowers reactivity, this effect is much smaller than the impact of the moderator and fuel temperature. Therefore, depletion at credible but high power, moderator temperature, and fuel temperature is typically conservative. Previous studies [17] have also identified a small reactivity impact due to power history, with a low power coast down providing a conservative end of life reactivity. If load follow (variation of reactor power to adjust to demand) is exercised, this should be evaluated against the high constant power assumption.

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

A conservative (and computationally simpler) approach to the choice of depletion moderator and fuel temperatures would be to use a maximum value along the entire axial length of the fuel

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

20. *Provide clear guidance in 4.3, "Peak Reactivity Analysis for BWRs."*

    a. *NCS analysis guidance for the peak reactivity condition is discussed. Explicitly state that the BWR NCS analysis guidance is limited to the peak reactivity or augment the guidance accordingly.*

Response:

The title of Section 4.3 "Peak Reactivity Analysis" already clearly states applicability to peak reactivity analysis.

   b. *Define and provide guidance for the standard method(s) used for BWR SFP NCS. Describe and provide references for standard practices.*

   c. *Revise the report to add guidance addressing the following topics:*

   - *Modeling of axially and radially varying enrichments*
   - *Modeling of part length rods including the effect of non-fuel bearing portions of part length rods*
   - *Modeling of axial blankets*
   - *Establishing conservative in-reactor depletion conditions*
   - *Evaluating the impact of control blade usage*
   - *Modeling of bypass flow in water rods and outside fuel channels*
   - *2-dimensional (2-D) versus 3-dimensional (3D) modeling*
   - *Determining the bounding lattice(s)*
   - *Modeling with and without fuel channels in storage racks*
   - *Handling of fuel rod density, material specification, and treatment of associated uncertainties in fuel rods containing gadolinium oxide (Gd_2O_3 or alternatively Gd)*
   - *Nuclides modeled in burned fuel compositions*
   - *Supplemental criticality control requirements (i.e., minimum Gd-rod usage) that may be needed to ensure that the lattice designs evaluated in the NCS analysis cover fuel bundle designs that have been or will be used at the site*
   - *Fresh fuel storage rack NCS analysis*

Response (b, c):

Section 4.3 pertains specifically to the depletion portion of a BWR criticality analysis. Section 5 covers the in-rack criticality analysis modeling for PWRs and BWRS. Section 4.3 has been updated to provide additional detail for performing the peak reactivity depletion analysis for BWR spent fuel pools.

BWR assemblies are irradiated in the core with channels present, and therefore the depletion analysis should always include the channels in the depletion model. The reactivity impact of the channel presence or absence in the spent fuel pool has been added to Section 5.1.1.

22. *Section 4.3.2 discusses the determination of BWR depletion uncertainty. Revise this section to provide more comprehensive guidance on the calculation of BWR depletion uncertainty by addressing the following topics:*

    - *Fuel assembly design lattice variation*
    - *Moderator temperature and density variation*
    - *Nuclides credited in burned fuel compositions*
    - *Fuel channel presence*
    - *Calculation of reactivity decrement when Gd is not credited*

Response:

Section 4.3.2 is consistent with the previous guidance contained in Section 2.a of DSS-ISG-2010-001:

"**Depletion Analysis:** NCS analysis for SNF for both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs) typically includes a portion that simulates the use of fuel in a reactor. These depletion simulations are used to create the isotopic number densities used in the criticality analysis.

a. *Depletion Uncertainty:* The Kopp memorandum (Reference 2) states the following:

> A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

The staff should use the Kopp memorandum as follows:

i. "Depletion uncertainty" as cited in the Kopp memorandum should only be construed as covering the uncertainty in the isotopic number densities generated during the depletion simulations.

ii. The "reactivity decrement" should be the decrement associated with the k_{eff} of a fresh unburned fuel assembly that has no integral burnable neutron absorbers, to the keff of the fuel assembly with the burnup of interest either with or without residual integral burnable neutron absorbers, whichever results in the larger reactivity decrement."

In conjunction with the response to RAI-20, Section 4.3 of NEI 12-16 has been updated to provide additional detail on the BWR criticality analysis, including determination of the design basis lattice that considers fuel assembly design lattice variations, moderator and temperature variation. Presence or absence of the fuel channel is not considered as part of the depletion analysis, as the BWR reactor operates with the channel present, however the presence or absence of the channel should be considered in the criticality calculations.

The determination of the reactivity decrement when Gadolinia is not credited would also be based on the design basis lattice. Figure 4-5 of NEI 12-16 demonstrates pictorially how the reactivity decrement is calculated when Gadolinium is credited. The depletion uncertainty would be 5% of the

difference in reactivity at beginning of life and the burnup at which the maximum allowable k_{inf} in the SCCG occurs.

23. *The first sentence in Section 4.3.2 states:*

> *The BWR lattice physics/depletion codes used for SFP criticality analyses are the same depletion codes used and validated for BWR core design and core monitoring applications.*

This is an inherent assumption when using the "5% of the reactivity decrement" method for determining a depletion uncertainty. It is equally true for PWR as BWR usage of the method. Revise the report to capture the inherent assumption for both.

Response:

This section is specific to the determination of the depletion uncertainty for BWR spent fuel criticality calculations. The depletion uncertainty for PWR spent fuel pool criticality calculations is addressed in Section 4.2.3.

The depletion uncertainty covers only the uncertainty in the burned fuel compositions, not k_{eff} validation. The last two sentences in the last paragraph in Section 4.3.2 should be deleted because the uncertainty presented in the referenced NUREG/CR-7109 is not related to depletion uncertainty.

Response:

Agreed. The last two sentences in the last paragraph of Section 4.3.2 are deleted as treatment of the fission product and minor actinide bias is addressed in Section A.2. This Section has been revised to reference Section A.2:

> "Treatment of the bias associated with the minor actinides and fission products are addressed in Section A.2."

It can be inferred from the last two sentences in Section 4.3.2 that the fission product and minor actinide bias equal to 1.5% of the fission product and minor actinide worth described in NUREG/CR-7109 could be used to cover poor validation of k_{eff} calculations for BWR SNF with gadolinium nuclides built into the fuel. Provide a discussion in Section A.2 or elsewhere as appropriate to cover this issue explicitly and revise the existing guidance accordingly.

Response:

Section A.2 has been updated in accordance with the response to RAI-57, RAI-58 to address the use of NUREG/CR-7109.

25. *Regarding Section 5.1.1 covering determination of the design basis fuel assembly (DBFA):*

   a. *Revise this section to more comprehensively cover the method for identifying and using bounding fuel assembly designs. The demonstration that a design is bounding includes consideration of the effects of burnup, initial enrichment, allowed storage configurations (infinite array), variation in biases and uncertainties, credited soluble boron range, temperature range, moderator density range, accident conditions, etc. The DBFA determination analysis needs to consider all proposed, current, and past fuel assembly variations, including modified, damaged and consolidated fuel and must take into account changes in reactor operating conditions (e.g. uprates).*

   Response:

   The design basis assembly discussion already includes consideration of the effects of burnup. Section 5.1.1 has been updated to include consideration of initial enrichment. The design basis assembly would be determined based on a single assembly, infinite array basis at the moderator temperature and density that produces the highest reactivity for each storage configuration. Biases and uncertainties are only determined for the design basis assembly (See Section 5.1.2). Accident analysis should consider whether the design basis assembly is different for a fresh assembly versus at the burnup associated with the loading curve.

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

   Reactor operating conditions are considered in Section 4.2, with the most limiting operating conditions used in the determination of the design basis assembly.

   b. *Due to the use of SCCG-k-infinite-based methods in BWR SFP NCS analyses, expand the guidance to describe how the DBFA is determined for BWR fuel analyses.*

   Response:

   Section 4.3 has been updated in conjunction with the response to RAI-20 to more clearly describe the peak reactivity analysis for BWR spent fuel pools, including determination of the design basis assembly.

26. *Section 5.1.2 covers fuel assembly manufacturing tolerances. Revise this section to include guidance on:*

   a. *Handling of uncertainties associated with integral burnable absorbers(Gd/Erbium/IFBA). These include dimensions, axial location, pellet density, neutron absorber loading, and local variation of neutron absorber loading.*

   Response:

   As described in the response to RAI-16, the use of a planar averaged enrichment supports the approach of conservatively neglecting integral burnable absorbers such as Gadolinium or Erbium, therefore uncertainties are not needed to be addressed. Section 5.1.5 has been updated in conjunction with the response to RAI-30d to address the crediting of IFBA for fresh fuel.

   b. *Identification and handling of manufacturing tolerances and uncertainties that are not independent.*

   Response:

   See response to RAI-42.

   c. *Calculation of 95/95 uncertainty using sensitivity calculations.*

   Response:

   As described in Section 8:

   > "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."

   Section 8 has been updated to clarify that the recommended approach is a where the parameter of interest is varied to the maximum/minimum value allowed by the tolerance specification that maximizes reactivity. The reactivity effect of all tolerances are then combined statistically as indicated in Equation 2 in Section 8.

   d. *Variation of uncertainty with:*
      - *Other fuel assembly parameters such as enrichment*
      - *Storage configuration such as 2-of-4 storage and use of inserts*
      - *Environmental variations such as temperature, water density, and soluble boron concentration*
      - *Normal and accident conditions*

Response:

The fuel assembly tolerances need to be determined for the design basis fuel assembly (See RAI-25), in each unique storage rack design at the moderator temperature and density that maximizes reactivity with no soluble boron. The uncertainties need to be determined for different enrichments and burnup in conjunction with the established burnup/enrichment requirements for the proposed storage configuration(s). EPRI Report 3002003073, Section 5 includes analysis to justify use of the unborated uncertainties in the borated cases by reserving an additional 50ppm under borated conditions. This 50ppm covers the use of tolerance calculation in unborated conditions for the borated conditions and neglecting the spacer grid in the fuel assembly model.

For accident conditions, the uncertainties used in the normal conditions should be compared to the uncertainties associated with the limiting fuel assembly in the accident analysis (e.g., a fresh fuel assembly of maximum allowable enrichment). The maximum uncertainties should be used in the final determination of the maximum k_{eff} under accident conditions.

28. *Section 5.1.3 discusses handling of the axial burnup distribution:*

   a. *The guidance provides three options for determining the axial burnup distribution. Is one option considered to be the standard method or are all three options considered to be standard methods? Update the guidance accordingly.*

   Response:

   As discussed in Section 5.1.3, three options are provided for modeling the axial burnup distribution, "depending on the amount of information available to support the analysis." Option 1 could be used when no detailed information is available on the site-specific axial burnup distributions of the discharged fuel in the spent fuel pool. Option 2 could be used when site-specific axial burnup distributions are available and a simple method for determining a bounding profile is desired. Option 3 could be used when the licensee desires a more accurate determination of a bounding profile, without applying unnecessary conservatism inherent in Option 1 and 2. Given that all three options are preserved, no update to the guidance was deemed necessary.

   b. *The last sentence in the first paragraph says the results generated using the explicit axial burnup distribution should be compared to the results generated using an axially uniform burnup distribution. Revise this sentence to be a requirement rather than a recommendation.*

   c. *Expand the guidance to require examination of both distributed and uniform burnup profiles when performing calculations with mixed fresh, low burnup and high burnup fuel systems, such as fresh fuel checkerboarded with spent fuel or accident conditions with fresh fuel placed next to spent fuel.*

   Response (b, c):

   Agreed. The last sentence in the first paragraph is replaced with the following guidance:

   > "In all three options, the results with an explicit axial burnup distribution are compared to the axially uniform profile, which assumes the same burnup along the entire axial length. This includes all storage configurations, including those with different loading requirements in different storage cells (e.g., checkerboard of fresh and spent fuel, mixing of high and low burnup fuel, etc.)"

   d. *The following text near the top of Page 19 was extracted from NUREG/CR-6801:*

   > *because the axial blankets have significantly lower enrichment than the central region, the end effect for assemblies with axial blankets is typically very small or negative... consequently, profiles from assemblies with axial blankets were not considered...*

   NEI 12-16 then follows with the following statement:

   > *It is acceptable to use the profiles from NUREG/CR-6801 to bound axially blanketed fuel assemblies. It should be noted that this does not allow for credit of the lower enrichment of the axial blanket region in the criticality analysis.*

This discussion sends mixed messages. First the guidance indicates that using axial burnup profiles from NUREG/CR-6801 to analyze blanketed fuel assemblies is appropriate. If this is the intent of the guidance, provide further justification for this allowance. Then the guidance notes that modeling of natural or reduced enrichment axial blankets is not allowed.

The text from NUREG/CR-6801 does not say it is acceptable to use non-blanket profiles for blanketed fuel. Instead, it says profiles from blanketed assembly were not considered.

It appears that the report is defining the standard method as not crediting axial blankets (i.e., the full active fuel length will be modeled as enriched) and thus use of the profiles from NUREG/CR-6801 is appropriate. Clarify how blanketed fuel will be handled in the standard method and update the guidance accordingly.

Response:

It is agreed that the text in Section 5.1.3 for Option 1 is confusing with respect to fuel assemblies with low enriched or natural uranium blankets. The attempt of the discussion is to provide guidance to applicants who have fuel assemblies that contain low enriched or natural uranium blanketed fuel, but do not have access to the detailed axial burnup distributions (and for which there is not an equivalent database of axial burnup profiles similar to NUREG/CR-6801).

The general guidance is that the reactivity of fuel assemblies with no axial blankets (fully enriched the full length of the assembly) bound the reactivity of assemblies with axial blankets (low enriched or natural uranium). See the response to part e) below for the changes to NEI 12-16.

e. *Revise the text under Option 1 to note that the NUREG/CR-6801 axial burnup shapes are not to be used for:*

- *BWRs*
- *Natural or lower enrichment axial blankets*
- *Mixed blanketed/non-blanketed cores, such as non-blanketed fuel that was used during a transition to axially blanketed fuel*
- *All assemblies in cores that experienced atypical control rod usage, such as extended operation with control rods partially inserted*
- *All assemblies in cores that experienced atypical significantly reduced power level operation for extended periods*
- *Assemblies that had or were adjacent to assemblies that had part-length pressure vessel neutron fluence reduction inserts*
- *Fuel with initial enrichments below 1.2 wt. % ^{235}U*
- *New fuel assembly designs that vary significantly from those present in the database used to generate the NUREG/CR-6801 profiles.*

Response:

The intent of the guidance is not to stipulate all of the limitations (explicit or implicit) in the data base contained in NUREG/CR-6801, but to point out to potential applicants of the availability of the

database for PWR fuel assemblies. Some of the limitations listed above are readily apparent from reading the document (i.e., the title of the document makes it apparent that it is only applicable to PWR fuel assemblies). However, the last three paragraphs of Section 5.1.3, "Option 1" have been modified to include the requested clarifications:

> "Therefore, because the end effect for assemblies with axial blankets is very small or negative, it is acceptable to consider axially blanketed fuel assemblies bounded by fuel assemblies with no axial blankets.
>
> Because of the broad range of applicability and the conservative nature of using the most reactive axial burnup profile for each identified burnup range, there is reasonable assurance that axial burnup profiles from future discharged fuel assemblies will also be bounded by the database of profiles contained in NUREG/CR-6801. If drastic changes are made to the core operation (i.e., load following, significant low-power operation, gray rods, flux suppression assemblies, etc.), it should be verified that the new axial burnup distributions still behave in a similar manner as the axial burnup distribution before the core design change.
>
> The NUREG/CR-6801 limiting shapes were selected assuming the rack is uniform axially. If the rack has reduced length absorber panels that leave a significant portion of the active fuel outside of the absorber panels, new limiting axial burnup distributions must be determined."

It is noted that NUREG/CR-7108 also confirms the conservative nature of the axial burnup profiles contained in NUREG/CR-6801:

> "... the burnup values for the axial zones were based on burnup-dependent axial burnup profiles, which have been demonstrated to be conservative with respect to criticality (Refs. 41 and 42)."
>
> 41. *Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages*, DOE/RW-0472, Rev. 2, U.S. Department of Energy (1998).
>
> 42. J. C. Wagner, M. D. DeHart, and C. V. Parks, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*, NUREG/CR-6801 (ORNL/TM-2001/273), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tennessee (2003)."

f. Revise the text under Option 2 to define and provide specific guidance for determination of the plant-specific bounding profiles. Describe the process, acceptable data sources, minimum data set sizes, whether not this option is burnup-dependent, how burnup-dependent data is to be used, identification and documentation of limitations on using the plant-specific bounding profiles, and the reload-specific checks that must be made to ensure use of profiles with future fuel is acceptable.

Response:

The determination of the plant specific bounding profile would be determined by the licensee. For a bounding plant specific profile (Option 2) there would be no need to do a reload check as taking the

minimum relative burnup at each node produces a weighted relative burnup of approximately 0.95 (versus a normalized burnup profile equal to 1.0).

g. *Revise the text under Option 3 to define and provide specific guidance for determination of the most reactive plant specific profile.*

Response:

The determination of the plant most reactive profile would be on an application specific basis. Specific guidance is not provided in NEI 12-16 as analysis would be needed to justify that the selected profile produces a conservative reactivity. No changes were made to NEI 12-16.

h. Revise the text under the Section 5.1.3 sub-heading, "Nodalization," to define the standard method and provide guidance for using an alternative method.

Response:

The recommended approach is to utilize equally sized nodes along the length of the active fuel no larger than eight inches. This recommendation is added to Section 5.1.3.

29. *The second paragraph of Section 5.1.4 includes the following sentence:*

    It should be noted that both studies indicate that when using properly calibrated core follow software which is updated with in-core measurements the uncertainty is less than 2%.

    The ORNL report, Reference 21, merely reported information from another reference and did not endorse use of the 2% value.

    a. *Revise the basis for using a value of 5% for the burnup measurement uncertainty (BMU) to accurately reflect ORNL's conclusions as stated in the second paragraph of Section 7.2 of Reference 21:*

        Based on these comparisons, it may be concluded that the uncertainty in the utility-assigned burnup values is less than 5%.

        And from Section 8 of Reference 21:

        There is a significant amount of data available from 1980 to the present to support a finding that utility records for fuel burnup are accurate for individual spent fuel assemblies to at least 5% of "true" assembly burnup.

Response:

There are several places within NUREG/CR-6998 that indicate that the uncertainty of the reactor record burnup value is less than 5%:

Executive Summary:

"The modern in-core systems, which use core measurements and some data from design codes, produce burnup values that are generally within approximately 2% of the burnup values predicted by design codes"

Section 3.4.2

"A subsequent evaluation showed an uncertainty of less than 2% between the utility-calculated burnup and the gamma-ray spectroscopy-measured burnup.

Section 4.5

"The report stated that 'with respect to utility-supplied data it was concluded that because utilities measure total core power to within about 1% and make extensive in-core measurements of power distribution, assembly-average burnup values at end of life can be determined to within about 2% (to one standard deviation) of actual values.'"

Section 7.2

"Using the total core power information and in-core flux maps, assembly average burnup values were determined to within about 2% of the predicted burnup values when computer models,

correctly normalized to start-of-cycle conditions and adjusted periodically on the basis of in-core measurements"

Therefore, the NUREG/CR-6998 supports the sentence referenced in this RAI.

b. *Clarify that the standard method is to use a 5% BMU, as the guidance implies, and the decision to defend using a lower value should be left to an applicant.*

Response:

Agreed. The last sentence in the second paragraph of Section 5.1.4 has been updated to more clearly specify the need to justify a lower value on an application specific basis:

> "It should be noted that both studies indicate that when using properly calibrated core follow software which is updated with in-core measurements the uncertainty is less than 2%, however this would need to be justified on an application-specific basis."

c. *Provide guidance that directs the analyst to clearly document whether the BMU is to be included in the burnup credit limits or whether it is to be included when fuel assembly burnups are compared to the burnup credit limits by plant operators.*

Response:

The last paragraph in Section 5.1.4 already includes guidance that the burnup measurement uncertainty may be either included as an uncertainty for statistical combination with other uncertainties or that the fuel assembly recorded burnup may be reduced by 5% for the purposes of comparing to established burnup versus enrichment limits. The checklist referenced in the response to RAI-2 has been updated to allow applicants to clearly specify which option was chosen.

30. *Address the following comments regarding Section 5.1.5, "Assembly Inserts," by revising the report accordingly:*

   a. *The second sentence in the second paragraph appears to permit credit for neutron absorption in BWR control rod blades. The wording "control rod blade" should be changed to "control rods" to maintain consistency of terminology throughout the subsection.*

   Response:

   Agreed. The terminology "control rod blade" is changed to "control rods"

   b. *Use of any insert, fresh or used, will require extra analysis work in the area of uncertainty analysis, accident analysis, and validation. Provide details for the standard method for crediting inserts.*

   Response:

   Section 5.1.5 has been updated to include additional details to support the analysis to credit assembly inserts. Materials of construction of inserts should be included in the validation. Additional uncertainty analysis is not needed if the applicant assumes a conservatively low neutron absorber content. Accident analysis should consider the consequences of a removable insert not being contained in the assembly (however this will typically be bounded by the fresh fuel misload accident analysis already described in Section 6.3.3) See also the response to RAI-30c.

   c. *The third paragraph is the following:*

      Non-irradiated removable burnable absorbers (i.e., WABA's, BPRA's) can also be credited to provide additional reduction in the required burnup for storage. The primary effect is associated with moderator displacement from the guide tube and can provide some small benefit.

      i. *The second sentence says the primary effect is moderator displacement. For non-irradiated removable burnable absorbers, moderator displacement would be a secondary effect since the presence of the strong neutron absorbing material would be associated with the primary negative reactivity effect. Furthermore, both depleted and fresh burnable absorbers have been credited in SFP NCS analysis. Provide clarification for what was intended.*

      ii. *Provide appropriate guidance for the use of removable burnable absorbers. Guidance should include discussion of poison loading, manufacturing tolerances, uncertainties, and modeling techniques.*

      iii. *Some licensees use borated stainless steel rods inserted into the fuel assemblies for reactivity control. These are assembly inserts but are not currently addressed in this section nor are they include elsewhere in NEI 12-16. Is the intention that they be excluded from the standard guidance? If so, NEI 12-16 should explicitly state this intention. Otherwise provide appropriate guidance on the use of borated stainless*

steel rods inserted into the fuel. Guidance should include discussion of poison loading, manufacturing tolerances, uncertainties, and modeling techniques.

Response:

The third paragraph of Section 5.1.5 has been updated to include additional generic guidance for crediting the presence of removable assembly inserts, either irradiated or non-irradiated:

"Non-irradiated removable burnable absorbers (i.e., WABA's, BPRA's, borated SS rods) can be credited to provide additional reduction in the required burnup for storage. The primary effect is associated with crediting the neutron absorption capabilities of the insert, with a secondary effect associated with moderator displacement from the guide tube. A conservative approach is to model the insert with nominal geometrical dimensions in conjunction with a minimum absorber loading. In this approach, additional uncertainty analysis for the absorber in not necessary.

Irradiated removable burnable absorbers (i.e., WABA's, BPRA's) can also be credited to provide additional reduction in the required burnup for storage. Since the strong neutron absorber is no longer present the primary effect is associated with moderator displacement from the guide tube and can provide some small benefit. Residual absorber should not be credited without additional justification."

d. *The final paragraph in Section 5.1.5 discusses credit of burnable absorbers that are integral to the fuel assembly, but provides no guidance on how that would be accomplished.*

   - *Provide appropriate guidance for the use of integral burnable absorbers. Guidance should include discussion of poison loading, manufacturing tolerances, uncertainties, and modeling techniques.*

Response

The final paragraph of Section 5.1.5 has been updated to include additional generic guidance for crediting the presence of integral absorbers in fresh fuel:

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

   - *Since integral burnable absorbers are not inserted into the assembly, the section title is misleading.*

Response

Agreed. The title of Section 5.1.5 is changed to "Assembly Inserts and Integral Absorber Credit."

31. *Address the following comments on Section 5.2.1, "New Fuel Vault," by revising the report accordingly:*

    a. *Provide guidance and justification for the standard method, covering at least the following issues:*

    - *Assembly structure to be modeled*
    - *Rack structure to be modeled*
    - *2D vs 3D modeling and infinite versus finite modeling*
    - *Concrete modeling considerations for the walls and floor*
    - *Model of space above and below fuel*
    - *Model of space outside the new fuel vault*
    - *Performing calculations at enough water density points to capture the optimum moderation peak k_{eff} and the k_{eff} for full density water*
    - *Normal and credible abnormal temperature range*
    - *Uncertainties associated with size and spacing of storage cells, concrete composition, and spacing to walls and floor*
    - *Whether any material is or can be stored in the new fuel vault*

    Response:

    Section 5.2.1 has been updated to more fully describe the analysis performed for the new fuel vault. This includes addressing the issues identified in the RAI.

    b. *Why is it recommended rather than required that tolerance calculations be performed for both full and optimum moderation?*

    Response:

    The response to RAI-3 includes a discussion on the use of permissive statements, such as "recommended" versus "required." In addition, the discussion during the public meeting on February 19th, 2016 on the first partial RAI response indicates that this change is no longer necessary.

    c. *Provide guidance for when analysis of NFSV flooding may be omitted. Include essential elements to be considered and recommended documentation that would be needed to justify not analyzing NFSV flooding.*

    Response:

    Providing guidance on when analysis of flooding of the NFSV is outside the scope of NEI 12-16. No changes were made to NEI 12-16 as a result of this RAI.

32. *Somewhere in Section 5.2.2 or one of its subsections, add guidance for spent fuel storage rack modeling.*

    a. *Describe and justify one or more standard modeling approaches, including modeling approximations and simplifications. If a simplified fuel storage rack model is adopted, it is necessary to perform sufficient calculations with detailed rack models to justify the simplified model as conservative or to generate a conservative bias term to cover the effects of the simplification.*

Response:

The first paragraph in Section 5.2.2 has been updated to provide guidance for modelling of the spent fuel storage rack:

> The spent fuel pool rack criticality model consists of a representation of the dimensions and materials of construction, including any installed neutron absorber as well as flux traps (if present). The rack structure should be modeled using nominal dimensions with an axial length equal to the active fuel region. If the neutron absorber does not extend the entire length of the active fuel region, it should be appropriately modelled, depending on the location of the active fuel region in relation to the neutron absorber. The rack structure above and below the active fuel region are neglected and replaced with unborated water (even when borated water is used in the active fuel region). It is acceptable for minor parts of the rack construction (i.e., welds) to be neglected and replaced by water. Credit can be taken for radial leakage near the walls of the spent fuel pool for the purposes of allowing lower burnup fuel requirements on the periphery of the spent fuel pool. In general, the concrete composition has a negligible impact on reactivity [31], but a conservative dry concrete density of 2.90 g/cm^3 is recommended [31].
>
> To ensure the model captures any reactivity increases due to uncertainties associated with manufacturing tolerances, each parameter that may contribute to a significant positive reactivity effect should be evaluated. The following spent fuel pool rack tolerances should be, at a minimum, considered when evaluating the uncertainties due to tolerances:"

b. *Provide guidance discussing necessary axial modeling detail. An evaluation must consider the fuel in its most reactive approved position, which may vary depending on fuel assembly properties, fuel storage rack design, and storage configuration.*

Response:

See response to RAI-32a

c. *Is the standard approach a 3D infinite array of storage cells?*

Response:

For PWR storage racks, a 3D infinite array of storage cells with design basis fuel is the recommended approach. This may be an infinite array of a single storage cell, or a larger 2x2 or 3x3 model, depending on the proposed storage configurations. For BWR storage racks, a 2D infinite array of storage cells with design basis fuel (modeled as a single storage cell) is recommended.

33. *Including "eccentric fuel positioning" in the list of uncertainties documented in Section 5.2.2 is unjustified. Including this as an uncertainty presupposes, without justification, that fuel assemblies will not be naturally, mechanically, or intentionally placed in their most reactive position. Also, spent fuel assemblies are not all straight and bowed collections of fuel rods are not typically modeled in SFP NCS analyses. Furthermore, even one static spent fuel assembly has variable lateral location within its storage cell.*

    While NEI 12-16 Reference 31 did provide some analysis of eccentric assembly positioning, the analysis was too limited to draw general conclusions concerning the most reactive location for assemblies in the storage locations. It will be necessary to provide either additional generic analysis or application-specific analysis.

    Describe the standard method for determining the most reactive assembly locations within spent fuel storage racks. This should include consideration of the potential for reactivity variation with water density variation, different storage arrangements, and other criticality control measures.

Response:

The calculations performed in Reference 31 of NEI 12-16 were discussed during the series of meetings between September 2013 and February 2014, with the goal of reaching agreement on the scope of analysis needed to address various issues, including the impact of eccentric positioning. A review of Section 8 of Reference 31 allows the following conclusions to be reached:

- When neutron absorber panels are present, a centrally located positioning of the fuel assembly in the storage cell is the most reactive configuration.
- When the neutron absorber is not present (or not credited), an eccentrically located positioning of the fuel assembly in the storage cell is the most reactive configuration:
    - As the size of the model increases (and therefore more assemblies are eccentrically located) the reactivity increases. However, the likelihood of an increasing number of fuel assemblies being eccentrically located in the most reactive configuration also decreases.

While it would be conservative to consider the maximum impact due to eccentric positioning from a very large model (e.g., 400 assemblies in a 20x20 array all located eccentrically towards the center of the storage rack) as a bias, it also unnecessarily penalizes the analysis for a configuration that is not deemed credible.

The true issue is in determining a credible number of eccentrically located fuel assemblies and determining the maximum reactivity effect associated with this configuration. If it is conservatively considered that a fuel assembly may either be in a centrally or eccentrically located position (a binary problem), then the likelihood of all fuel assemblies being in the eccentrically located position is equal to $(1/2)^N$, where N is the number of assemblies in the configuration. Table RAI-33.1 below

includes the likelihood of all fuel assemblies being located eccentrically to the center of the configuration model for several different sized models.

Table RAI-33.1

Number of Assemblies	Array	Likelihood
1	Single Cell	0.5
4	2x2	6.25E-02
9	3x3	1.95E-03
16	4x4	1.53E-05
25	5x5	2.98E-08
400	20x20	3.9E-121

Therefore, it is proposed that to balance the size of the model and capturing the effect of eccentric positioning appropriately, a 4x4 model (16 assemblies) eccentrically located to the center of the model (all assemblies pushed to the center), with reflecting boundary conditions be used to capture the maximum reactivity effect associated with eccentric positioning. This reactivity effect would be applied as a bias to the design basis, centrally located results. Alternatively, the applicant can incorporate eccentric positioning into the design basis calculation models, so that the reactivity impact is already captured in the calculation of k_{eff}.

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

Eccentric positioning is removed from the list of uncertainties in Section 5.2.2.

38. *Revise Section 6, "Configuration Modeling," and its subsections to more accurately describe the scope of normal and accident conditions that needs to be considered based on the following comments:*

   a. *Normal conditions are those that are allowed and could occur during plant operations. Care should be taken to include consideration of all potential operations that are not prevented by operational controls. Credit should not be taken for informal operational habits, such as an undocumented practice to not move a fuel assembly closer than one foot from other fuel. The analysis of normal conditions includes both static and transient operations such as fuel stored in storage racks, fuel in an inspection station or elevator, fuel inspection, fuel on pedestals in the storage racks, fuel reconstitution or repair, movement of fuel to and from storage racks, movement of fuel near other fuel in approved locations. When soluble boron is credited, all of these conditions would need to be shown to be subcritical without soluble boron. Except where prohibited by operational controls, occurrence of multiple, proximal normal operations should be anticipated and evaluated. The normal conditions analysis must also include consideration of the full range of permitted environmental conditions such as spent fuel pool temperature.*

   Response:

   Section 6.1 of NEI 12-16 already addresses many of the issues identified in RAI-38a, however Section 6.1 has been updated to reflect the comments above that are not already addressed in this section:

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

   The evaluation of temperature variation under normal and accident conditions are already addressed in Section 5.2.2.1, "Spent Fuel Pool Temperature" and Section 6.3.1, "Temperatures Beyond Normal Operating Range," respectively.

   b. *The abnormal and accident conditions to be considered typically involve credible human performance or equipment failures, or the credible consequences of facility events such as flooding, fires, and earthquakes. It is the applicant's responsibility to identify and analyze all credible abnormal events.*

   Abnormal conditions or accidents to consider include, but are not limited to the following:

- *One or more misloaded assemblies*
  - *wrong assembly moved*
  - *wrong loading curve used*
  - *assembly moved to incorrect location*
  - *assembly enrichment/burnup/cooling time mischaracterized*
- *Non-compliant region, sub-region and rack module interface condition*
- *Dropped or otherwise damaged/modified fuel assembly*
- *Dropped heavy load on or near fuel*
- *Dropped consolidated fuel canister*
- *Soluble boron dilution*
- *Depletion of ^{10}B in soluble boron*
- *Misload, loss or failure of assembly reactivity control devices*
- *Loss of spent fuel cooling (i.e., high spent fuel pool temperature)*
- *Assembly movement to prohibited position*
- *Violation of assembly spacing requirement during fuel handling, inspection, repair, etc.*
- *Fuel storage rack movement, modification, or damage*
- *Fuel storage rack insert removal, modification, or damage*
- *NFSV flooding during fuel receipt, handling, or storage*

Response:

Many of the issues identified in RAI-38b are already addressed in Section 6.3 of NEI 12-16 or through the conservative modelling of normal conditions (e.g., neglecting the gaps between adjacent storage racks in an infinite array model eliminates the need to evaluate seismic shifting of the storage racks). Those issues not already included in NEI 12-16, Revision 1 have been evaluated and included (where applicable) in the new revision of NEI 12-16 (i.e., see NEI 12-16, Revision 2). The following is an assessment of those items already addressed in NEI 12-16, and those that will be addressed in the new revision:

- *One or more misloaded assemblies*
  - *wrong assembly moved*
  - *wrong loading curve used*
  - *assembly moved to incorrect location*
  - *assembly enrichment/burnup/cooling time mischaracterized*

Response: See Section 6.3.3

- *Non-compliant region, sub-region and rack module interface condition*

Response: See Section 6.3.3

- *Dropped or otherwise damaged/modified fuel assembly*

Response: Bounded by the single assembly fresh fuel misload

- *Dropped heavy load on or near fuel*

Response: Heavy loads are prohibited over fuel in the spent fuel pool by operational controls

- *Dropped consolidated fuel canister*

Response: Bounded by the single assembly fresh fuel misload

- *Soluble boron dilution*

Response: See Section 7.3

- *Depletion of ^{10}B in soluble boron*

Response: Neutron flux levels in the SFP are many orders of magnitude lower than in a reactor, so neutron absorption by soluble boron will not have a measurable effect on the ^{10}B content of soluble boron. Fresh boron comes from sources constrained by the natural abundance of ^{10}B in boron deposits, so new boron added to the SFP will not affect the ^{10}B content (unless higher than natural ^{10}B has been credited in the criticality analysis). A plausible way to gradually reduce the ^{10}B ratio of soluble boron in the SFP is to regularly introduce recycled boron into the SFP either during refueling operations of re-filling of the SFP. For plants that recycle soluble boron, ^{10}B content can be directly verified by periodic measurements. Alternatively, a bounding low ^{10}B content can be justified and used in the portion of the SFP criticality analysis crediting soluble boron.

- *Misload, loss or failure of assembly reactivity control devices*

Response: Added to Section 6.3

- *Loss of spent fuel cooling (i.e., high spent fuel pool temperature)*

Response: See Section 6.3.1

- *Assembly movement to prohibited position*

Response: See Section 6.3.2

- *Violation of assembly spacing requirement during fuel handling, inspection, repair, etc.*

Response: See Section 6.3.2

- *Fuel storage rack movement, modification, or damage*

Response: See Section 6.3.4

- *Fuel storage rack insert removal, modification, or damage*

Response: Added to Section 6.3

- *NFSV flooding during fuel receipt, handling, or storage*

Response: See Section 5.2.1

c. *It is also important to consider the potential for what are termed common mode failures. These events may cause multiple changes whose concurrent occurrence might otherwise be considered to be beyond the DCP.*

Response:

Section 6.3.3 already addresses the potential for a "single initiation event" to result in multiple fuel assemblies being misloaded. The specific aspects of a defense-in-depth program to reduce the likelihood of such an event or mitigate its impact is already addressed in Section 6.3.3.

d. *NCS analyses must identify and justify the credible size of accidents and failures. Of particular interest is the misloading of multiple fuel assemblies. NEI 12-16 section 6.3.3 and its sub-sections go into detail about licensees having a "...multi-tier defense-in-depth program in place to prevent or mitigate the severity of a scenario where multiple assemblies are located into the wrong storage locations." However, this discussion appears to be more philosophical in nature rather than specific guidance. Revise the report to identify specific* guidance.

Response:

See response to RAI-40d

39. *Consider the following comments regarding Section 6.2 and revise the report accordingly:*

    a. *Change "should" to "shall" in the first sentence of the first paragraph of Section 6.2.*

Response:

The response to RAI-3 includes a discussion on the use of permissive statements, such as "should" versus "shall." The spent fuel pool configuration could include adjacent racks being separated by an adequate distance that would preclude the need for a detailed interface analysis.

    b. *The guidance does not appear to capture the need to evaluate transitions within the same rack design. For example the gap between rack modules should be considered. Revise the guidance to include the transitions within the same rack design, or demonstrate that they don't need to be considered.*

Response:

The first sentence of Section 6.2 already includes guidance that an interface may exist either between different rack designs or between different storage configurations with a single rack design (i.e., 2x2 checkerboard and uniform loading).

    c. *Provide a definition for "storage configuration."*

Response:

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

    d. *The second sentence in the third paragraph of Section 6.2 specifies that the resulting keff of the interface cannot be less than the keff of the two individual storage configurations. However, the first bulleted criterion under that paragraph is that the calculated keff for the interface is less than the keff for both regions. These statements are in conflict and should be revised accordingly. Furthermore, the last sentence in the third paragraph says that if the criteria following the paragraph are met, no further restrictions are needed. The second bulleted item says that if the interface condition has a higher keff than either region, then restrictions should be specified, which is counter to the last sentence in the third paragraph. Revise the text in Section 6.2 to clearly define the standard method and to eliminate the contradictions.*

Response:

The third and fourth paragraphs of Section 6.2 has been updated as shown below to resolve the identified contradictions:

> "When an interface calculation is performed, essentially two semi-infinite arrays of each storage configuration are placed in the same model, possibly with a small gap between them in the case

of rack-to-rack interfaces (i.e., no leakage is credited). If the model is sufficiently large enough (4 or more rows of storage cells of each configuration), the resulting k_{eff} of the interface can determine if the interface results in a more limiting condition than the individual storage configurations. If the interface calculations show that the reactivity of the interface is essentially equivalent to (within the statistical uncertainty of the calculations) or less than the reactivity of the most reactive of the two storage configurations, then there is no additional neutronic coupling between the individual storage configurations. The following criteria are specified for interface calculations between two different storage configuration:

- If the calculated k_{eff} of the interface model is less than or equal to (within the statistical uncertainty of the calculations) the maximum calculated k_{eff} of each individual storage configuration, then no further restrictions are necessary.
- If the interface calculation has a higher calculated k_{eff} (outside the statistical uncertainty of the calculations), compared to the maximum calculated k_{eff}, of each individual storage configuration, then appropriate restrictions should be specified to limit these storage patterns from being used adjacent to one another.

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

e. *The last paragraph in Section 6.2 states:*

> *"If the separation distance between the new and old racks is more than 6 inches at the interface, then there is no need to evaluate the interface between storage racks/configurations."*

Provide justification for this assertion. Considering the unqualified nature of this assertion, the justification should address spacing between various rack module designs. Some older non-poisoned low-density rack modules had significant center-to-center spacing. Is it acceptable to move these low density rack modules to within 6 inches of each other? The justification should also include consideration of the impact of SFP temperature variation.

Response:

The last paragraph of Section 6.2 is not applicable to older "non-poisoned low density rack module designs" that relied upon geometric separation for maintaining sufficient sub-criticality to meet regulatory requirements. However, it is applicable to high-density storage rack designs, with or without neutron absorber, that rely on burnup credit (PWR) and/or maximum reactivity credit (BWR) for maintain sub-criticality.

40. *Consider the following comments regarding Section 6.3.3, "Assembly Misload," and revise the report accordingly:*

  a. *The last sentence in the first paragraph of Section 6.3.3 states:*

     For all storage configurations, an evaluation of a fresh fuel assembly of the maximum allowable enrichment, with no burnable absorbers should be evaluated in the storage location that provides the largest positive reactivity increase.

  It is necessary for the analyst to identify the most limiting case. While the guidance provided is good, it does not ensure the limiting case is identified. For SFPs that credit soluble boron to meet sub-criticality requirements, the limiting misload, and all accidents/abnormal conditions, will be the accident which requires the most soluble boron to ensure compliance with the requirement that k-effective must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water. Revise the guidance accordingly.

  Response:

  Agreed. The following sentence is added to the end of the first paragraph of Section 6.3.3:

  "For PWR spent fuel pools that credit soluble boron, the limiting misload will be the accident which requires the highest soluble boron to ensure that the maximum k_{eff} does not exceed 0.95."

  b. *The second paragraph in Section 6.3.3 discusses fuel assembly misloads for BWR SFPs. The first sentence discusses "a single region of uniformly loaded fuel." A checkerboard pattern of fuel assemblies and empty cells would be considered to be "uniformly loaded," and in this case a misload analysis would need to be considered. Revise the guidance to clarify when a misload may not be required.*

  Response:

  The interpretation of "uniformly loaded fuel" to encompass a checkerboard of fuel assemblies and empty storage cells is not consistent with the commonly identified definition of uniform loading. Uniform loading is only applicable to a homogenous loading of the spent fuel storage rack with the same limiting fuel assembly in every storage location. The first sentence in the second paragraph of Section 6.3.3 has been modified to read:

  "For BWRs spent fuel pools that contain a homogeneous loading of the spent fuel storage rack with fuel with a limiting peak reactivity in each storage location (i.e. uniform loading), the misload event does not need to be considered."

  c. *The Section 6.3.3 second sentence says that "...a misloaded bundle with highest peak reactivity limit should be evaluated." Misloaded fuel assemblies must be assumed to be the most reactive allowed by license, therefore this statement is applicable for PWR SFPs as well as BWR SFPs. Revise the guidance to provide the necessary clarity.*

Response:

The second paragraph of Section 6.3.3 is applicable to BWRs only. The analysis of a misloaded fuel assembly for PWR spent fuel pools is contained in the first paragraph of Section 6.3.3. The first paragraph of Section 6.3.3 has been updated to clarify that it is applicable to PWR spent fuel pools.

d. *It is not clear how a NCS analyst will use the discussion in Sections 6.3.3.1 through 6.3.3.4. Revise the guidance to clarify why this information is included and how the NCS analyst is to use the information.*

Response:

The potential to reduce the likelihood of misloading multiple fuel assemblies can be most effectively controlled through the multitier, defense-in-depth approach described at the end of Section 6.3.3 and detailed in Section 6.3.3.1 to 6.3.3.4. Each of these approaches could limit the type or number of misloaded fuel assemblies. For example, empty storage cells that have a blocking device would eliminate the possibility of multiple fresh fuel assemblies from being loaded into empty storage cells, which could have a significant reactivity impact. The extent of implementation of each of these features would be determined on a site-specific basis, but could significantly limit the likelihood and/or impact of multiple misloaded assembly scenarios, thereby increasing safety.

e. *The next to last paragraph in Section 6.3.3.4 ends with the phrase "Multiple Misload Analysis." It appears that the authors may have meant this to be the heading for a new section. Similarly, the final sentence of Section 6.3.3.4 contains the word "the procedural." Review and revise as appropriate.*

Response:

Agreed. The discussion of multiple misload analysis at the end of Section 6.3.3 will be contained in Section 6.3.3.5, "Multiple Misload Analysis." The final sentence of Section 6.3.3.4 (now Section 6.3.3.5) is corrected.

41. *Consider the following comments regarding Section 7, "Soluble Boron Credit," and revise the report accordingly:*

    a. *Provide more detail on how the boron dilution accident analysis is to be performed and the associated acceptance criteria or provide a reference that does.*

    Response:

    A detailed analytical method for performing the boron dilution accident analysis is beyond the scope of NEI 12-16.

    b. *Provide guidance for how the boron dilution accident analysis is to be documented in the criticality analysis report.*

    Response:

    The checklist contained in Appendix C has included a line item for the boron dilution accident analysis. Additionally, Appendix B includes a discussion to include the boron dilution analysis (if applicable) in the applicant's submittal.

    c. *The last sentence in Section 7.3 is:*

    > *Similarly, an analysis that determines that the credible soluble boron dilution event would not reduce boron below the required amount in less than 24 hours, would not need to provide additional justification for the assumptions in the boron dilution analysis if it can be credibly shown that action would be taken to prevent further dilution in less than 8 hours.*

    The phrase "if it can be credibly shown" does not accurately describe the intent of the specified criterion. It must be demonstrated that the boron dilution will be identified and remedied within the claimed period. An appropriate criterion would specify that failure to identify and remedy boron dilution in less than 8 hours is not credible. Revise the text to clarify the analysis criterion.

    Response:

    The last paragraph in Section 7.3 has been removed because it does not provide any additional value. The criteria associated with acceptability of the boron dilution analysis is already contained in the last sentence of the first paragraph in Section 7.3.

42. *Provide guidance for handling uncertainties that are correlated (i.e. not independent) and for handling of non-conservative biases and uncertainties*

Response:

The uncertainties identified for inclusion in the spent fuel pool criticality analysis are independent, and thus are treated as such. No guidance is provided for correlated uncertainties as none were identified. Development of a statistical approach to address correlated uncertainties is beyond the scope of the guidance document, whose intent is described in the response to RAI-1.

It is not anticipated that an applicant would include non-conservative biases or uncertainties in an analysis submitted to the NRC for approval, but would include the appropriate conservatisms in the model, or apply a *conservative* bias or uncertainty. In past applications, licensees have included biases and/or uncertainties to ensure that the maximum k_{eff} is conservative for comparison to the regulatory limit. A prime example is the use of a bias to account for the difference in temperature of the Monte-Carlo criticality calculations (where cross-sections are valid at 300K) and the temperature and density of maximum reactivity at 4°C.

43. *In the list of biases, include:*

- *Lattice code bias*
- *Modeling simplification biases*
- *Safety margin covering validation gaps/deficiencies?*
- *Other biases*

Response:

The list of biases in Section 8 intended is to capture those biases that are identified in NEI 12-16 however is not an exhaustive compilation of all biases that could possibly exist, depending upon how an individual applicant performs the analysis. The suggested biases will be added to the list of biases in Section 8, with the exception of "modeling simplification biases" and "other biases:"

Biases

Criticality Code Validation Bias

Moderator Temperature Bias

Design Basis Fuel Assembly Bias

Eccentric Positioning Bias

Depletion Code Bias

Actinide and Fission Product Worth Bias

Bias for Validation Gaps

44. *In the list of uncertainties, include:*

- *Facility structural and material uncertainties*
- *Lattice code uncertainties*
- *Modeling simplification uncertainties*
- *Uncertainties for validation gaps/deficiencies*
- *Other uncertainties*

Response:

The list of uncertainties in Section 8 is intended to capture the uncertainties that are identified in NEI 12-16, however is not an exhaustive list of all uncertainties that could possibly exist, depending upon how an individual applicant performs the analysis. The current list of uncertainties already addresses the lattice code uncertainty (i.e., depletion code uncertainty). Section 8 has been updated to expand upon the list of uncertainties, with the exception of "modeling simplification uncertainties" and "other uncertainties:"

Uncertainties

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

46. *Section A.1 draws heavily upon NUREG/CR-6698 for the criticality code validation methodology. Even when NUREG/CR-6698 is not explicitly referenced, the discussion appears to assume that analysts are using the methodology described in NUREG/CR-6698. If the intent is to recommend that the NUREG/CR-6698 methodology be used for criticality code validation, explicitly state as such and make it clear that analysts are expected to follow the methodology documented in NUREG/CR-6698 in its entirety, with supplemental guidance provided in NEI 12-16. If an alternative methodology may be used, provide an equivalent level of detail to NUREG/CR-6698, including but not limited to the following:*

    - *Critical experiment selection criteria;*
    - *Trending methodology;*
    - *Definition, documentation, and use of the area of applicability;*
    - *Statistical analysis methodology to determine the bias and uncertainty;*
    - *Format and content of the criticality calculation validation study documentation.*

    Response:

    In general, NUREG/CR-6698 is acceptable guidance for performing the validation of the criticality code using fresh fuel critical benchmark experiments. Section A.1 is intended to supplement the guidance contained in NUREG/CR-6698 specifically for spent fuel storage racks. This was deemed prudent, given that NUREG/CR-6698 was developed for nuclear fuel cycle facility licensees as described in the abstract on NUREG/CR-6698. However, the methodology contained in NUREG/CR-6698 is sufficiently generic to be applicable to criticality codes analyzing spent fuel storage racks (and critical benchmark experiments that simulate spent fuel storage arrangements).

    Therefore, NUREG/CR-6698 provides useful guidance for performing the validation, with supplemental guidance provided by Section A.1.

48. *Add text to Section A.1 that includes potential impacts from spent fuel isotopic modeling and new fuel vault rack configurations to the range of parameters to be considered in the validation.*

Response:

The response to RAI-47 clarified the purposes of Section A.1 and A.2 in Appendix A, as Criticality Code Validation Using Fresh Fuel Experiments" and "Depletion Code Validation", respectively.

Given that Section A.1 is for the purposes of providing guidance on the validation of the criticality code using fresh fuel experiments, it would not be applicable to include the potential impact from spent fuel isotopic modeling in Section A.1.

Additionally, Section A.1 is already written generically to address the applicability of the fresh fuel validation to the new fuel vault analysis, in the fully flooded condition. However, there are limited critical benchmark experiments to cover the optimum moderation condition for the new fuel vault. New fuel vault racks are typically designed as part of an open rack structure (storage cell walls do not extend the length of the fuel assembly), but have the same materials, fuel geometry and general structure as the spent fuel pool racks. Therefore, it is recommended to apply the criticality code validation using fresh fuel experiments to the optimum moderation condition.

50. *Revise the text in Section A.1.1 to consider concrete (when modeled), and temperature.*

Response:

Section A.1.1 has been updated to address materials in the surrounding geometry (if applicable) and temperature variation:

- Isotopic Content
    - Experiments should cover material for the rack structure (e.g., stainless steel), materials in the surrounding geometry (e.g., water/concrete), material for the cladding (e.g., zirconium), fissile isotopes in the applicable enrichment range (e.g., ^{235}U for low enriched UO_2, ^{239}Pu for MOX), water temperature, 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.

52. *The second paragraph in Section A.1.2 says selected experiments should ensure a statistically appropriate validation. Change this recommendation to a requirement.*

Response:

The response to RAI-3 includes a discussion on the use of permissive statements, such as "should" versus "shall." In addition, the discussion during the public meeting on February 19th, 2016 on the first partial RAI response indicates that this change is no longer necessary.

54. *Provide additional guidance for performance of trending analysis, addressing the following:*

    a. *Significance level to use for trending analysis*

    b. *What to do with trending analysis results*

Response:

The guidance contained in NUREG/CR-6698, supplemented by the discussion in Section A.1.4 and Section A.1.5, already provides an acceptable level of guidance for performing trend analysis. Therefore, no revisions were needed to NEI 12-16 as a result of this RAI.

57. *The 5th paragraph in Section A.2.1.2 includes the following text:*

> *It has been shown that the isotopes in excess of the 28 major isotopes selected in NUREG/CR-7108 have a relatively small worth. Therefore, for analyses crediting more than the 28 major isotopes, it is recommended that the bias and uncertainty from the chemical assays be applied for all isotopes.*

The fifth paragraph in Section A.2.1.2 discusses treatment of the bias and uncertainty associated with nuclides not explicitly addressed in NUREG/CR-7108. However, the meaning of the last sentence is not clear. It seems to support the approach recommended in the next paragraph, but the link is not clearly indicated in the text. Rewrite the text to clarify that the conclusion in the last sentence leads to the use of the 1.5% bias for all nuclides, even though this bias was determined using a more limited set of nuclides.

Response:

The fifth paragraph in Section A.2.1.2 has been updated as requested:

> "Both of these chemical assay approaches result in a conservative estimate of the bias and uncertainty for the 28 major isotopes selected. However, NUREG/CR-7108 states that most of the bias and uncertainty (90-95%) is attributable to ^{235}U and ^{239}Pu. It has been shown that the isotopes in excess of the 28 major isotopes selected in NUREG/CR-7108 have a relatively small worth. Therefore, for analyses crediting more than the 28 major isotopes, it is recommended that the bias and uncertainty from the chemical assays also be augmented by an additional bias for other minor actinides and fission products based on NUREG/CR-7109 [15]."

58. *The 6th paragraph in Section A.2.1.2 states,*

"NUREG/CR-7109 [15] recommends a bias of 1.5 % of the reactivity worth of the isotopes not included in benchmark critical experiments to cover cross section bias and uncertainty. The isotopes used in addition to the 28 isotopes directly evaluated are expected to behave similarly so a bias of 1.5% of the reactivity worth of all depletion isotopes except U, Pu, and ^{241}Am is recommended. This recommendation applies for calculations using ENDF/B-VII cross sections. Additional justification should be provided for evaluations using other cross section data."

a. *The NUREG/CR-7109 recommendation is caveated. Revise NEI 12-16 to capture the limitations and condition associated with the recommendation.*

b. *Additionally, the NUREG/CR-7109 recommendation is applicable to ENDF/B-V and VI nuclear data libraries as well as the ENDF/B-VII library. Revise the NEI 12-16 accordingly.*

Response (a, b):

The sixth paragraph in Section A.2.1.2 has been updated as follows:

"NUREG/CR-7109 [15] recommends a bias of 1.5 % of the reactivity worth of the isotopes not included in benchmark critical experiments to cover cross section bias and uncertainty. The isotopes used in addition to the 28 isotopes directly evaluated are expected to behave similarly so a bias of 1.5% of the reactivity worth of all depletion isotopes except U, Pu, and ^{241}Am is recommended. This recommendation applies to enrichments less than 5.0 wt% ^{235}U (nominal) using ENDF/B-V, ENDF/B-VI and ENDF/B-VII cross sections, for burnups up to 70 GWD/MTU and total minor actinide and fission product nuclide worth not to exceed 0.1 Δk. Additional justification should be provided for evaluations using other cross section data."

60. *Sections A.2.2.1 and A.3 does not appear to have a logical numbering scheme relative to the rest of the document. Revise the section numbering so it is clear how the information in these sections is organized.*

Response:

Section A.2.2.1 is intended to describe an option available for BWR licensees to benchmark the depletion and criticality code in conjunction through the cold-critical control configuration of BWR reactors prior to start-up, therefore, it is applicable to be placed in Section A.2 which is retitled "Depletion Code Validation" (See RAI-47).

Meanwhile, Section A.3 provides guidance on how to apply the depletion and criticality code validation results to the overall determination of the maximum k_{eff}.

Method A is considered the conventional approach where 1) the criticality code is validated with fresh fuel benchmark experiments to validate the fissile and major actinides, 2) the depletion code bias and bias uncertainty is relevant to the isotopic composition of spent fuel (i.e., use of the 5% depletion uncertainty, and 3) minor actinides and fission products not explicitly represented in the critical experiments are covered by the approach described in Section A.2.1.2 (i.e., NUREG/CR-7109).

Method B covers the approach contained in Section A.2.2.1 where the BWR start-up configurations are used to validate the depletion code and criticality code in conjunction.

No changes were made to NEI 12-16 as a result of this RAI.

61. *Sections A.2.2 through A.3 describe a methodology for using measured cold critical data from BWR start-ups in code validation. These sections do not provide enough detail for NRC staff to judge the acceptability of the proposed methodology. The following details are lacking:*

    a. *Determination of the measured keff value and its uncertainty;*
    b. *Definition of the benchmark models;*
    c. *Generation of the calculated keff values and uncertainties, including uncertainties in reactor geometry, materials, and conditions, as well as modeling assumptions;*
    d. *Applicability of in-reactor bias and uncertainty values to the full range of storage rack configuration and conditions;*
    e. *A technical basis that the final bias and uncertainty values will ensure that the required subcriticality margin is maintained with a 95% probability and with a 95% confidence level.*

    Provide details or a reference with details, or remove these sections.

    Response:

    It is recognized that the approach contained in Section A.2.2 has not been implemented by any individual licensee to support a BWR spent fuel pool criticality analysis. However, this approach has potential advantages over the more complex separate depletion code and criticality code validation described in Section A.1 and application of the 5% reactivity decrement used for the depletion code uncertainty. Therefore, this section is maintained to provide guidance to industry that such an approach is viable. The details requested in the RAI would be determined on an application specific basis, if a licensee chose to take the approach detailed in Section A.2.2.

62. *Section A.4 addresses validation through code-to-code comparisons. Use of codes that cannot be directly validated must address the additional bias and bias uncertainty associated with the code-to-code-to-experiment (CCE) validation. A CCE validation involves the elements listed below.*

   a. *Secondary code validation using benchmark experiments. This is the standard computational methodology validation.*

   Response:

   The third sentence of Section A.4 already contains this requirement:

   > "The secondary code should be validated by benchmarking to experiments that are similar to the neutronics and geometry of the criticality safety analysis."

   b. *Code-to-code comparison validation. The bias and uncertainty determined as part of (a) were determined based on critical experiment geometries and compositions, while the final criticality analyses will be performed using different geometries and compositions AND a different code methodology. Therefore, an evaluation of the applicability of the bias and uncertainties determined in (a) for the secondary code to the primary code methodology needs to be performed, considering the entire range of parameters of interest for the safety analysis.*

   c. *Code-to-code validation. The primary code is validated by comparison to secondary code results over the full range of parameters considered in the safety analysis (geometries, compositions, lattice design variations, boundary conditions, temperature variation, neutron absorbers, depletion-specific characteristics, etc.). This validation should be comparable to (a), including trending analysis.*

   Response (b, c):

   The bulleted list in Section A.4 already provides a listing of the parameters of interest to be validated between the primary and secondary code:

   > "The primary code can then be validated by benchmarking to the secondary code over a range of parameters (neutronic and geometric) that bound the range of parameters for the criticality safety analysis. Those parameters that are important to be validated between the primary and secondary code include:
   >
   > - Enrichment,
   > - Burnup,
   > - Absorber areal density,
   > - Soluble boron content, and
   > - Storage rack geometry"

   However, Section A.4 has been updated to add Energy Spectrum to the list of parameters (see below).

   d. *Application of any additional penalty associated with validation gaps or identified weaknesses associated with the CCE validation.*

Response:

Agreed. Section A.4 of Appendix A has been updated (see below).

Please provide more detailed guidance that discusses each element (with the exception of (a)) in greater depth, or remove this section.

Responses:

Section A.4 of Appendix A has been updated to address the questions contained in the RAI:

"If a code (the primary code) is not capable of directly modeling the benchmark experiments, then an intermediary code (i.e., a secondary code) may be used that is validated to the benchmark experiments, and to which the primary code is validated. The primary code (code used for the criticality safety analyses) should still be capable of accurately modeling all the important neutronic and geometric aspects of storage. The secondary code should be validated against benchmark experiments that are similar to the neutronics and geometry of the criticality safety analysis in accordance with Section A.1. The primary code can then be validated by benchmarking to the secondary code over a range of parameters (neutronic and geometric) that bound the range of parameters for the criticality safety analysis. Those parameters that are important to be validated between the primary and secondary code include:

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

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

Response to Request for Additional Information Related to EPRI Report 3002003073, "Sensitivity Analysis for Spent Fuel Pool Criticality"

2. *EPRI TR 3002003073 was distributed by EPRI in December 2014 and is intended to be supporting analysis for NEI 12-16. Numerous discrepancies with the computer models used to perform the analysis were noted. The following is a partial list:*

   a. *In the CE16x16 fuel depletion calculations, the model did not include water in the volume between fuel assemblies.*

   b. *The burned fuel calculations used a fuel density of 10.34 g UO_2/cm^3. The fresh fuel calculations used a fuel density of 96% of the UO_2 maximum theoretical density (i.e., 10.5216 g UO_2/cm^3). Thus, the fresh fuel calculations had 1.8% more fuel than the burned fuel compositions.*

   c. *In the W17x17std fuel depletion calculations, soluble boron was omitted from the IFBA fuel pin unit cell model.*

   d. *In the fresh fuel, region 1, checkerboard calculations, parts of the rack structure are missing.*

Response (a-d):

As discussed in response to RAI-1, originally the report was reviewed by four reviewers but not the input files. After becoming aware of the error in one of the input files, EPRI had independent reviews of the input files performed.

The error identified in RAI-2.a has been corrected. It should be noted that although the computational results showed changes in delta-k values for some of the cases, none of the conclusions of the report are impacted.

For RAI-2.b, it is true that the density is different; however, it should be noted that the report compares the delta-k values and reach conclusions based on those values. Furthermore, the study does not make any recommendation as a function of burnup; hence, there is no impact.

For RAI-2c, the review of input files showed that soluble boron was not omitted from the IFBA fuel pin unit cell model (material 31).

For RAI-2d, the review of the input files did not show any missing structure.

   e. *A review of the gadolinium depletion calculations revealed that the gadolinium rods were depleted using the constant power option (default), rather than the constant flux option that is recommended in the TRITON primer (NUREG/CR-7041, page A-3).*

Response:

The computations were repeated using constant flux option. However, since the results are reported as delta-k, no impact on the reported results were observed, as expected.

Revise the computer models to correct all the discrepancies (including any others not listed above that may exist) and revise the report to reflect the use of the corrected computer models.

Response:

As indicated in response to RAI-1 and RAI-2, the models have been reviewed and corresponding tables have been updated. No changes to conclusions were made since no impact on the conclusions were observed as this is a qualitative study that is mainly interested in delta-k values.

4. *All fuel depletion calculations in this study were performed using the SCALE TRITON t5-depl sequence. This appears to be a method that has not been previously used to support NRC approved licenses. It is not clear whether or not use of this non-standard method affects the results of the sensitivity studies.*

   Provide precedents where use of this method has been accepted by the NRC or documented verification studies demonstrating the acceptability of the method. Unless they are addressed in the precedents or verification studies, describe how the following were handled in the sensitivity studies:

   a. *How was local flux convergence ensured?*
   b. *How was k_{eff} convergence checked?*
   c. *Confirm that all calculations passed the KENO generation k_{eff} value distribution normality tests.*
   d. *Confirm that all warning and error messages were reviewed.*
   e. *What is the burnup dependent bias associated with this method?*
   f. *How was time-dependent depletion of IFBA verified?*
   g. *How was time-dependent depletion of WABA verified?*
   h. *How was time-dependent depletion of gadolinium verified?*
   i. *How well did use of mixture average depletion compare with pin-by-pin depletion?*
   j. *Justify the model (i.e., single annular absorber region using infinite-homogeneous resonance processing) used for depletion of WABA rods.*
   k. *Describe and justify the model used for the IFBA fuel rods. Address IFBA layer composition and IFBA fuel rod "celldata" model, which did not include the IFBA layer.*
   l. *Confirm that the modeling approach (calculational options, cross section processing options, depletion strategies, modeling simplifications, mesh/grid sizes, etc.) is consistent with the precedents or verification studies.*

Response:

The use of SCALE TRITON t5-depl sequence was discussed during the NRC-NEI public meetings on October 31st, 2013 and January 17th, 2014. Furthermore, this depletion code has been used at least one recent license amendment application (Millstone Unit 2 – ML12362A392). The sensitivity analysis contained in EPRI Report 3002003073 focuses primarily on differences in reactivity to determine whether various modeling approaches have an impact on reactivity. Additionally, industry has not requested NRC to endorse the EPRI Report 3002003073.

a. Depletion of fuel was not performed on a pin-wise basis, but assuming all pins are the same mixture (except for assemblies containing IFBA or Gadolinium where all IFBA or gadolinium pins are another separate material from all non-IFBA pins). Additionally, the 3-D model is setup with reflective boundary conditions effectively equivalent to a 2-D model, with only 1cm input as axial length. Considering this, the convergence of the average flux is highly likely in the areas of concern (namely, the fuel and burnable absorber regions which are depleted).
b. Convergence of k_{eff} is ensured by the following:
   i. Visual inspection of the trend in k_{eff} given in the KENO output file.

ii. Ensuring that the k_{eff} Chi-squared test for normality is passed at the 95% level.

c. This is confirmed by inspection.
d. All warning and error messaged were reviewed and dispositioned.
e. A burnup dependent bias was not determined for this work. See Note 1 below.
f. Benchmarking of depletion of burnable absorbers was not performed for this work. See Note 1 below.
g. See response to f.
h. See response to f.
i. A comparison was not performed. Assembly average isotopics are considered acceptable as is standard practice for spent fuel pool criticality analyses.
j. See Note 1.
k. See Note 1.
l. The modeling approach (calculational options, cross section processing options, depletion strategies, modeling simplifications, mesh/grid sizes, etc.) is consistent with the precedents including one recent license amendment application (Millstone Unit 2 – ML12362A392).

Note 1: This work focuses on sensitivity analyses performed to determine the impact of certain parameters on the criticality analysis and to provide technical justification on low worth items that have a negligible impact on reactivity. Evaluation of low worth items was determined via calculated reactivity impact from direct simulation. Comparing simulation in this manner will assure that any minor concern associated with the depletion analysis will not impact the final conclusions of the study.

5. *Section 2.3.1 describes the depletion models that were used in the analysis. Rather than using the decay functions in SCALE 6.1, a "...short program was used to correct the isotopic inventory due to radioactive decay for 100 hours at the desired burnups." With respect to the "short program" provide the following information:*

   a. *A description of the data used, where the decay and branching fraction data came from, exactly what decay calculations were performed, how simultaneous in-growth and decay were handled, and any simplifications used by the program.*

   b. *Provide a demonstration of the accuracy of the short program by performing at least two detailed calculations, one around 10 GWd/MTU and one at maximum burnup, and comparing the short program results to the detailed calculated results.*

   c. *Some of the burned fuel compositions were modified manually. This includes a "coast-down" model for ^{149}Pm/^{149}Sm. Provide a demonstration of the adequacy of the end-of-cycle model used by performing a fuel depletion calculation, to maximum burnup, that explicitly models reduced power level at the end of the cycle. Provide the details of how the end of cycle was modeled (e.g. fuel depleted at 50% power for final 15 days). Describe how this model compares with actual end-of-cycle operations. Provide the in-rack Δk_{eff} values obtained when using, and not using, the end-of-cycle reduced power model.*

   d. *From a review of the composition files, it appears that ^{239}Np and ^{149}Pm were retained in the burned fuel compositions used. Confirm that the ^{239}Pu and ^{149}Sm were indeed modified as described in the report.*

Response:

The purpose of the analysis contained in the EPRI sensitivity report is to determine the impact of certain parameters on reactivity in spent fuel storage racks and to provide technical justification to demonstrate that certain parameters or modelling approaches have a negligible impact on reactivity. Subsequently, future applications would not require extensive analysis in areas that have a negligible impact on the final results. Elimination of low impact items from the analysis would be beneficial for both applicant and reviewer.

The short program described in Section 2.3.1 does not perform explicit decay calculations, but simply automates manual editing operations to conservatively represent fuel at 100 hours decay time. These operations were intended as follows:

i. Addition of ^{135}I and ^{135}Xe number density to the ^{135}Cs number density, zeroing the ^{135}I and ^{135}Xe number densities.
ii. Addition of the zero decay time ^{105}Ru number density to the ^{105}Rh number density, zeroing the ^{105}Ru number density.
iii. Addition of the zero decay time ^{239}Np to the ^{239}Pu number density. Consequently, only 181 isotopes are followed in the criticality analysis.

iv. Additional isotopes are eliminated if their atom densities are less than 1E-12 atoms per barn cm.

During independent review, the computations were repeated without using the short program and demonstrated that there is no impact on the results (i.e., the same reactivity differences were computed).

Additionally, the EPRI has reviewed the items in this RAI regarding the modification of the isotopic inventory described above due to the simplified approach does not have any impact on any conclusions within this report since reactivity comparisons are determined as differences in reactivity between calculations with models containing the same assumptions.

6. *Table 2-5 provides a WABA rod model that is based on the WABA rod model presented in EPRI TR1022909. Appendix B of TR1022909 indicates that NUREG/CR-6761 was used to develop its WABA model. NUREG/CR-6761 further references an OCRWM report (B00000000-01717-5705-00064, Revision 01; ADAMS Accession No. ML033530015). No reference is provided for the overall density of the material or for the composition of the material other than the boron linear loading.*

   A comparison of the data from these two reports and the EPRI model is provided below:

	EPRI	NUREG/CR-6761	OCRWM
all	3.6 g/cm^3	2.593 g/cm^3	2.527 g/cm^3
B_4C	2.16 wt %	14 wt %	14 wt %
Al	51.78 wt %	45.52 wt %	45.52 wt %
O	46.05 wt %	40.48 wt %	40.48 wt %
^{10}B	1.38 wt %	2.02 wt %	2.02 wt %
^{10}B	6.006 mg/cm	6.165 mg/cm	6.165 mg/cm
^{11}B	0	8.94 wt %	8.94 wt %
C	0.78 wt %	3.04 wt %	3.04 wt %

   Provide complete references for the WABA rod composition model used. Update the models and report as appropriate. Provide justification for modeling simplifications used.

   Response:

   As described in the response to RAI-5, the purpose of the analysis contained in the EPRI sensitivity report is to determine the impact of certain parameters on reactivity in spent fuel storage racks and to provide technical justification that certain parameters or modelling approaches have a negligible impact on reactivity and therefore, can be neglected in future analysis/applications. For the analysis contained in EPRI 3002003073, the WABA specification has a second order effect on the Δk results, and does not impact the conclusions reached. The W17x17 assembly contained IFBA and WABA absorbers and the CE 16x16 assembly contained either integral Gadolinium or no absorbers. Despite the differences in physical dimensions and absorber types, the conclusions reached in the EPRI Sensitivity report were confirmed for both fuel types. The bounding cases for the generic conclusions were typically based on fresh fuel with no absorbers.

   Therefore, the WABA specification has minimal impact on results (although it might have a minor impact) and no impact on the generic conclusions provided in the study.

7. *Section 3 discusses a sensitivity study regarding the impact of an in-core flux monitor in fuel assembly instrument tubes. The last paragraph in this section provides two different recommendations. Revise this paragraph to be consistent with the recommendation in the Executive Summary.*

   Make last paragraph in Section 3 and Executive Summary consistent.

   Response:

   The last paragraph of Section 3 of EPRI Sensitivity report will be revised as:

   > "As shown in the figure and tables, the change in reactivity increases with increasing burnup. Although the reactivity effect is small, non-negligible reactivities (greater than 50 pcm for some cases) have been found for a subset of the cases. Therefore, it is recommended that instrument thimbles should be included in future analysis and modeled as a void in the calculations."

8. *The following questions address the studies presented in Section 4:*
   a. *In Section 4.1 it was noted that the guide tube tolerances were increased by what appears to be an arbitrary factor of 4.1 for W17x17 fuel and 2.3 for CE16x16 fuel. How were these values determined? Was this factor applied anywhere else in the analysis?*

   Response:

   These values were chosen to increase the tolerance to the maximum physically realistic value to facilitate determination of the reactivity effect of the tolerance and distinguish the reactivity effect of the tolerance from the combination of the Monte Carlo uncertainties.

   b. *How do these tolerances and materials compare with those from all other PWR assembly designs and manufacturers?*

   Response:

   These tolerances are representative of other fuel designs. However even if the tolerance were twice as large (and the reactivity effect was also twice as large as shown in Table 4-1 and 4-2), the reactivity effect would produce a negligible impact on the overall uncertainties. The material is Zirconium, which is representative of commonly used cladding material.

   How were the k_{eff} uncertainties calculated from the sensitivity results? The uncertainty estimate should be accurate or conservative. For Monte Carlo style calculations, a conservative estimate would be calculated as $\sigma = |k_1 - k_2| + 2*(\sigma_{k1}^2+\sigma_{k2}^2)^{0.5}$. The analyst may reduce the conservatism by running more neutron histories to reduce the Monte Carlo uncertainties.

   Response:

   The calculations were performed as indicated in RAI. The computed results were divided by the factor of 4.1 and 2.3 respectively to account for the increase in the tolerance.

9. *The first paragraph in Section 4.1 identifies the tolerances as "typical" and implies that the CE guide tube tolerance is from Reference 15. A review of Reference 15 shows that this tolerance is not from Reference 15. The CE guide tube tolerance appears to be an arbitrary increase for the larger CE guide tube. Provide a better justification for this tolerance. Are these absolute tolerances, 95/95 uncertainties, or one-standard deviation uncertainties?*

   Response:

   The guide tube tolerance is taken from Reference 15, Section 7.1.3 of Attachment 5 (page 13 of Attachment 5 of Reference 5; page 48 of Reference 5).

10. *Section 4.2 addresses the fuel cladding inner diameter manufacturing tolerance. The following questions pertain to this section:*

   a. *The second sentence states "Zirconium was selected as the base cladding material because of its low cross section." Explain why using the material with the lowest cross section is appropriate. Considering that the analyst is attempting to show that the tolerance on the inner diameter yields negligible impact on k_{eff}, it seems more appropriate to use the material with the largest cross section.*

   Response:

   The second sentence reflects the historical decision by fuel vendors to use zirconium as the cladding material due to its low cross section. Subsequently, to reflect reality, Zirconium was used to represent the cladding material for the sensitivity studies in this Section. Hence, no change is made in the analyses; however, the report is updated to clarify this statement.

   b. *Note that, since this analysis does not include consideration of a flooded pellet-clad gap, the conclusion is not applicable to systems with flooded pellet-clad gaps. This limitation needs to be reflected in the guidance.*

   Response:

   This limitation is already addressed in NEI 12-16 in Section 5. See also response to RAI 24 for NEI 12-16 RAIs.

   c. *No reference is provided for the CE clad inner diameter tolerance. Reference 15 states a tolerance on the OD and has a minimum clad thickness, but does not state a maximum clad thickness. Consequently, it is not clear that using 0.002 in. for the clad inner diameter tolerance is appropriate. Provide justification for the tolerance value used.*

   Response:

   For CE fuel, the tolerance value for clad inner diameter was not available in open sources. However, since the CE fuel pin diameter is slightly larger than W17x7 fuel, a tolerance of 0.002 in (instead of 0.0015 for W17x17) was used. As evident from the values provided in Tables 4-3 and 4-2, the impact is negligible.

11. *Section 5.1 discusses the impact of modeling the fuel rod spacer grids. The following questions are provided regarding this section:*

    a. *Provide a detailed description of, and justification for, how the grids were modeled. Address the following issues: grid modeling during depletion, grid modeling in criticality calculations (i.e., smeared or modeled locally), modeling of burnup depression under the grids, etc.*

    Response:

    The grids were modelled in the criticality analysis but not in the depletion analysis. The modeling for burnup depression under the grids was not included as the study did not include the axial burnup distribution. As discussed in Section 5.1 of the EPRI Sensitivity report, the grids were modeled as void. It was recognized this was a conservative approach but due to low cross section of zirconium, the impact of conservatism is expected to be small. The grid volume is selected to be 2% of the volume of the water between the pins in the assembly. One example to justify the grid volume and modeling approach is the recent Millstone application (Millstone Unit 2 – ML12362A392).

    b. *The text notes that the "selected" grid volume is 2% of the volume of water between the pins in the assembly. Provide a reference for this value or describe and justify the way in which it was derived. Is the 2% of volume an assembly average value or a local average at the grid locations?*

    Response:

    The 2% volume was determined to be a representative bounding value based on a review of several grid designs. In order to avoid performing extensive calculations to determine the reactivity impact of fuel tolerances and the fuel assembly grid, the analyst can choose to reserve 50 ppm as margin. If this path is chosen, then the analyst should easily be able to confirm that the total grid volume of fuel assembly designs in their spent fuel pool are less than 2% of the total volume based on the data available to them.

    c. *The text goes on to say the grid volume assumption can be checked against proprietary data available to licensees. To avoid confusion and misinterpretation, provide details concerning how a licensee is to check for consistency with the 2% grid volume assumption.*

    Response:

    The licensee can check for the accuracy of 2% by comparing the area of the grid over the area of the water at the axial grid height location for the fuel lattice. The confirmation that the grid volume is less than 2% of the total volume of water is a straightforward volume calculation.

    d. *Starting on page 5-6, the report presents analysis of the soluble boron worth for the various Region 1 and 2 configurations. The soluble boron worth was used to quantify the worth of the grids in terms of soluble boron. From the text, it looks like the soluble boron worth is calculated based on change in k_{eff} values from 1700 to 2000 ppm of soluble boron. The resulting boron worth is at something closer to 1850 ppm rather than 2000 ppm. The soluble boron worth at 2000 ppm will be lower than the calculated values provided in Tables 5-5 through 5-7. Thus the maximum grid worth of 18 ppm presented on page 5-6 is likely too low. The maximum grid*

worth should also be increased to include uncertainty in the way it was calculated. Review and update the soluble boron worth and grid worth analysis to accurately describe the boron worth and to include consideration of uncertainties associated with estimating soluble boron and grid worth.

Response:

The grid calculations described in Section 5.1 were performed in a simplified manner (2D depletion calculations, axially uniform burnup distribution, smeared volume, etc.) to gain an estimate of the order of magnitude of the reactivity effect of neglecting the grid. To account for these modeling simplifications, a conservative adder was included in the final recommendation to reserve an additional 50 ppm to account for neglecting the grid spacer and using non-borated uncertainties in the borated case.

12. *The second paragraph in Section 5.2.1 asserts that validation uncertainty does not vary with soluble boron concentration. Sets of critical experiments used in validation studies supporting criticality analyses that credit soluble boron need to include a significant number of configurations with soluble boron. The analysis of the validation results is to include analysis of trends as a function of soluble boron concentration. Computational method validation, including validation of major credited features such as soluble boron, is not optional.*

    Revise the text to make it clear that validation, including validation of credited soluble boron, is required.

    Response:

    The purpose of this paragraph was to point to the fact that historically the validation uncertainty does not change significantly with soluble boron content, i.e., there is no statistically significant trend with boron content.

    For better clarification, the second paragraph in Section 5.2.1 of Sensitivity report will be revised as following:

    > "The validation uncertainty generally does not depend on the soluble boron level. In most validation suites, soluble boron cases are included. The soluble boron worth has been well predicted when using ENDF/B-V to ENDF/B-VII. When critical experiments are grouped into separate sets with and without soluble boron, the validation uncertainty of the two sets are essentially the same (i.e., there is no significant trend associated with soluble boron content)."

    The need for the code validation to include experiments with soluble boron is already included in NEI 12-16 in Section A.1.1.

13. *Section 7 presents sensitivity studies performed to show the impact of increasing spacing between assemblies. Historically, 12 inches (~30 cm) has been used as the separation distance to ensure commercial reactor fuel is neutronically decoupled. The section concludes that a shorter distance of 25 cm could be used to ensure the fuel groups are neutronically decoupled and suggests that under some circumstances an even shorter distance may be justifiable. With respect to the neutronic decoupling sensitivity study, provide the following:*

    a. *Justification for using vacuum boundary conditions.*

    Response:
    Vacuum boundary condition was used to simply isolate the impact of separation for analysis purposes.

    b. *The model details, especially the moderator temperature and density used. Justify the details and describe how departures from the specific analysis affect the conclusions.*

    Response:

    Computations for spacing were performed at base temperature and densities (293 K, density 1 g/cm^3).

    c. *It is unclear whether anything less than 25 cm is being recommended. Clarify whether anything less than 25 cm is being recommended. If anything less than 25 cm is being recommended, provide supporting analysis for that recommendation.*

    Response:

    As indicated in the last two paragraphs of Section 7:

    > "As is evident from the data presented in both tables and Figure 7-4, a gap of 25 cm (~9.84 in) of unborated water is sufficient for assemblies to effectively decouple neutronically from each other. If there are neutron absorber panels, as in cases 5 to 7 in Table 7-2; then, the decoupling of assemblies is achieved with less separation. This is due to the absorption of neutrons in neutron absorber panels after full thermalization in the water gap. However, as seen on Table 7-2, the rate of decoupling is fairly independent of the enrichment or burnup of the fuel.
    >
    > Considering that the cell pitch in PWR racks varies from 20 cm (~7.874 in) to 25 cm (~9.843 in), it can be concluded that an empty row between two cells decouples them from adjacent areas of rack. Therefore, analysis to determine if increased reactivity occurs at an interface is not needed when there is 25 cm of water between racks or an empty row of cells."

    The final recommendation is that **a separation of 25 cm (~10 in) is sufficient for neutronic decoupling of assemblies**, as indicated in the Executive Summary and Conclusions sections. Therefore, interface analysis is not needed when a PWR rack has an empty row of cells between regions. Spacing less than 25 cm applies when neutron absorber panels are present.

14. *The analysis, discussion, and conclusions appear to have no relevance to performing 10 CFR 50.68-compliant analyses. The NRC staff has stated that a relatively small volume of fuel can drive the k_{eff} for the entire SFP. As a result, it is inappropriate to assume average parameter values to claim additional margin in the safety analyses. However, the information presented in Section 8 only appears to be relevant to a situation where small numbers of non-compliant fuel is loaded in an empty SFP, which would be an extremely atypical situation. Revise the text to clarify the intent of this study and where the conclusions would be applicable.*

    Response:

    During the series of four meetings between September 2013 and February 2014, NRC staff stated that a relatively small number of fuel assemblies can drive the k_{eff} for the entire SFP analysis. However, there was also uncertainty as to actual number of fuel assemblies (i.e., 4 versus 10 versus 20) that are needed to produce a reactivity equivalent to an infinite array. The study in Section 8 was performed to determine how many fuel assemblies are needed (with radial leakage) to produce a reactivity equivalent to the infinite array cases that are modelled in SFP criticality analyses.

    The results presented in Table 8-1 through 8-3 demonstrate that a significant number of fuel assemblies are needed to produce a reactivity equivalent to the infinite array cases. In particular, Tables 8-1 and 8-2 show that even an 8x8 model with 64 assemblies have a reactivity at least 1% lower than the infinite array case.

    The statement that Section 8 is "relevant to a situation where small numbers of non-compliant fuel is loaded in an empty SFP, which would be an extremely atypical situation" is not correct. The purpose of the study was to investigate the difference in reactivity between an infinite versus finite array, not to investigate a small number of non-compliant fuel assemblies.

16. The following questions are provided on Section 10, "Impact of Concrete Composition on Reactivity":

    a. *The studies performed are mainly based on the four concrete compositions included in the SCALE libraries. There does not appear to be any effort to determine if the final "conservative" concrete composition is, in fact, conservative relative to a variety of real-world concrete compositions. Include some discussion of applicability to concrete from different geographic regions of the country, given their varying aggregates.*

    Response:

    As described in the report, the conservative concrete composition was determined by taking the maximum content of elements that produced the highest positive reactivity and minimum amount of elements that produce a negative reactivity effect, based on the computed results provided in Table 10-4. The elemental composition for two conservative concretes, wet and dry concretes, are presented in Table 10-5. Then, the impact of the concrete was determined by comparing the reactivity against "reference", Rocky Flat. As evident from the values provided in Table 10-6, dry concrete produces the maximum impact on reactivity. In reality, independent of the region, any concrete composition would be less conservative compared to the composition proposed in this study since it will contain some hydrogen, which will reduce the reactivity.

    b. *Include a statement that the scope of the study did not include low-moderator density optimal moderation conditions frequently evaluated for new fuel storage racks.*

    Response:

    The following statement will be added to Section 10 of the Sensitivity report:

    > "The proposed concrete composition is provided for spent fuel pools and does not include new fuel vaults, which typically contains low-moderator density concrete."

    c. *The final conclusion concerning use of the conservative concrete is overly broad. The study did not consider use of high density concretes or concretes used in new fuel storage rooms/vaults. Revise the conclusion to reflect the limitations of the study.*

Response:

Section 10 included a study of the impact of concrete composition in a large spent fuel rack model that is analogous to the configuration of a spent fuel pool. This study reached the following conclusion:

> "Initially, a 400-cell rack array surrounded by concrete was modeled. In the model, neutron absorber panels for the last two rows were removed. This large model showed that the impact of concrete composition on reactivity is negligible as the variations were within the Monte Carlo uncertainties."

The analysis performed with the smaller 4x4 model was performed specifically to exacerbate the difference in reactivity between the different concrete compositions, so that a bounding concrete composition could be determined, however this does not invalidate the previous conclusion quoted above.

As stated in the previous response, a statement for new fuel vault will be added to the report.

17. *The following questions are related to the text in Section 11, which is on "Impact of Pool Temperature on Reactivity:"*

    a. *Revise the 2nd sentence in the final paragraph to "The Doppler feedback in the fuel is always negative."*

    b. *Revise or delete the 3rd sentence in the final paragraph. No basis has been presented for the assertion that the two effects could not provide a local maximum.*

    c. *The 4th sentence in the final paragraph states, "There is no indication that the change in reactivity is not monotonic." However, the Region 2 curve in Figure 11-1 shows the reactivity dropping from 277 to 290K and then rising. This is indication that under some circumstances the change in reactivity is not monotonic. Provide additional analysis that fully addresses the potential impact of pool temperature on reactivity and revise the report as appropriate.*

Response (a, b, c):

Agreed. The last paragraph is revised as following:

> "For all of the cases analyzed, the highest reactivity is produced at the highest or lowest temperature. The Doppler feedback in the fuel is always negative. If the water reactivity change is positive, the two effects could cause a local minimum in reactivity. As evident from Figure 11-1, the behavior can deviate from monotonic increase. Hence, it is recommended to compute reactivity at highest, lowest, and intermediate temperatures to capture full behavior."

The recommendation in Executive Summary and Conclusions (Section 13) of the Sensitivity report will be revised to reflect the need to compute the reactivity at intermediate temperatures.

This issue will also be addressed in NEI 12-16 in Section 5.2.2.1.

18. *Section 12 is titled "Impact of Gadolinium Burnable Absorbers on Spent Fuel Reactivity." The gadolinium and erbium work reported in NUREG/CR-6760 and, apparently, in the gadolinium work in the EPRI report was performed using only two-dimensional (2D) models in reactor geometry. A review of Section 3.3.5.5 in NUREG/CR-6760 shows that three-dimensional (3D) modeling of IFBA fuel in a cask geometry can result in a positive reactivity effects that were not seen in the 2D calculations. This reviewer is not aware of any calculations that have been performed with gadolinium or erbium using real 3D in-rack (or in cask) models and part-length integral absorber sections. Three-dimensional sensitivity studies (i.e., real axial burnup and part-length absorber sections) are needed to support a generic conclusion that it is conservative to ignore gadolinium and erbium.*

    Provide supplementary sensitivity studies or eliminate the conclusions stating that it is conservative to ignore gadolinium and erbium in fuel rods.

    Response:

    There are fundamental differences between IFBA and gadolinium/erbium that explain the results presented in Section 3.3.5.5 of NUREG/CR-6760. This is best summarized by the conclusions in Section 3.3.5.2:

    > "These analyses show that there is a negative residual effect for gadolinia-bearing fuel but no such effect for fuel designs with IFBA rods. The main difference between the UO2-Gd2O3 rods and the IFBA rods is that gadolinium is an integral part of the fuel matrix in the UO2-Gd2O3 rod while the boron is placed as a thin coating on the outer surface of the fuel pellets in the IFBA rod. The gadolinium isotopes displace uranium in the UO2-Gd2O3 rod, resulting in reduced reactivity (due to the reduction in heavy metal mass). The IFBA coating does not displace uranium, and thus, there is no negative residual effect."

    The assessment in the RAI that "A review of Section 3.3.5.5 in NUREG/CR-6760 shows that three-dimensional (3D) modeling of IFBA fuel in a cask geometry can result in a positive reactivity effects that were not seen in the 2D calculations" is not accurate. The results contained in Table 10 versus Table 11 of NUREG/CR-6760 compare the results of two separate 3D models, one where the IFBA rods are modelled along the entire active fuel length, the second where the IFBA is modeled shorter than the active fuel length (and more realistically). The non-conservative results presented in Table 10 (compared to the results in Table 11) are from crediting residual ^{10}B in the IFBA that is not actually present in the fuel assembly.

    Additionally, the first sentence in Section 3.3.5.2 of NUREG/CR-6760 states:

    > "Sections 3.3.1 and 3.3.2 [which are 2D analysis] demonstrate that the Δk values become positive for fuel assembly designs containing IFBA rods but remain negative for gadolinia-bearing fuel assembly designs."

    The 2D analysis contained in Sections 3.3.1 for IFBA assemblies identified the additional positive reactivity effect from the harder spectrum compared to fuel assemblies without IFBA. The 2D analysis in Section 3.3.2 also confirmed that when burnable absorbers are integrated with the fuel

(i.e., Gadolinium and Erbium) they produce a lower reactivity versus when the burnable absorber is neglected. 3D calculations with axial distributions of Gadolinium are not necessary, because the two-dimensional models are always conservative when Gadolinium is neglected (This was not the case for the 2D calculations for fuel assemblies with IFBA rods).

19. *Section 13 provides a summary of the report and some of the conclusions. Questions on this section are provided below:*

    a. *The first of the key conclusions listed is related to modeling "instrument thimbles." This conclusion should be modified to make it clear that the instrument thimble is not the same thing as the instrument tube.*

Response:

The first bullet of conclusion is changed to:

> "The impact of the in-core instrumentation thimbles on reactivity is small but not negligible (greater than 50 pcm for some cases). It is recommended that the instrument thimble be included in future analysis and modeled as a void in the calculations."

b. *Revise the conclusions stated in Section 13 to reflect any limitations associated with the limited scope of the sensitivity studies upon which the conclusions are based.*

Response:

As agreed during the four meeting, the conclusions from the sensitivity studies are valid for current PWR fuel types, rack types and configurations.

c. *The 3rd bullet starts with the sentence: "The limiting condition in the criticality safety analysis is the unborated condition." Generally, this is true. However this not necessarily always true. The absolute nature of this statement in the text does not seem to be important to the conclusion. Revise the statement to include a "usually" or "typically" in an appropriate place.*

Response:

The first sentence of 3rd bullet in the Conclusion section (Section 13) of EPRI Sensitivity report is revised as following:

> "Typically, the limiting condition in the criticality safety analysis is the unborated condition."

d. *The 6th bullet discusses the conclusion related to the study of small sets of assemblies. As discussed in RAI #13, this study does not appear to address any issues that are of relevance in the context of compliance with criticality safety regulatory requirements. Therefore, remove this conclusion or clarify how an applicant would use this conclusion.*

Response:

It appears that the RAI should reference RAI-14 (not RAI-13), which is related to the analysis of finite versus infinite arrays of fuel assemblies. See the response to RAI-14.

20. *The NRC staff recommends NEI incorporate the following editorial corrections:*

  a. *Change the reference to "10 CFR 52.157(a)(8)" in the 6th bullet in Section 1.3 to "10 CFR 52.157(1)(8)."*

  a. *In the 7th bullet in Section 1.3, change the reference to NUREG-0800, Section 9.1.1, Revision 4 to Revision 3.*

  b. *In the 3th bullet in Section 1.3, change the reference to NUREG-0800, Section 9.1.2, Revision 3 to Revision 4.*

  c. *The final sentence in the second paragraph of Section 4.2.3, PWR Depletion Uncertainty, states, "Therefore, no other uncertainties are needed to be applied to the calculation of the maximum keff." This statement appears to be unintentionally broad. Clearly, there many other uncertainties that must be applied to the calculation of the maximum ke11. Revise or remove this statement and all other overly broad statements.*

  c. *The first sentence in Section 5.2.2.2 states that modeling of rack dimensions is described in Section 4.3. Section 4.3 is "Peak Reactivity Analysis for BWRs." This text needs to be eliminated or revised to refer to an appropriate reference.*

  d. *Reference 6 - The title should be Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.*

  e. *Reference 10 - Revise to note that the ANSI/ANS-8.24 was reaffirmed in 2012.*

  f. *Reference 15 - The lead author for this reference is J.M. Scaglione.*

  g. *Reference 16 - Include authors G. Ilas and J.C. Wagner or revise to G. Radulescu et al.*

  h. *Reference 18 - Revise title from " ... on PWR ... " to " ... for PWR ... "*

  i. *Reference 21 - Revise the title from " ... Nuclear Fuel Confirmation" to " ... Nuclear Fuel Burnup Confirmation." Include the ORNL report number, which is ORNL/TM-2007/229.*

  j. *Reference 23 - Revise the title from "Human Reliability ... " to "Preliminary, Qualitative Human Reliability ... ."*

  k. *Reference 29 - Include the author, which is K. Lindquist.*

  l. *Reference 31 - Update this Reference to D. Lancaster, Sensitivity Analyses for Spent Fuel Pool Criticality, EPRI Report 3002003073, EPRI, Palo Alto, CA, December 2014.*

  m. *Reference 33 - Revise to list or indicate that there were coauthors.*

Response:

The editorial corrections identified have been incorporated into NEI 12-16, Revision2, with the exception of sub-items k) and l). All references to EPRI reports were updated to be consistent with EPRI recommendations for citation of their technical reports. This information is typically contained on the acknowledgement page at the beginning of EPRI reports.

NEI 12-16, Revision 2 - DRAFT A

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

May 2016

NEI 12-16, Revision 2 - DRAFT A

Nuclear Energy Institute

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

May 2016

ACKNOWLEDGEMENTS

This guidance was developed by the NEI Criticality Task Force. We also recognize the direct participation of the licensees and vendors who contributed to the development of the guidance. The dedicated and timely effort of the many participants, including management support of the effort, is greatly appreciated. Finally, we would like to thank the U.S. Nuclear Regulatory Commission for providing feedback during the series of public meetings between September 2013 and April 2016. This guidance has been updated to incorporate NRC feedback.

NOTICE

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

FOREWORD

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

Criticality requirements for the spent fuel pool of nuclear power plants are found in 10 CFR 50.68 or 10 CFR 70.24. Guidance for performing criticality analyses in compliance with these regulations 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 document is developed as a comprehensive guide that presents an acceptable approach to comply with the regulations upon NRC endorsement. Individual vendors or licensees can deviate from the method supplied herein, with appropriate justification and approval by the NRC.

TABLE OF CONTENTS

ABBREVIATIONS AND ACRONYMS

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

1 INTRODUCTION

1.1 PURPOSE

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

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

1.2 BACKGROUND

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

The first guidance on acceptable methods for performing criticality analyses at LWR plants, following promulgation of 10 CFR 50.68 [1], was issued in 1998 through an NRC internal memorandum from L. Kopp to T. Collins, often referred to as the "Kopp Memorandum" [24]. Although this was an internal NRC memorandum, it was quickly adopted by industry for use in performing criticality analyses, referenced in LARs, and referred to by NRC staff in the Safety Evaluation Reports for the associated license amendments due to the lack of formal guidance. The guidance in the Kopp Memorandum provided regulatory clarity and stability for many years. In 2010, the NRC issued an action plan to develop new interim staff review guidance followed by a durable Regulatory Guide that would replace the Kopp Memorandum and better reflect the staff positions on acceptable criticality analysis methods that evolved through interactions with licensees since 2005.

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

1.3 APPLICABLE REGULATIONS

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

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 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(f)(8), "Contents of applications; technical information in final safety analysis report." [4]

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

- NUREG-0800, Standard Review Plan, Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," Revision 3. [11]

- NUREG-0800, Standard Review Plan, Section 9.1.2, "New and Spent Fuel Storage," Revision 4. [12]

1.4 DOUBLE CONTINGENCY PRINCIPLE

The double contingency principle [9] states, "process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." In other words, the nuclear criticality analysis is required to demonstrate that criticality cannot occur without at least two unlikely, independent and concurrent incidents or abnormal occurrences. This will ensure that no single occurrence can lead to an inadvertent criticality event. The double contingency principle means that a realistic condition may be assumed for the criticality analysis when calculating the effects

of incidents or abnormal occurrences. When applying the double contingency principle, the chosen conditions need to be independent from one another (i.e. do not result from a common initiator) and are unlikely (i.e. low probability) to occur. For example, for PWRs, the loss of soluble boron below the minimum Technical Specification requirement is considered as one accident condition and a second concurrent accident need not be assumed (e.g., such as a fuel assembly misloading or misplacement). Therefore, compliance with the Technical Specifications minimum required soluble boron concentration may be credited when evaluating other accident conditions.

1.5 USE OF PRECEDENCE

The use of precedence (i.e., adopting methods or conclusions previously approved in another application, but not necessarily documented in a generic regulatory document) is a well-established principle by the NRC in the process of reviewing applications. The use of 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 or differences should be technically supported and demonstrated as appropriate. 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

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

The licensee assumptions used in the criticality analysis should be explicitly identified and clearly stated. 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 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 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 incorporate risk insights as part of a "graded" licensing approach and is acceptable as long as the assessments consider relevant safety margins and defense-in-depth attributes. For example, a criticality analysis that demonstrates a maximum k_{eff} with a relatively large margin to the regulatory k_{eff} limit, may be permitted to make more

assumptions about results or uncertainties than a criticality analysis that demonstrates a maximum k_{eff} with a relatively small margin to the regulatory k_{eff} limit.

2 ACCEPTANCE CRITERIA

Fresh (New) Fuel Storage

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

a) With the new fuel storage racks assumed to be loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must not exceed 0.95, at a 95 percent probability, 95 percent confidence level (10CFR 50.68(b)(2)).

b) With the new fuel storage racks loaded with fresh fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid corresponding to optimum moderation, the k_{eff} must not exceed 0.98, at a 95 percent probability, 95 percent confidence level (10CFR 50.68(b)(3)).

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

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:

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

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

   a. With the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water, the k_{eff} must remain below 1.0

(subcritical), at a 95-percent probability, 95 percent confidence level (10CFR 50.68(b)(4)).

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

3 COMPUTER CODES

3.1 TYPES AND USES OF COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent the important neutronic and geometrical aspects of the storage racks. In spent fuel pool criticality safety analyses, there are two general types of computer codes that are used. These are criticality codes and depletion codes.

The criticality codes are used to determine the eigenvalue (k_{eff}) of the analyzed system. The isotopic concentrations of the materials in the system are determined from manufacturing data and depletion analysis. The Monte Carlo method relies on repeated random sampling to compute the answer. Cross sections are used as probabilities of interaction and the Monte Carlo code then calculates and tracks individual neutron lifecycles. Although many criticality codes utilize Monte Carlo methods, there are other criticality codes that provide acceptable results utilizing deterministic transport methods.

The convergence of the Monte-Carlo criticality code is sensitive to various parameters, including:

- the number of neutrons per generation,
- number of generations skipped prior to averaging,
- the total number of generations, and
- the initial source distribution.

The choice of these parameters is intended to optimize calculational accuracy and computer processing time. The initial source distribution should be specified appropriately for the type of code being utilized and geometric configuration(s) being analyzed.

The Monte Carlo uncertainty is a result of the parameters selected above. There is no stipulation or requirement specified on the magnitude of the resultant Monte Carlo uncertainty, as it is incorporated into the overall calculational result with other uncertainties.

A description of the criteria for determining acceptability of convergence should be included as part of the application.

The depletion codes are used to calculate the nuclide density changes that occur in the fuel during operation in the reactor core. In addition, decay changes in nuclide concentrations due to

non-power cooling times are also captured in depletion calculations. In general, depletion codes utilize deterministic transport methods in lieu of Monte Carlo methods. However, Monte Carlo methods may also be used for depletion calculations.

The codes to perform depletion and criticality calculations rely upon the use of cross-section libraries. Cross-section libraries used in the criticality analysis should be widely accepted and peer reviewed. Cross-section libraries that have previously been found acceptable for use include the multi-group and continuous energy ENDF/B 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. It is highly recommended that the code version and cross section set used in the analysis be the same as those used in the validation of the codes for simplicity and reduction of calculational burden.

Applicants may credit all actinides and fission products available in the criticality analysis code and depletion code used in the application. Credit for all actinides and fission products is based upon the discussion in Section A.2.1 and prior NRC approval of this approach on previous submittals to the NRC over the last two decades. It is recognized that applicants would need to account for an additional bias associated with fission product and minor actinide worth as recommended in NUREG/CR-7109. The one exception is the exclusion of ^{135}Xe as discussed in the Cooling Time discussion of Section 4.2.1.

3.2 Computer Code Validation

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

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

4 REACTIVITY EFFECTS OF DEPLETION

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

4.1 Depletion Models

Historically, depletion models consisted of a model to produce one-group cross sections followed by a solution of the isotopic production and loss equations. The one-group cross

sections were produced using the flux from an infinite model of pin cells. Although this approach produced good results, modern nodal methods used in core reload design use a two-dimensional lattice model which determines the one group fluxes used in the isotopic production and loss analysis. Separate lattice models are developed for each unique axial plane, such as low enrichment blankets, control rods insertion, and burnable absorbers.

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

4.2 REACTIVITY EFFECTS OF DEPLETION FOR PWRS

The most significant parameters that could impact reactivity of used fuel in depletion analyses for PWRs are:

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

Additional guidance in selecting operating parameters for depletion analysis is provided in NUREG/CR-6665 [17]

4.2.1 Depletion Analysis

Power, Moderator Temperature and Fuel Temperature during Depletion

The power density, fuel temperature and moderator temperature (and associated moderator density) are grouped together because of the unique inter-relationship between these three values during in-reactor fuel depletion. The power density and moderator flow rate of a fuel assembly during depletion will directly impact the moderator and fuel temperature with a higher power (and/or lower moderator flow) resulting in higher moderator and fuel temperatures. Higher moderator and fuel temperatures during depletion result in increased reactivity of used fuel in the storage rack. While a higher power will lead to a higher ^{149}Sm content after the decay of ^{149}Pm, which lowers reactivity, this effect is much smaller than the impact of the moderator and fuel temperature. Therefore, depletion at credible but high power, moderator temperature, and fuel temperature is typically conservative. Previous studies [17] have also identified a small reactivity impact due to power history, with a low power coast down providing a conservative end of life reactivity. If load follow (variation of reactor power to adjust to demand) is exercised, this should be evaluated against the high constant power assumption.

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

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

Soluble Boron during Depletion

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

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

A licensee would evaluate a mid-cycle offload in accordance with the licensee's corrective action program and current NRC guidance for identifying and resolving potential non-conservatisms or unanalyzed conditions in a design basis analysis. If an issue is identified, the licensee would make an initial operability determination, and subsequently evaluate in accordance with 50.59 to determine whether NRC approval is required. As a default, any fuel assembly could be conservatively treated as a fresh fuel assembly.

Burnable Absorbers

PWR reactors use a variety of burnable absorbers during operation for the purposes of reactivity control and power distribution control. These absorbers can be mixed into the fuel pellet (e.g., Gadolinium, Erbium, etc.), added as a coating on the fuel pellet (ZrB_2 IFBA) or be included as inserts in the guide tubes (e.g., WABA, BPRA, Pyrex, etc.). For PWR fuel with Gadolinium and Erbium absorbers, it is conservative to neglect the presence of the burnable absorber. Therefore, it is recommended that the impact of absorbers integrated into the fuel pellet such as Gadolinium and Erbium be neglected in the criticality analysis.

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

Burnable absorbers harden the energy spectrum during operation due to the presence of the neutron absorber (i.e., absorption of thermal neutrons) and the displacement of water from the guide tubes. The reactivity effect on the fuel assembly is a function of the duration of the

removable absorber in the fuel assembly (determined through the amount of burnup the fuel assembly experiences while the burnable absorber is present). Therefore, the maximum burnup that a fuel assembly receives while containing a burnable absorber must be determined and used in the analysis.

Studies have shown that Gadolinium and Erbium burnable absorbers can be conservatively neglected [18]. The residual content of Gadolinium and Erbium and the displacement of fissile material (UO_2) has more negative reactivity worth than the positive worth due to harder spectrum depletion, regardless of the burnup of the fuel assembly. If Gadolinium or Erbium is to be neglected, the planar averaged enrichment may be used in the criticality model. Recent analysis confirmed that neglecting Gadolinium and Erbium burnable absorbers is a conservative approach [31].

It is also important to note that multiple absorbers, such as WABAs and IFBAs, can be present in a fuel assembly undergoing depletion in any given cycle. In the event of multiple absorbers, the depletion analysis should take into account all burnable absorbers present in the fuel assembly.

Further, other moderator displacing inserts should be addressed, such as primary and secondary sources. Normally, primary and secondary sources will be covered by the conservatism in the burnable absorber assumptions, but confirmation is necessary.

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

Part Length Burnable Absorbers

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

Rodded Operation

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

Cooling Time

The standard practice is to perform the depletion analysis at a very short cooling time (hours or days) with no ^{135}Xe to determine the spent fuel isotopic inventory after discharge. This is commonly referred to a zero cooling time and is intended to represent freshly discharged fuel. However, as the short lived fission products decay and ^{241}Pu decays to ^{241}Am, the fuel assembly reactivity continues to decline. This additional reduction of reactivity with cooling time can be credited to allow for greater flexibility in managing the spent fuel inventory. Many of the modern depletion codes can perform the change in isotopic inventory with additional time automatically.

Other Depletion Parameters

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

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

The burnup step size in the depletion calculations needs to be sufficiently small to ensure proper calculation of the isotopic inventory from burnup step to burnup step. The analyst should review the recommendation of the depletion code documentation.

It is recommended that the applicant include a summary of the core depletion parameters (operating parameters, presence of burnable absorber, etc) used in the analysis in sufficient detail. The summary should include sketches or figures and a table with dimensions and materials. This information can also serve the applicant as a guide to the inputs used in the analysis for evaluating future changes in operation.

4.2.2 Fuel Assembly Physical Changes with Depletion

During reactor operation, the fuel rods undergo small physical changes. These changes are driven by the behavior of the ceramic uranium dioxide fuel pellets as they generate energy. These may have an impact on the reactivity of the fuel in the SFP environment. The specific physical changes of concern are changes to fuel density, clad outer diameter (OD), and clad thickness. It should be noted data for fuel pellet diameter is also captured because fuel pellet diameter changes are directly correlated to fuel density changes.

The impact of fuel geometry changes on reactivity changes was analyzed in a proprietary Westinghouse study which is summarized here. The study is based on calculations performed with NRC-approved fuel performance and fuel depletion codes [38,39] for a Westinghouse 3-loop PWR core operating with a 15x15 fuel lattice. The study included both IFBA and non-IFBA fuel, modeled fuel pellets near both the center and top of the assembly, and covered a burnup range from 0 – 62 GWD/MTU. The study was divided into three major sections:

1. Modeling the physical behavior of fuel during operation using the PAD code to determine the minimum and maximum values for fuel density, clad outer diameter, and clad thickness;
2. Modeling the depletion of the fuel with PARAGON [39] using the minimum and maximum values calculated with PAD [38] to determine fuel assembly isotopics; and
3. Determining the changes in reactivity due to the physical changes in the fuel over depletion.

The change in the physical behavior of the fuel during operation is provided in Figure 4-1 through Figure 4-4. These figures are based on PAD/PARAGON calculations and represent plant-specific values, however, their importance is in the demonstration of the behavior of fuel rods with irradiation. The behaviors exhibited by the pellet and clad are not unique to a specific reactor design, but instead, are applicable to all UO_2 fueled plants. For the purpose of illustrating the pellet and clad behavior, values on the y-axis are omitted to avoid unnecessary attention on specific values.

Figures 4-1 and 4-2 show the density and diameter changes of the fuel pellet with respect to depletion. Figure 4-1 shows the pellet density initially increases to a maximum very quickly and then decreases with additional burnup. Figure 4-2 shows the corresponding pellet diameter changes with burnup. Both figures clearly demonstrate the two widely known phenomena of fuel densification and fuel swelling. Early in reactor operation the heat generated by fission causes fuel to densify and the fuel pellet diameter to correspondingly decrease. As operation continues, the fission products produced in the pellet cause the pellet diameter to expand and the fuel density to decrease. It should be noted that while the fuel density is changing, this is solely due to changes in pellet dimensions as the mass within the fuel is unchanged.

Figures 4-3 and 4-4 show the changes in cladding thickness and outer diameter due to irradiation. The behavior of these parameters align with the behavior of the fuel pellets. Initially the clad outer diameter decreases, thickening slightly, until the clad comes into contact with the fuel pellet. Once the clad and pellet come into contact, the clad expands and thins as the fuel pellet swells and causes the rod diameter to expand.

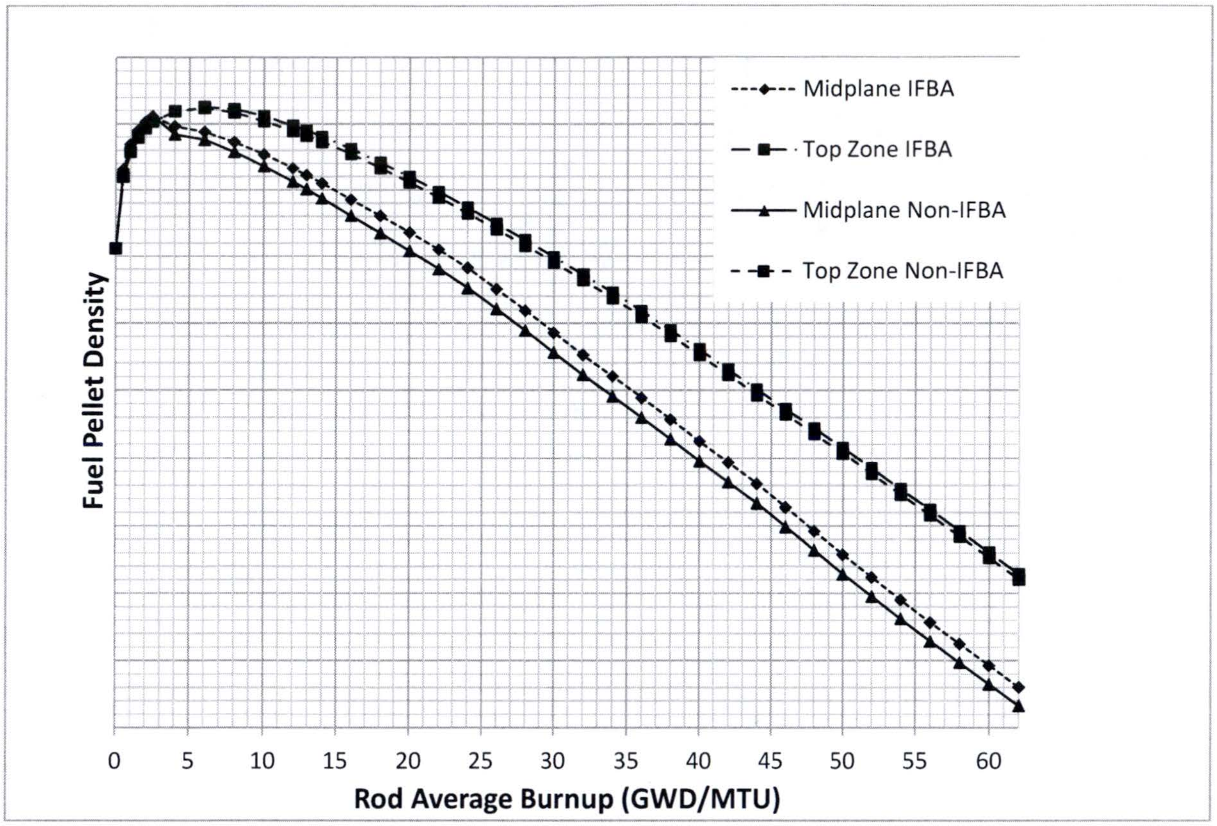


Figure 4-1: Fuel Density Behavior over Depletion

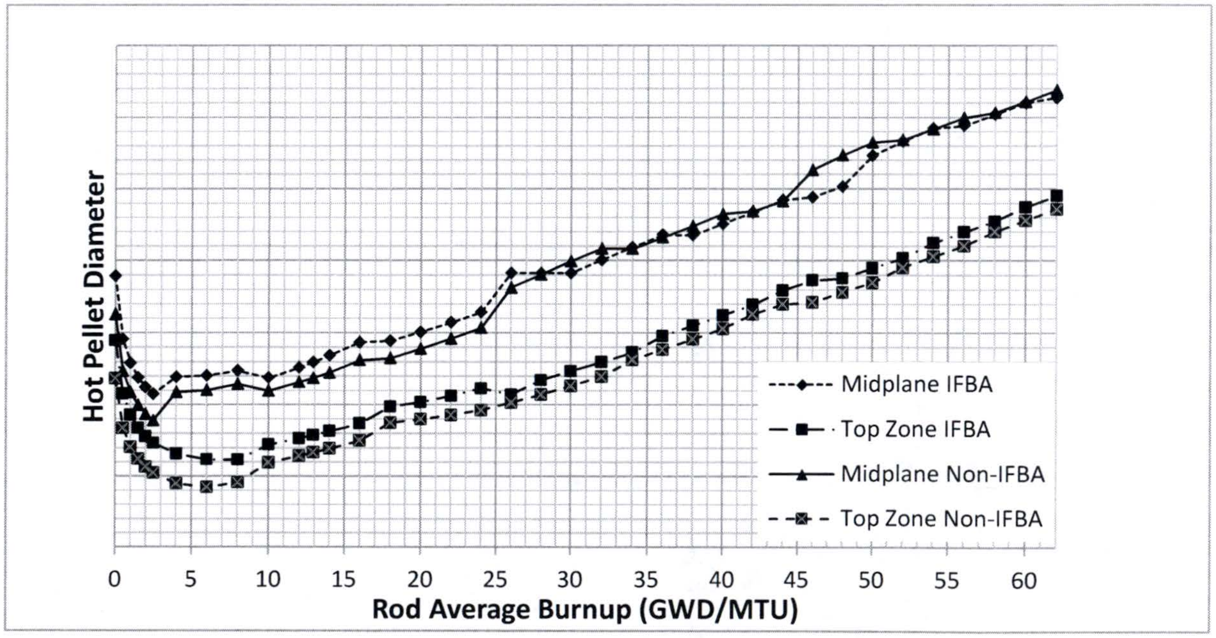


Figure 4-2: Fuel Pellet Diameter Behavior over Depletion

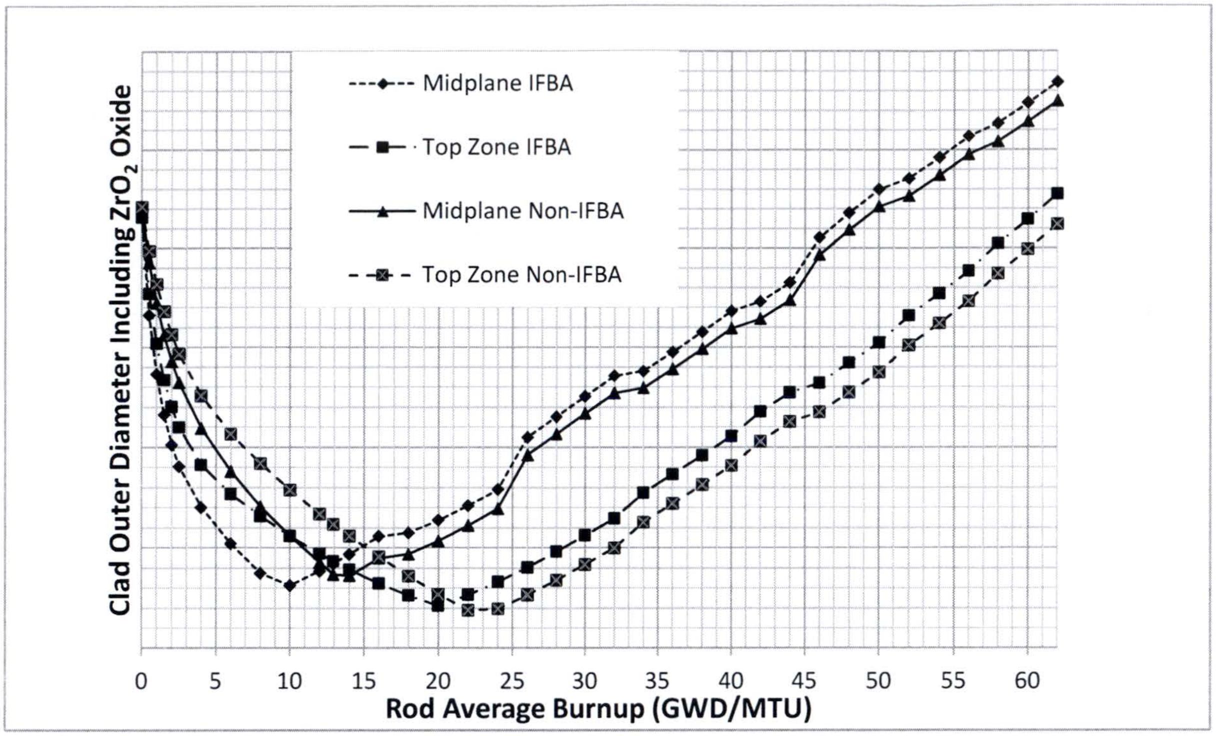


Figure 4-3: Clad Diameter Behavior over Depletion

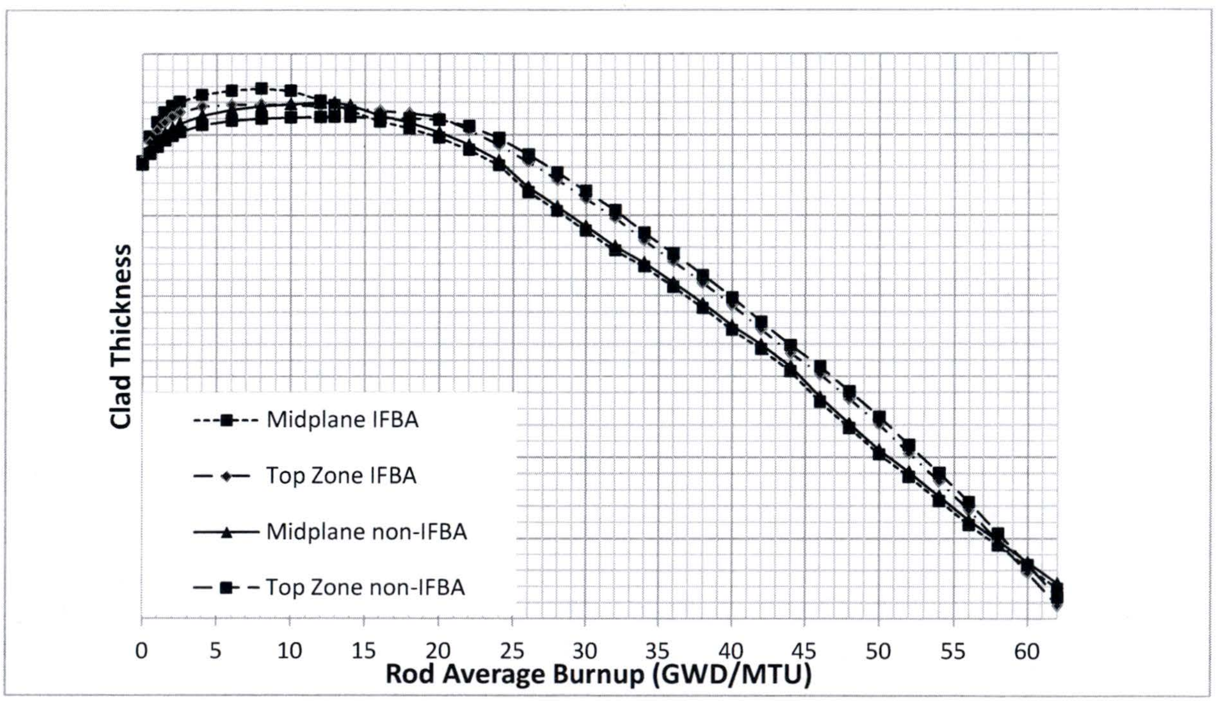


Figure 4-4: Clad Thickness Behavior over Depletion

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

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

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

To provide a better estimate of the true reactivity impact, two calculations for both IFBA and non-IFBA fuel were performed. The first condition evaluated the fuel pin geometry associated with peak fuel density and the second condition evaluated end of life conditions (62 GWD/MTU). The results of these calculations are provided in Table 4-1.

Table 4-1: Overall Reactivity Impact of Fuel Changes

Case Name	**Δk**
non-IFBA End of Life	-0.00093
non-IFBA Maximum Density	-0.00123
IFBA End of Life	-0.00040
IFBA Maximum Density	-0.00409

The results of Table 4-1 demonstrate that each individual fuel geometry change has an impact on fuel reactivity. When these changes are looked at holistically, the overall impact of fuel geometry changes with depletion is small. These results are not unexpected because it aligns with standard procedures for performing fuel management calculations and reactor operating experience. These procedures essentially ignore fuel geometry changes, which would not be the case if they had a significant role (either positively or negatively) on fuel reactivity. Based on this study and its alignment with general fuel management practices, fuel geometry changes with depletion do not need to be explicitly modeled in depletion calculations.

4.2.3 PWR Depletion Uncertainty

Historically, engineering judgment was used to estimate the uncertainty associated with fuel depletion calculations as a percentage of the change in reactivity associated with depletion [24]. Two independent evaluations [27, 33, 34] have been conducted to determine the magnitude of this uncertainty and show that the use of 5% of the reactivity decrement as an uncertainty is conservative for the cross-section sets ENDF/B-V through ENDF/B-VII . Both analyses also confirm that a zero bias is appropriate. When calculating the depletion uncertainty, the change in reactivity between the zero burnup, fresh fuel condition and the burnup of interest is determined without burnable absorbers.

For low burnup, the depletion reactivity benchmarks suggest a higher uncertainty, however this is based on limited data at low burnups. The chemical assay approach, using the direct

difference method, results in an uncertainty that is less than 5%, even at low burnups [34]. Therefore, a simple approach of using 5% of the change in reactivity associated with depletion is an acceptable method for accounting for the uncertainty associated with the depletion calculations for all burnups. Because these methods are an integral benchmark of the entire system modeled by the depletion codes it covers all uncertainties associated with depletion, such as uncertainty in computation of the isotopic inventory by the depletion code, uncertainty in cross-sections (both actinides and fission products), etc.

Licensees may use of the 5% based on the justification provided above. However, if credit is desired for a more accurate assessment of the depletion uncertainty, a licensee may use either of the approaches above and detailed in Appendix A [27, 33] to reduce the conservatism in this parameter.

4.3 Peak Reactivity Analysis for BWRs

It is standard practice that BWR spent fuel pool criticality analyses are performed at the burnup that produces the lattice peak reactivity. BWR fuel lattices that contain an integral burnable absorber typically result in a lattice peak reactivity at a specific burnup value, usually under 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 general methodology for BWR spent fuel pool criticality analyses is to perform in-core depletion calculation for the various assembly designs in use, then to restart the calculations with the assemblies in the standard cold core geometry (SCCG) and then in the storage rack geometry. The SCCG is defined as an infinite array of fuel assemblies on a 6-inch lattice spacing at 20°C, without any control rods or voids. The burnup at the limiting k_{inf} in the SCCG is determined and then the k_{inf} in the storage rack geometry is calculated at this burnup. A reactivity allowance for applicable biases and uncertainties is added to the calculated k_{inf} in the rack geometry and the resulting k_{eff} is compared to the regulatory limit of 0.95.

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

4.3.1 Depletion Parameters

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

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

- **Reactor operating parameters**:
  - Void fraction – 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.
  - 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.
  - Control Rod Usage - The SCCG is calculated uncontrolled (i.e. no control rods insertion). However, the reactivity impact of control rod usage should be accounted for separately in the criticality analysis.

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

- **Lattice specific parameters:**
  - Enrichment – Typically highest planar average U^{235} enrichment of all the bundle types being evaluated is bounding.
  - Part Length Rods – Each unique axial plane in the bundle designs being evaluated including number and location of partial length rods should be evaluated.
  - Integral Burnable Absorber Fuel Rods - Number, location and nominal concentration of integral burnable absorber fuel rods should be evaluated appropriately for the given application.
  - Nuclides modeled – Appropriate nuclides used in PWR depletion analyses or those nuclides used in BWR core design and core monitoring analyses are acceptable as described in Section 3.1.

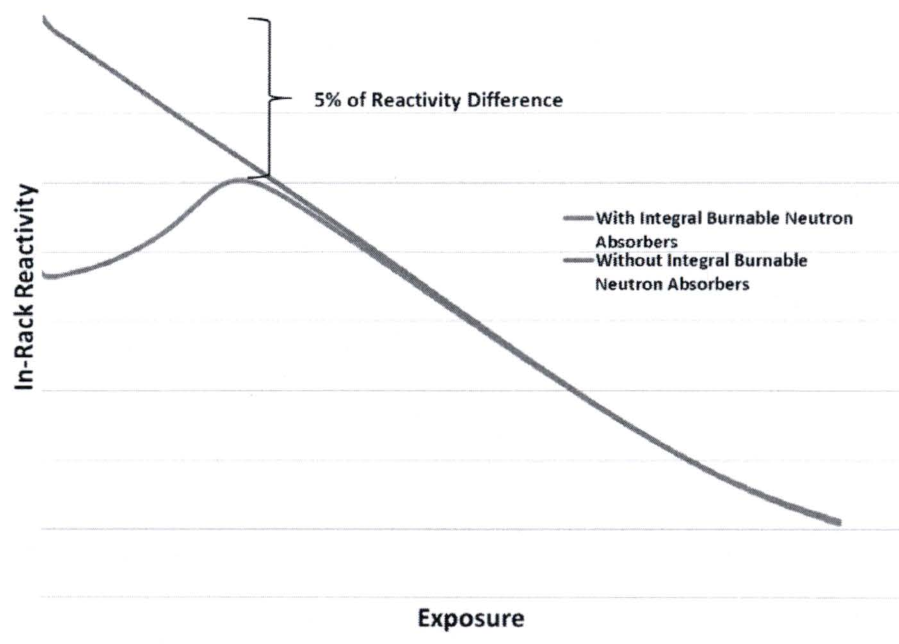
All BWR criticality calculations should ensure a conservative reactivity is analyzed in the storage configuration with consideration given to possible cooling and discharge times. Nominal values for lattice parameters like fuel pellet density, fuel rod diameter, etc. should be used and the tolerances on these parameters should be evaluated in the tolerance analysis described in Section 5.1.2.

4.3.2 BWR Depletion Uncertainty

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

The reactivity decrement to the burnup of interest is, specifically, the cold, beginning-of life (BOL) reactivity of the spent fuel rack analyzed bundle with no integral burnable neutron absorber present compared to the reactivity of the cold, analyzed bundle at the exposure statepoint used in the analysis as shown in Figure 4-5. Both reactivities are calculated for comparison in the rack system. Five percent of the difference in reactivities between these two cases is included as an uncertainty to the spent fuel pool criticality analysis to cover the depletion isotopic benchmarking gap. Figure 4-5 illustrates determination of the reactivity decrement for BWR criticality analysis where the burnup of interest is the peak reactivity.

Figure 4-5: BWR Peak Reactivity Depletion Uncertainty



The depletion uncertainty covers the uncertainty in the change in macroscopic cross-sections (i.e., the change in isotopic number densities generated during the depletion simulations multiplied by the microscopic cross-sections). If a reduction in the recommended 5% of the Δk

of depletion is required and a chemical assay approach is used, the uncertainty from the minor actinides and fission product cross sections, which are not explicitly represented in the critical experiments needs to be covered. Treatment of the bias associated with the minor actinides and fission products are addressed in Section A.2.

5 RACK AND FRESH FUEL MODELING

5.1 FUEL ASSEMBLY MODELING

The fuel assembly modeling in the criticality code includes an explicit representation of the fuel rods. For 3D analysis, the fuel rods are modeled with a length equal to the active fuel length. Hardware above and below the active fuel length is ignored and modelled as a water reflector (with no soluble boron) of the same temperature and density as the moderator in the active fuel region. For simplification, grids can be neglected in the fuel assembly model, as long as an additional 50ppm of soluble boron is reserved when soluble boron is credited [31]. For BWR fuel, the fuel assembly can be stored in the rack either with or without the channel and thus the impact of the channel presence (or absence) should be investigated. The fuel assembly is modeled using nominal dimensions with manufacturing tolerances addressed separately as described in Section 5.1.2. The gap between the fuel pellet and cladding is modelled as the appropriate fill gas (i.e., Helium).

5.1.1 Design Basis Fuel Assembly

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

Calculations need to be performed for each unique rack design and fuel assembly type, using nominal dimensions to establish which fuel assembly type is most limiting. It is also important to address the change in reactivity with depletion, as the bounding fuel type can change with burnup (because of differences in the fuel to moderator ratio between different fuel designs, a fuel assembly that is bounding at fresh fuel conditions, may not be limiting at other burnups) and enrichments. The design basis fuel assembly is that assembly that provides the most limiting reactivity at a given burnup and enrichment. In the case where a single fuel assembly is not bounding over all burnup and enrichment combinations, the difference between the design basis assembly and the other more bounding assembly type(s) is applied as a bias to the calculation of maximum k_{eff}.

In the determination of the design basis assembly, it is acceptable to use a hybrid set of parameters from multiple assemblies that result in a bounding, more limiting design basis assembly. A prime example of this approach is the use of the maximum nominal fuel density

that bounds all fuel designs in the spent fuel pool. This approach also provides additional conservatism in the analysis.

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

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

5.1.2 Fuel Assembly Manufacturing Tolerances

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

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

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

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}. It is also acceptable to use a combination of these two approaches. For example, a maximum pellet density may be used and the other parameters are statistically combined.

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

previous calculations of similar systems (fuel assemblies and/or rack designs) along with engineering judgement.

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

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

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

5.1.3 Axial Burnup Distribution

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

Option 1: Use of Generic Axial Burnup Distributions

NUREG/CR-6801 [20] evaluated 3169 axial burnup profiles to determine the most reactive representatives in each burnup range. Included in the population are B&W 15x15, CE 14x14,

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

The database does not include axial burnup profiles associated with fuel assemblies containing lower enriched axial blankets at the top/bottom of the fuel assembly. However, as stated in NUREG/CR-6801,

> *"because the axial blankets have significantly lower enrichment than the central region, the end effect for assemblies with axial blankets is typically very small or negative... consequently, profiles from assemblies with axial blankets were not considered..."*

Therefore, because the end effect for assemblies with axial blankets is very small or negative, it is acceptable to consider axially blanketed fuel assemblies bounded by fuel assemblies with no axial blankets.

Because of the broad range of applicability and the conservative nature of using the most reactive axial burnup profile for each identified burnup range, there is reasonable assurance that axial burnup profiles from future discharged fuel assemblies will also be bounded by the database of profiles contained in NUREG/CR-6801. If significant changes are made to the core operation (e.g., load following, significant low-power operation, gray rods, flux suppression assemblies, etc.), it should be verified that the new axial burnup distributions still behave in a similar manner as the axial burnup distribution before the core design change.

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

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

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

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

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

Nodalization

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

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

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

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

These two references are consistent in recognizing the importance of the size of the nodes at the ends of the assembly (approximately 8 inches or less) and the non-importance of the nodal structure at the center of an assembly modeled with a distributed axial burnup profile. Therefore, the analyst should confirm that the nodes of the axial burnup distribution are appropriately sized, especially at the ends of the assembly. The recommended approach is to utilize equally sized nodes along the length of the active fuel no larger than eight inches.

5.1.4 Reactor Record Burnup Uncertainty

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

Both EPRI and ORNL have performed studies to evaluate the accuracy of reactor records [21, 35]. The EPRI and ORNL reports agree that burnup estimations based on the flux measurements followed by time integration are within 5% of the true assembly burnup, and as such using 5% as the BMU is conservative. It should be noted that both studies indicate that when using properly calibrated core follow software which is updated with in-core measurements the uncertainty is less than 2%, however this would need to be justified on an application-specific basis. Therefore, the burnup uncertainty should be accounted for either by including a stand-alone uncertainty in the calculation of the sum of biases and uncertainties, or directly reducing the burnup of assemblies before storing them in the SFP.

5.1.5 Assembly Inserts and Integral Absorber Credit

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

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

Non-irradiated removable burnable absorbers (i.e., WABA's, BPRA's, borated SS rods) can be credited to provide additional reduction in the required burnup for storage. The primary effect is associated with crediting the neutron absorption capabilities of the insert, with a secondary effect associated with moderator displacement from the guide tube. A conservative approach is to model the insert with nominal geometrical dimensions in conjunction with a minimum absorber loading. In this approach, additional uncertainty analysis for the absorber in not necessary.

Irradiated removable burnable absorbers (i.e., WABA's, BPRA's) can also be credited to provide additional reduction in the required burnup for storage. Since the strong neutron absorber is no longer present the primary effect is associated with moderator displacement from the guide tube and can provide some small benefit. Any residual absorber should be neglected as a conservative approach.

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

5.2 STORAGE RACK MODELING

5.2.1 New Fuel Vault

While the New Fuel Vault is a dry environment for unirradiated fuel assemblies, both full (100% density) moderator condition as well as optimum low density moderator condition (i.e., mist or foam) 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. If used for storage of new fuel, specific calculations are necessary to assure the maximum k_{eff} is no greater than the regulatory limits. In the evaluation of the new fuel vaults, characteristics of the fuel assemblies, rack, vault construction, and any materials or equipment stored in the new fuel vault should be explicitly identified and evaluated, as applicable.

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

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

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

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

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

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

5.2.2 Spent Fuel Pool Racks

The spent fuel pool rack criticality model consists of a representation of the dimensions and materials of construction, including any installed neutron absorber as well as flux traps (if present). The rack structure should be modeled using nominal dimensions with an axial length equal to the active fuel region. If the neutron absorber does not extend the entire length of the active fuel region, it should be appropriately modelled, depending on the location of the active fuel region in relation to the neutron absorber. The rack structure above and below the active fuel region are neglected and replaced with unborated water (even when borated water is used in the active fuel region). It is acceptable for minor parts of the rack construction (i.e., welds) to be neglected and replaced by water. Credit can be taken for radial leakage near the walls of the spent fuel pool for the purposes of allowing lower burnup fuel requirements on the periphery of the spent fuel pool. In general, the concrete composition has a negligible impact on reactivity [31], but a conservative dry concrete density of 2.90 g/cm^3 is recommended [31].

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

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

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

5.2.2.1 Spent Fuel Pool Temperature

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

5.2.2.2 Dimensions

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 the length and width may be used in lieu of tolerance analyses.

Many racks have thin stainless steel sheets covering the neutron absorber material. The reactivity effect of the manufacturing tolerance on this cover is negligible in non-flux-trap rack designs that contain absorber panels and further calculations for this reactivity is not required [31]. For flux-trap rack designs, the uncertainty due to the manufacturing tolerance on these sheets is small, but cannot generically be declared negligible. Similarly, in racks without neutron absorber, the steel of the sheathing has a small non-negligible effect.

5.2.2.3 Rack Neutron Absorbers

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

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

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

5.2.2.3.1 Boron Content

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

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

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

5.2.2.3.2 Neutron Absorber Aging Effects

Certain neutron absorbers may undergo aging effects (i.e. changes in material dimensions or composition over the service life of the neutron absorber). The mechanisms for undergoing changes and the potential impact on their ability to perform their criticality control function are typically specific to the absorber material and rack design. The criticality analysis 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 included in the criticality analysis sufficient to ensure the analysis is bounding for the intended service life of the pool.

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

5.2.2.4 Eccentric Positioning

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

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

- When neutron absorber panels are present, a centrally located positioning of the fuel assembly in the storage cell is the most reactive configuration.
- When the neutron absorber is not present (or not credited), an eccentrically located positioning of the fuel assembly in the storage cell is the most reactive configuration:
  - As the size of the model increases (and therefore more assemblies are eccentrically located) the reactivity increases. However, the likelihood of an increasing number of fuel assemblies being eccentrically located in the most reactive configuration also decreases.

While it would be conservative to consider the maximum impact due to eccentric positioning from a very large model (e.g., 400 assemblies in a 20x20 array all located eccentrically towards the center of the storage rack) as a bias, it also unnecessarily penalizes the analysis for a configuration that is not deemed credible.

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

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

6 CONFIGURATION MODELING

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

6.1 NORMAL CONDITIONS

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

6.2 INTERFACES

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

Previous guidance provided in DSS-ISG-2010-01 [25] has provided two possible paths to show acceptability of storage configuration interfaces. The first option was to use the maximum biases and uncertainties from the individual storage configurations. The second option was to determine biases and uncertainties specific to the interface. This first option will always show a higher reactivity than either of the two storage configurations that are part of the interface. This could lead to the conclusion that there is an increase in reactivity due to the interface when that is not necessarily the case. The second option is computationally burdensome, especially when there are multiple options for different storage configurations and hence many possible interfaces.

When an interface calculation is performed, essentially two semi-infinite arrays of each storage configuration are placed in the same model, possibly with a small gap between them in the case of rack-to-rack interfaces (i.e., no leakage is credited). If the model is sufficiently large enough (4 or more rows of storage cells of each configuration), the resulting k_{eff} of the interface can determine if the interface results in a more limiting condition than the individual storage configurations. If the interface calculations show that the reactivity of the interface is essentially equivalent to (within the statistical uncertainty of the calculations) or less than the reactivity of the most reactive of the two storage configurations, then there is no additional neutronic coupling between the individual storage configurations. The following criteria are specified for interface calculations between two different storage configuration:

- If the calculated k_{eff} of the interface model is less than or equal to (within the statistical uncertainty of the calculations) the maximum calculated k_{eff} of each individual storage configuration, then no further restrictions are necessary.
- If the interface calculation has a higher calculated k_{eff} (outside the statistical uncertainty of the calculations), compared to the maximum calculated k_{eff}, of each individual storage configuration, then appropriate restrictions should be specified to limit these storage patterns from being used adjacent to one another.

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

An interface can also occur between old and new racks. If the separation distance between the new and old racks is more than 6 inches at the interface, then there is no need to evaluate the interface between storage racks/configurations. This stipulation is not applicable to older "non-

poisoned low density rack module designs" that relied upon geometric separation for maintaining sufficient sub-criticality to meet regulatory requirements. However, it is applicable to high-density storage rack designs, with or without neutron absorber, that rely on burnup credit (PWR) and/or maximum reactivity credit (BWR) to maintain sub-criticality.

6.3 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. For PWR spent fuel pools that credit soluble boron, the limiting misload will be the accident which requires the highest soluble boron to ensure that the maximum k_{eff} does not exceed 0.95. The separate boron dilution accident is discussed in Section 7.3.

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

6.3.1 Temperatures Beyond Normal Operating Range

The spent fuel pool has a normal operating range for the bulk temperature of the spent fuel pool water. Under accident conditions (loss of cooling) this temperature could be elevated beyond the normal operating range. Because the pool temperature is not a major contributor to reactivity and soluble boron credit can be taken for accident conditions, analysis should be done at boiling conditions with a void fraction up to 20% to confirm that higher temperature conditions are not limiting for PWR pools.

6.3.2 Dropped and Mislocated Assembly

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

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

6.3.3 Neutron Absorber Insert Misload

Some storage configurations may credit the neutron absorption capabilities of neutron absorber inserts, RCCAs, WABAs, BPRAs, etc. The potential exists for these devices to not be located

where assumed in the criticality analysis and therefore should be investigated as part of the accident analysis. In most cases, this scenario will be bounded by the fresh fuel misload described in Section 6.3.4, but should nonetheless be evaluated or justified as being bounding by other scenarios.

6.3.4 Assembly Misload

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

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

Additionally, there is the possibility of an error occurring in the selection of appropriate storage configurations such that a single initiation event can result in multiple fuel assemblies being misloaded. Whereas a single misload is typically a result of an error in the fuel handling selection or relocation of an assembly (i.e., picking up and moving an assembly other than the intended assembly), a single event resulting in multiple misloaded assemblies is typically the result of a planning or process error. Therefore, whether multiple misloaded assemblies is credible from a single event 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 therefore becomes vital to precluding a common cause failure of misloading multiple assemblies.

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

- Licensee controlled procedures, programs
- Event tree analysis
- Post-movement fuel assembly verification
- Storage cell blocking devices
- Analysis of multiple misload scenarios, if applicable

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

6.3.4.1 Licensee Controlled Administrative Programs

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

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

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

6.3.4.2 Event Tree Analysis

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

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

6.3.4.3 Post-Movement Assembly Verification

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

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

6.3.4.4 Storage Cell Blocking Devices

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

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

Fuel-debris trash cans may be used as blocking devices, provided that they meet all of the above criteria except the requirement for specialized tools. Specialized tools are not required for trash cans as their physical appearance is easily distinguishable from a fuel assembly

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

6.3.4.5 Multiple Misload Analysis

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 choose to 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.

6.3.5 Seismic Movement

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

7 SOLUBLE BORON CREDIT

7.1 Normal Conditions

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

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

7.2 Accident Conditions

For conditions with soluble boron, the accident conditions in Section 6.3 should be evaluated at the minimum soluble boron concentration allowable under the site's Technical Specification. In other words the accident condition does not need to consider a simultaneous boron dilution event, per the double-contingency principle, if the accident does not also result in boron dilution. This is

justified through application of risk insights, in that the probability of a significant boron dilution event (violating the minimum pool soluble boron concentration) is remote, and that there have not been any known cases of its occurrence in the history of nuclear power operations.

For the accident conditions, the maximum calculated k_{eff}, including all biases and uncertainties, must be less than the regulatory k_{eff} limit of 0.95.

7.3 BORON DILUTION

In the event the licensee is crediting soluble boron in the criticality safety calculation, a boron dilution accident 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.

8 CALCULATION OF MAXIMUM K_{EFF}

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

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

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. These are summarized here for completeness:

Biases
Criticality Code Validation bias
Moderator Temperature Bias
Design Basis Fuel Assembly Bias
Eccentric Positioning Bias
Depletion Code Bias
Actinide and Fission Product Worth Bias
Bias for Validation Gaps

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

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

9 LICENSEE CONTROLS

9.1 LICENSEE CONTROLS

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

9.2 PROCEDURAL CONTROLS

A licensee establishes procedural controls in order to ensure that used fuel is stored in accordance with the Technical Specifications, and to govern the planning and performance of fuel movements. These procedures implement the requirements for tracking the location of fuel assemblies in accordance with Special Nuclear Material (SNM) regulations and the spent fuel pool criticality analysis. They also ensure proper assembly selection for core loading activities, thermal management, gamma flux, etc. In addition, programs and procedures are established to ensure that the licensee is following their software QA plan. The software QA program covers the use of codes for criticality analyses and software used to plan and implement fuel movements.

Procedural controls should be developed based upon the complexity of storage patterns in order to provide reasonable assurance of adequate public health and safety. The procedures may also affect the assumed accident conditions (see Section 6.3) For example, if storage patterns are

relatively straight-forward and the procedures 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 procedures used by licensees. Additional procedures 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)
  - Technical Specification for soluble boron surveillance
  - Neutron Absorber Panel material behavior monitoring program
- Software Requirements:
  - Independent review of software implementation and revision, testing and documentation is performed by an independent reviewer
  - Configuration controls to ensure integrity of executable files and data files
  - Cyber security controls prevent tampering / inadvertent changes

- Database Requirements:
  - Independent review and approval of all database updates
  - Procedures to ensure integrity of database prior to utilizing the data

9.3 NEW (FUTURE) FUEL TYPES

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

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

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

9.4 PRE- AND POST-IRRADIATION FUEL CHARACTERIZATION

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

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

For any given fuel assembly, fuel characterization consists of two processes. The first process is pre-irradiation characterization, and its purpose is to review the design of the fuel assembly against the parameters assumed in the criticality analysis. Ideally, this is performed as part of the core design process. In any case, it is performed before the fuel in question is placed, for the first time, in the new or spent fuel racks. For pressurized water reactors, the key inputs pertain to the fuel loading (fuel pellet mass, diameter, density, enrichment, etc.) and to the fuel-to-moderator ratio (fuel rod diameter, fuel rod pitch, etc.). Boiling water reactors should also consider the lattice itself (8x8, 9x9, 10x10, etc.), as well as the characteristics of the fuel channel. One acceptable method for BWR fuel characterization is the in-core k_∞ methodology. This method establishes infinite-lattice reactivity limits for each fuel storage region as part of the criticality safety analysis. Each unique fuel design is then validated against this reactivity limit to establish

its acceptability for storage. Other characteristics to be considered will depend upon the nature of the criticality analysis itself. For example, if the analysis took credit for the initial presence of burnable absorbers in the fuel, then the characteristics of the burnable absorber (type, loading, and configuration) should also be considered.

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

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

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

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

- Operation for a significant period of time at reduced power or with control rods inserted more than during normal operations
- Changes to plant configuration that result in higher-than-expected reactor coolant temperatures

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

10 REFERENCES

10.1 REGULATIONS

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

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

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

5. 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.

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APPENDIX A: COMPUTER CODE VALIDATION

A.1 Criticality Code Validation using Fresh Fuel Experiments

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

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

NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," provides guidance on one approach for performing the validation [13].

A.1.1 Identify Range of Parameters

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

- Isotopic Content
    - Experiments should cover material for the rack structure (e.g., stainless steel), materials in the surrounding geometry (e.g., water/concrete), material for the cladding (e.g., zirconium), fissile isotopes in the applicable enrichment range (e.g., ^{235}U for low enriched UO_2, ^{239}Pu for MOX), water temperature, 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.

A.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* [26] 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 selection of critical experiments to be included in the validation should include benchmark experiments from multiple facilities and experiment series to eliminate the possibility of a facility-specific or experiment series-specific bias.

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.

A.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). Additionally, the analyst must verify and ensure the accuracy of the critical experiment models used in the validation, even if provided from a third party reputable source.

A.1.4 Analysis of the Critical Experiment Data

NUREG/CR-6698 [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 [26] 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).

A.1.5 Area of Applicability

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

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

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.

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

A.2 DEPLETION CODE 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. This section provides several validation approaches for both PWR and BWR racks to explicitly quantify a depletion uncertainty. The use of the 5% of the reactivity decrement as a depletion uncertainty is acceptable for ENDF/B-V through VII. [27, 33, 34].

A.2.1 PWR USED FUEL VALIDATION

Two acceptable approaches for PWR used fuel validation are presented. These approaches can be combined or used separately. The approaches are:

1. Use benchmarks based on depletion reactivity inferred from reaction rate measurements at power plants.
2. Use chemical assays and cross section uncertainties.

A.2.1.1 Validation Using Measured Flux Data from Power Reactors

PWR depletion benchmarks were developed by EPRI [27,28] using a large set of power distribution measurements to ascertain reactivity biases. The predicted reactivity of the fuel assemblies was adjusted to find the best match between the predicted and measured power distribution. EPRI used 680 flux maps from 44 cycles of PWR operation at 4 PWRs to infer the depletion reactivity [28]. The depletion reactivity has been used to create 11 benchmark cases for various burnups up to 60 GWd/T and 3 cooling times 100 hour, 5 years, and 15 years. All of these benchmark cases should be analyzed with the code set (depletion and criticality codes) to be used in the criticality analysis to establish a bias for the depletion reactivity. The uncertainty in the benchmarks should be used as the depletion reactivity uncertainty. These biases and uncertainties cover both the isotopic content uncertainty and the worth uncertainty associated with depletion. They account for all the changes from the initial fresh fuel condition. The bias and uncertainty associated with fresh fuel are also required to be included in the validation of the

criticality safety evaluation. A companion EPRI report describes in detail how to apply the benchmarks in the criticality safety analysis [27].

A.2.1.2 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]. NUREG/CR-7108 evaluates differences in measured and calculated isotope concentrations. NUREG/CR-7109 evaluates the bias and uncertainty attributable to cross section data uncertainty for minor actinides and fission products.

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 are similar to those used to produce the bias and uncertainty. It should be noted that depletion worth bias and uncertainty derived from chemical assay data tend to be significantly conservative due to the large experimental uncertainties in performing chemical assays. Experimental uncertainty in measured isotopic content is carried over into the calculation of measured versus predicted SFP rack k_{eff}, thereby increasing the apparent uncertainty of predicted depletion worth. Studies in the past have shown that predicted depletion uncertainty derived from chemical assay data changes very little with calculational 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.

In NUREG/CR-7108 a method was presented [16] that relies on determination of distribution functions (measured versus predicted isotopic content) for key isotopes. This 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 and produces a small number of data points in the lower burnup bins, resulting in very large relative uncertainty for burnups below 30 GWD/MTU.

A direct difference approach was also presented in NUREG/CR-7108 that directly models each chemical assay sample. Direct difference modeling 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 99 of 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. Note that the HB Robinson sample N-9C-J should not be included in the set since it was deliberately taken from under an Inconel grid.

Both of these chemical assay approaches result in a conservative estimate of the bias and uncertainty for the 28 major isotopes selected. However, NUREG/CR-7108 states that most of the bias and uncertainty (90-95%) is attributable to ^{235}U and ^{239}Pu. It has been shown that the isotopes in excess of the 28 major isotopes selected in NUREG/CR-7108 have a relatively small worth. Therefore, for analyses crediting more than the 28 major isotopes, it is recommended that the bias and uncertainty from the chemical assays also be augmented by an additional bias for other minor actinides and fission products based on NUREG/CR-7109 [15].

NUREG/CR-7109 [15] recommends a bias of 1.5 % of the reactivity worth of the isotopes not included in benchmark critical experiments to cover cross section bias and uncertainty. The isotopes used in addition to the 28 isotopes directly evaluated are expected to behave similarly so a bias of 1.5% of the reactivity worth of all depletion isotopes except U, Pu, and ^{241}Am is recommended. This recommendation applies to enrichments less than 5.0 wt% ^{235}U using ENDF/B-V, ENDF/B-VI and ENDF/B-VII cross sections, for burnups up to 70 GWD/MTU and total minor actinide and fission product nuclide worth not to exceed 0.1 Δk. Additional justification should be provided for evaluations using other cross section data.

This approach assumes that the major actinides have been validated using critical experiments. Therefore, critical experiments containing Plutonium isotopes are needed. However, several ENDF/B-V and VII validation analyses containing MOX critical experiments or the HTC critical experiments have a smaller bias than fresh UO_2 experiments. However, the applicant should include in the validation fresh UO_2, MOX and HTC experiments. The code bias and uncertainty should be determined both with UO_2 experiments alone and with HTC and MOX experiments included. The appropriate bias and uncertainty from each of these cases should be included for fresh and spent fuel, respectively.

A.2.2 BWR USED FUEL VALIDATION

An acceptable approach for BWR used fuel validation is presented.

A.2.2.1 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.

A.3 APPLICATION OF CODE VALIDATION

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. 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.

A.4 ALTERNATE CODE VALIDATION

If a code (the primary code) is not capable of directly modeling the benchmark experiments, then an intermediary code (i.e., a secondary code) may be used that is validated to the benchmark experiments, and to which the primary code is validated. The primary code (code used for the criticality safety analyses) should still be capable of accurately modeling all the important neutronic and geometric aspects of storage. The secondary code should be validated against benchmark experiments that are similar to the neutronics and geometry of the criticality safety analysis in accordance with Section A.1. The primary code can then be validated by benchmarking to the secondary code over a range of parameters (neutronic and geometric) that bound the range of parameters for the criticality safety analysis. Those parameters that are important to be validated between the primary and secondary code include:

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

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

APPENDIX B: APPLICATION FORMAT AND CONTENT

1.0 Introduction and Overview

Statement of purpose for the application with a summary of physical changes (e.g., new storage racks/inserts, extended power uprate), scope of the analysis, including the plant, unit, and storage area that are analyzed (i.e., spent fuel pool, new fuel racks or both).

Brief description of what is evaluated (e.g., rack types) and allowed credits, including burnup, post-irradiation cooling time, presence of soluble boron (PWR) and other neutron absorbing materials (e.g., poison panels or inserts), and any other plant-specific items (e.g., control rod assemblies, fresh burnable absorbers, restricted rack loading patterns).

2.0 Acceptance Criteria and Regulatory Guidance

Requirements of the Code of Federal Regulations, Title 10, Part 50, General Design Criteria for the spent fuel pool, regulatory guides, and standard review plan sections, if applicable, are stated in this section.

3.0 Reactor and Fuel Design Description

This section provides basic data on the type of reactor, operating characteristics and the fuel types that are being used and subsequently stored. Neutronically important physical characteristics, as well as mechanical features of each fuel design, are outlined here. This section discusses the non-mechanical fuel features (e.g., burnable absorbers, axial blankets) which are important to criticality safety and considered in the analysis. Sketches, figures and summary tables of important parameters for reactor operation and fuel design are detailed in this section.

4.0 Spent Fuel Pool/ Storage Rack Description

The physical characteristics of the SFP(s) are detailed in this section, including the pool dimensions, and pool wall and liner thickness. Rack designs (e.g., low/high-density), dimensions, associated tolerances, applicable absorbers that are integral to the racks are discussed. Sketches, figures and summary tables of important parameters for the storage racks/facilities are detailed in this section.

5.0 Overview of the Method of Analysis

Applicable fuel assembly categories suitable for storage in storage arrays are described here. The acceptability of the storage arrays developed in the analysis is ensured by controlling the assemblies that can be stored in each array. Assemblies may be divided into groups (e.g., based on assembly type) and/or fuel categories based on assembly reactivity determined as a function of assembly average burnup, initial enrichment, burnable poison content, decay time, etc.

6.0 Computer Codes, Cross Sections and Validation Overview

Computer codes and cross-section libraries employed in the analysis are described in this section. Traceable version numbers for the codes and the nuclear data libraries, as well as any patches applied are documented herein. The descriptions include:

a) The lattice-physics code and its cross-section library used for simulation of in-reactor fuel assembly depletion;
b) The Monte Carlo code package and its cross-section library to determine the reactivity of the storage arrays in the spent fuel pool.

7.0 Criticality Safety Analysis of the New Fuel Rack

This section describes the criticality code calculations and evaluations performed in developing the requirements for fuel storage (e.g., enrichment, storage location restrictions, absorber credit) in the new fuel storage vault and confirming continued regulatory compliance during both normal and accident conditions.

8.0 Depletion Analysis for Spent Fuel

This section describes the methods used to determine the appropriate inputs for the generation of isotopic number densities for input to the Monte Carlo analyses.

<u>Depletion Model Considerations</u>

Simplifications and modeling considerations for depletion calculations are outlined.

<u>Depletion Parameters</u>

Details on the choice of fuel depletion parameters (e.g., power, fuel/moderator temperatures, void content) are included in this section. A description of the conservatisms applied to the depletion parameter choices should be included. The description should be sufficiently detailed to enable direct comparisons with reactor operating parameters to validate continued conservatism of the inputs utilized.

9.0 Criticality Safety Analysis of Spent Fuel Pool Storage Racks

This section describes the criticality code calculations and evaluations performed in developing the requirements for fuel storage (e.g., burnup, enrichment) and confirming continued regulatory compliance during both normal and accident conditions.

<u>Monte Carlo Modeling Approach</u>

Modeling approach, including simplifications/assumptions in Monte Carlo criticality code calculations are presented in this section. Monte Carlo calculations are performed to determine the maximum reactivity of irradiated and fresh (for PWR) fuel assemblies loaded in storage

arrays, as well as the sensitivity of the storage arrays to manufacturing tolerances, fuel depletion, eccentric positioning, and the allowable temperature range of the pool. Accident scenarios are also modeled using Monte Carlo simulations to confirm that all requirements are met (crediting soluble boron where applicable for PWR SFPs). The dimensions and tolerances used in the analysis for the design basis fuel assemblies, fixed absorbers and the fuel storage racks are presented. These dimensions and tolerances are the basis for the Monte Carlo models used to determine the requirements (e.g., burnup, enrichment, etc.) for each fuel storage array and to confirm the safe storage of the spent fuel pool under normal and accident conditions.

Design Basis Fuel Description

Details on the determination of the fuel designs that are limiting are included in this section. Selected fuel design conservatisms are identified (e.g., the use of a single limiting fuel pin enrichment for a lattice, bounding fuel density).

Array Descriptions

Descriptions of the allowable fuel storage configuration(s) and the fuel categories that populate those configurations are described here. A storage configuration is allowed to be populated by assemblies of the fuel category dictated in the fuel category definition or a lower reactivity fuel category.

Burnup Limit Generation (PWR)/(SCCG k-inf gnereation – BWR)

To ensure the safe operation of the spent fuel pool, the analysis defines fuel storage configurations which dictate where assemblies can be placed in the SFP. For PWR spent fuel pools, limits are based on each assembly's enrichment (wt% ^{235}U), assembly average burnup (GWd/MTU), credit for presence of removable/nonremovable absorbers, and cooling time since discharge. For BWR spent fuel pools, the limit is based on the assembly's maximum k_{inf} in the Standard Cold Core Geometry (SCCG)

10.0 Interface Analysis

Interfaces are the locations in the spent fuel pool where more than one storage rack design or storage configuration are adjacent to one another. All types of interfaces present in the SFP analysis are defined and addressed in this section, along with any interface restrictions.

11.0 Normal Conditions

This section describes any other normally occurring fuel, or operational activities, etc. that occur in the spent fuel pool. Examples of such activities include fuel repair/reconstitution, fuel inspection, movement and placement of fuel in and around the spent fuel pool such as the fuel elevator.

Soluble Boron Credit (Typically only PWR Fuel)

The minimum soluble boron concentration to maintain $k_{eff} < 0.95$ for the limiting normal condition including biases, uncertainties, and administrative margin is defined here, considering ^{10}B abundance assumptions.

12.0 Accident Analysis

The criticality cases modeling the range of applicable accident scenarios for the SFP are presented here. The accident condition applicable biases and uncertainties are presented and combined with the calculated accident condition k_{eff} to produce the accident condition maximum k_{eff} that is compared against the regulatory limit.

13.0 Analysis Results and Conclusions

This section documents the results of the plant-specific criticality safety analysis. Included in this section are the requirements (e.g., burnup, enrichment) for the fuel storage arrays, area of applicability, and restrictions of the analysis.

Burnup Limits for Storage Arrays (PWR)

Minimum burnup requirements as a function of initial enrichment and decay time for each array are provided as tables of polynomial coefficients. Decay time interpolation rules are also addressed.

Storage Configuration Restrictions

The purpose of this section is to summarize the restrictions of the fuel being stored at the plant.

Soluble Boron Credit (Typically only PWR Fuel)

The amount of soluble boron credited to keep $k_{eff} < 0.95$ under all normal and credible accident scenarios is reported and compared to Technical Specification value.

14.0 References

References used in the analysis are listed here.

Appendix A: Computer Code Validation

This appendix describes the code validation performed for the depletion and/or criticality code validation.

APPENDIX C: CRITICALITY ANALYSIS CHECKLIST

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

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

Subject	Applicable	Notes / Explanation
1.0 Introduction and Overview		
Purpose of submittal		
Changes requested		
Summary of physical changes		
Summary of analytical scope		
2.0 Acceptance Criteria and Regulatory Guidance		
Summary of requirements and guidance	YES/NO	
Requirements documents referenced	YES/NO	
Guidance documents referenced	YES/NO	
Acceptance criteria described	YES/NO	
3.0 Reactor and Fuel Design Description		
Describe reactor operating parameters	YES/NO	
Describe all fuel in pool	YES/NO	
Nominal and tolerance dimensions	YES/NO	
Schematic of guide tube patterns	YES/NO	
Describe future fuel to be covered	YES/NO	
Describe all fuel inserts	YES/NO	
Nominal and tolerance dimensions	YES/NO	
Schematic (axial/cross-section)	YES/NO	
Describe non-standard fuel	YES/NO	
Describe non-fuel items in fuel cells	YES/NO	
Nominal and tolerance dimensions	YES/NO	
4.0 Spent Fuel Pool/Storage Rack Description		
New fuel vault & Storage rack description	YES/NO	
Nominal and tolerance dimensions	YES/NO	
Schematic (axial/cross-section)	YES/NO	
Spent fuel pool & Storage rack description	YES/NO	
Nominal and tolerance dimensions	YES/NO	
Schematic (axial/cross-section)	YES/NO	

Subject	Applicable	Notes / Explanation
5.0 Overview of the Method of Analysis		
New fuel rack analysis description	YES/NO	
Storage geometries	YES/NO	
Bounding assembly design(s)	YES/NO	
Integral absorber credit	YES/NO	
Accident analysis	YES/NO	
Spent fuel storage rack analysis description	YES/NO	
Storage geometries	YES/NO	
Bounding assembly design(s)	YES/NO	
Soluble boron credit	YES/NO	
Boron dilution analysis	YES/NO	
Burnup credit	YES/NO	
Decay time credit	YES/NO	
Integral absorber credit	YES/NO	
Other credit	YES/NO	
Fixed neutron absorbers	YES/NO	
Aging management program	YES/NO	
Accident analysis	YES/NO	
Temperature increase	YES/NO	
Assembly drop	YES/NO	
Multiple misload	YES/NO	
Boron dilution	YES/NO	
Other	YES/NO	
Fuel out of rack analysis	YES/NO	
Handling	YES/NO	
Movement	YES/NO	
Inspection	YES/NO	
6.0 Computer Codes, Cross Sections and Validation Overview		
Code/Modules Used for Calculation of k_{eff}	YES/NO	
Cross section library	YES/NO	
List all the isotopes used	YES/NO	
Convergence checks	YES/NO	
Code/Module Used for Depletion Calculation	YES/NO	
Cross section library	YES/NO	
List all the isotopes used	YES/NO	
Convergence checks	YES/NO	
Validation of Code and Library	YES/NO	
Major Actinides and Structural Materials	YES/NO	
Minor Actinides and Fission Products	YES/NO	
7.0 Criticality Safety Analysis of the New Fuel Rack		
Rack model	YES/NO	

Subject	Applicable	Notes / Explanation
Boundary conditions	YES/NO	
Source distribution	YES/NO	
Geometry restrictions	YES/NO	
Limiting fuel design	YES/NO	
Fuel density	YES/NO	
Burnable Poisons	YES/NO	
Fuel dimensions	YES/NO	
Axial blankets	YES/NO	
Limiting rack model	YES/NO	
Storage area walls	YES/NO	
Temperature	YES/NO	
Multiple regions	YES/NO	
Flooded	YES/NO	
Low density moderator	YES/NO	
Eccentric fuel placement	YES/NO	
Tolerances	YES/NO	
Fuel geometry	YES/NO	
Fuel pin pitch	YES/NO	
Fuel pellet OD	YES/NO	
Fuel clad OD	YES/NO	
Fuel content		
Enrichment	YES/NO	
Density	YES/NO	
Rack geometry		
Rack pitch	YES/NO	
Cell wall thickness	YES/NO	
Code uncertainty	YES/NO	
Biases		
Temperature	YES/NO	
Code bias	YES/NO	
Accident analysis		
Flooding (water and low density moderator)	YES/NO	
8.0 Depletion Analysis for Spent Fuel		
Depletion Model Considerations		
Time step verification	YES/NO	
Convergence verification	YES/NO	
Simplifications	YES/NO	
Non-uniform enrichments	YES/NO	
Depletion Parameters		
Burnable Absorbers	YES/NO	
Integral Absorbers	YES/NO	
Soluble Boron	YES/NO	
Fuel and Moderator Temperature	YES/NO	

Subject	Applicable	Notes / Explanation
Power	YES/NO	
Control rod insertion	YES/NO	
9.0 Criticality Safety Analysis of Spent Fuel Pool Storage Racks		
Rack model	YES/NO	
Boundary conditions	YES/NO	
Source distribution	YES/NO	
Geometry restrictions	YES/NO	
Design Basis Fuel Description	YES/NO	
Fuel density	YES/NO	
Burnable Poisons	YES/NO	
Fuel assembly inserts	YES/NO	
Fuel dimensions	YES/NO	
Axial blankets	YES/NO	
Configurations considered	YES/NO	
Borated	YES/NO	
Unborated	YES/NO	
Multiple rack designs	YES/NO	
Alternate storage geometry	YES/NO	
Axial burnup shapes		
Uniform/Distributed	YES/NO	Option 1, Option 2 or Option 3
Tolerances/Uncertainties		
Fuel geometry		
Fuel rod pin pitch	YES/NO	
Fuel pellet OD	YES/NO	
Cladding OD	YES/NO	
Axial fuel position	YES/NO	
Fuel content		
Enrichment	YES/NO	
Density	YES/NO	
Rack geometry		
Flux-trap size (width)	YES/NO	
Rack cell pitch	YES/NO	
Rack wall thickness	YES/NO	
Neutron Absorber Dimensions	YES/NO	
Code uncertainty	YES/NO	
KENO case uncertainty	YES/NO	
Depletion Uncertainty	YES/NO	Kopp (5%) or EPRI
Burnup Uncertainty	YES/NO	Included in Burnup requirements?
Biases		
Fuel design	YES/NO	
Minor actinides and fission product worth	YES/NO	
Code bias	YES/NO	

Subject	Applicable	Notes / Explanation
Temperature	YES/NO	
Incore thimble depletion effect	YES/NO	
NRC administrative margin	YES/NO	
Modeling simplifications		
Axial reflectors	YES/NO	
10.0 Interface Analysis		
Interface configurations analyzed	YES/NO	
Between dissimilar racks	YES/NO	
Between storage configurations within a rack	YES/NO	
Interface restrictions	YES/NO	
11.0 Normal Conditions		
Fuel handling equipment	YES/NO	
Administrative controls	YES/NO	
Fuel inspection equipment or processes	YES/NO	
12.0 Accident Analysis		
Boron dilution	YES/NO	
Normal conditions	YES/NO	
Accident conditions	YES/NO	
Single assembly misload	YES/NO	
Neutron Absorber Insert Misload	YES/NO	
Multiple fuel misload	YES/NO	
Dropped assembly	YES/NO	
Temperature	YES/NO	
Seismic event	YES/NO	
13.0 Analysis Results and Conclusions		
Summary of results	YES/NO	
Burnup curve interpolation	YES/NO	
New administrative controls	YES/NO	
Technical Specification markups	YES/NO	
14.0 References		
Appendix A: Computer Code Validation:		
Code validation methodology and bases	YES/NO	
New Fuel	YES/NO	
Depleted Fuel	YES/NO	
MOX	YES/NO	
HTC	YES/NO	
Convergence	YES/NO	

Subject	Applicable	Notes / Explanation
Trends	YES/NO	
Bias and uncertainty	YES/NO	
Range of applicability	YES/NO	