November 16, 2015

Rod McCullum, Director
Used Fuel Programs
Nuclear Energy Institute
1201 F Street, NW, Suite 1100
Washington, DC 20004

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO NEI 12-16, "GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS"

Dear Mr. McCullum:

By letter dated April 18, 2014 (Agencywide Documents Access and Management System Accession No. ML14112A517), the Nuclear Energy Institute (NEI) submitted NEI 12-16, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants" for review. Upon review of the information provided, the U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is needed to complete the review.

In an email dated September 29, 2015, Mr. Kristopher Cummings, representing the NEI, and I agreed that the NRC staff will receive your response to the enclosed Request for Additional Information (RAI) questions by December 16, 2015.

Please contact me at (301) 415-7297 if you have on questions on this subject.

Sincerely,

***/RA/***

Joseph J. Holonich, Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 689

Enclosure: RAI Questions

Rod McCullum, Director
Used Fuel Programs
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**REQUEST FOR ADDITIONAL INFORMATION FOR NEI 12-16, REVISION 1,**

**"GUIDANCE FOR PERFORMING CRITICALITY ANALYSES OF FUEL STORAGE AT**

**LIGHT-WATER REACTOR POWER PLANTS"**

**PROJECT NO. 689**

By letter dated April 18, 2014, the Nuclear Energy Institute (NEI) submitted NEI 12-16, *Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants*, Revision 1, for review and endorsement by the U.S. Nuclear Regulatory Commission (NRC) for use by licensees when performing nuclear criticality safety (NCS) analyses for fresh and spent fuel storage.

NEI 12-16's stated purpose is to provide "…acceptable methods 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." The guidance is intended to be "…comprehensive and durable guidance to improve consistency and clarity for performing criticality analyses that assure criticality safety and regulatory compliance."

With that purpose in mind the NRC staff provides the comments below for NEI's consideration. NEI may elect to revise NEI 12-16 incorporating changes sufficient to resolve some or all of the NRC staff's comments. NEI should keep in mind that any comment not resolved in a revision to NEI 12-16 will likely be addressed in any NRC document that endorses the use of NEI 12-16.

**Comments and Request for Additional Information (RAI) Questions on NEI 12-16**

**The following general comments and Request for Additional Information (RAI) Questions are provided concerning the proposed guidance for performing criticality analyses.**

1. In much of NEI 12-16 it is difficult to distinguish between specific guidance for standard methodologies and a general overview of the subject matter. Revise the report to clearly identify and define standard methodologies in enough detail that NRC staff can review and endorse using the methods described. When specific guidance is being given, include the following:

   For each topic, include:

   a. Detailed description of the standard method to be used, including:
      - analysis approach/technique description
      - conditions, prerequisites, and range of applicability
      - data requirements; and
      - simplifications, approximations, and assumptions used

   b. Justification and bases supporting the acceptability of the method

2. One of the objectives of NEI 12-16 is to facilitate a timely and efficient NRC review. Please consider revising the report to specifically address criticality analysis report format and content. Include recommendations that ensure criticality analysis report content will be

adequate to permit NRC staff to assess compliance with regulatory requirements, to gain a reasonable assurance of safety, and to perform any confirmatory calculations deemed necessary by NRC staff.

3. In order to simplify the NRC review of NEI 12-16, please revise the report to include discussion on how NEI 12-16 is to be used and provide guidance on the following:

    a. How analysts are to document use of the standard method(s).

    b. How requirements, recommendations, and permissive statements are to be interpreted by guidance users.

    c. The identification, description, and justification of deviations from standard methods and recommendations.

4. The next to last sentence in the "FOREWORD" is the following:

    *"This industry document is developed as a comprehensive guide that presents an acceptable approach to comply with the regulations, and upon NRC endorsement would supersede previous guidance documents."*

    Revise the report to modify or exclude this language. We cannot make general conclusions about topics as broad as previous guidance documents.

5. Revise the report to explicitly address the following issues, which were not addressed:

    a. Identification and justification of credited actinides and fission products.

    b. Inclusion of the impact of changes to reactor operation (i.e. power uprates and cycle lengths that may affect NCS analyses).

    c. Inclusion of the impact of changes to fuel designs (i.e. uranium–235 ($^{235}U$) content and increased pressure drop across fuel assemblies that may affect NCS analyses).

    d. Guidance for analysts to include:
        - Explicit conclusions concerning compliance with relevant regulatory requirements.
        - A clear statement of limits, controls, and requirements derived from the criticality analysis.
        - A clear statement of key limitations associated with the scope of the criticality analysis and methods used in the criticality analysis such as fuel assembly designs considered, reactor depletion conditions used, bounding modeling simplifications used, etc. This is important to support unambiguous screening of future changes.

    e. Identification of fuel and fuel designs covered by the guidance.

**The following comments are on Section 1.**

6. Section 1.4 covers the use of the double contingency principle (DCP). The DCP is consistent with current NRC guidance regarding NCS analysis for fresh and spent nuclear fuel. Additionally, the DCP is consistent with the overall concept of defense in depth.

    a. However, it is not the current intention of the NRC to endorse or appear to endorse in whole or in part American National Standards Institute (ANSI)/ American Nuclear Society (ANS)-8.1. Revise NEI 12-16 to remove the citation of ANSI/ANS-8.1.

    b. It is not clear from the guidance that independence must be shown before the DCP can be applied. The guidance should be clear that independence must be shown before the DCP is used to constrain the analysis.

7. Section 1.5 discusses the role and use of precedents.

    a. Consider expanding this section to include a statement similar to the following:

        *It should be noted that past acceptance by the NRC of a practice does not guarantee that use of the precedent will be accepted. Occasionally, an analysis flaw or error is recognized, thereby invalidating a precedent. When in doubt, discuss with NRC staff.*

    b. As currently written it appears as if the onus for identifying changes to precedents falls solely upon the NRC. Revise the guidance to indicate that applicants equally have a responsibility to identify changes to precedents.

8. Section 1.6 discusses the use of assumptions and engineering judgment. It is not clear how approximations and simplifications are treated. Approximations and simplifications are not assumptions. If the intention is that approximations and simplifications are treated the same as assumptions, then revise NEI 12-16 to make that clear. Otherwise provide guidance for documenting and justifying approximations and simplifications.

**The following comment is provided on Section 2 of NEI 12-16:**

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.

**The following comments are provided on Section 3 of NEI 12-16:**

10. Revise the report to include guidance for analysts to document which codes and data were used and whether or not any code or data patches or other modifications have been made. The guidance must direct the analysts to document the specific code and data versions used in the analysis and validation studies.

a. Include guidance to document significant computer input options and methods used, convergence criteria used, selection of $k_{eff}$ values, convergence checks performed, and evaluation of warning and error messages.

   Significant computer input options and methods typically include things like:
   - Neutrons per generation
   - Generations skipped
   - Number of generations
   - Monte Carlo calculation uncertainty
   - Nuclear data scattering order used
   - Multi-group data resonance processing models and parameters used
   - Convergence criteria used, such as maximum flux difference, maximum eigenvalue difference, or maximum Monte Carlo uncertainty
   - Initial neutron distribution for Monte Carlo method calculations

b. The guidance on nuclear data libraries is vague and implies that any library would be acceptable. However, there is no guidance on how to select or justify the library. Revise the guidance to include the use of particular libraries or provide guidance on selecting the library.

11. The last sentence in the last paragraph of Section 3.1 states: "The code version and cross section set used in the analysis **should** [emphasis added] be the same as those used in the validation of the codes." Revise the report to make this a requirement. Any exceptions would be treated as a deviation from the standard guidance and dealt with on a case by case basis.

**The following comments are on Section 4 of NEI 12-16:**

12. Expand the list of depletion-related parameters provided in Section 4.2 to include moderator density, fuel temperature, power density, axial and radial burnup distributions, post-irradiation cooling time, and fuel assembly design characteristics.

13. The first parameter listed as one of the "most important parameters" is the relative power during depletion. The NRC staff believes that the actual power density is more important than relative power as a focus on relative power may miss the impact that power uprates have on the nuclear criticality safety analysis.

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.

15. The “Soluble Boron during Depletion” paragraph states, “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].” The cited reference is misrepresented as it indicates using a burnup averaged soluble boron concentration value is likely conservative with respect to modeling the actual boron concentration changes for fuel assemblies that have completed the entire cycle. However, the reference also indicates that when the entire fuel cycle is not completed the use of a burnup averaged soluble boron concentration value is likely non-conservative with respect to modeling the actual boron concentration changes for fuel assemblies. Therefore, revise the report to:

    a. Provide guidance concerning how the burnup averaged soluble boron concentration should be calculated.

    b. Provide guidance concerning the potential non-conservatism of using a cycle average soluble boron concentration for a mid-cycle shut down and discharge of fuel to the SFP.

    c. Remove the mischaracterization of the cited reference.

16. Revise the subsection on “Burnable Absorbers” to provide more comprehensive guidance, defining the standard method, for modeling integral burnable absorbers, such as the Westinghouse Integral Fuel Burnable Absorber and $UO_2$ rods containing gadolinium or erbium, during depletion calculations. This guidance should include consideration of part-length absorber sections and reduced pellet density in $UO_2$ rods containing burnable absorbers. Provide justification for recommended methods.

    a. Revise this section to provide more complete guidance for modeling of removable burnable absorbers in depletion calculations. For example, how are part-length burnable absorbers handled? How are the axial spacer sections of burnable absorber rods modeled? What analysis techniques and practices are required for accurate or conservative burnable absorber rod depletion? Provide justification for the standard method.

    b. The 3rd paragraph under “Burnable Absorbers” includes the following text:

    > *If Gadolinium or Erbium is to be neglected, the volume averaged enrichment may be used in the criticality model. Recent analysis has shown this to be a conservative approach [31].*

    The analysis documented in Section 12 of Reference 31 is too limited to support adoption of a broad conclusion that use of the volume averaged enrichment is always conservative. Provide an appropriate justification or revise the recommended method to a more defensible approach. It should be clarified that it is not appropriate to apply volume averaging to axially-varying uranium enrichments.

17. Revise Section 4.2.1 to cover the following additional depletion related topics:

- Dimensions and materials used in depletion models.
- Depletion calculations for axial blankets.
- Modeling of water displacement above and below the active components in burnable absorber inserts, axial power shaping rods, and flux suppression inserts.
- Use of maximum, lattice-average, or rod-specific enrichments for depletion calculations.
- Depletion time or burnup step size.
- Use of lumped fission products during depletion calculations.
- Modeling of grids during depletion, including local flux/burnup depression.
- Modeling of bypass flow (i.e., in guide tubes and instrument tubes) water density
- Unusual depletion conditions such as extended part power operation and extended shutdown time between cycles.

18. The following comments are provided on Section 4.2.2, which covers "Fuel Assembly Physical Changes with Depletion:"

    a. From the second paragraph, it is indicated that the study was performed only for Westinghouse 15x15 fuel.

       The following text is provided near the top of page 10:

       > *"The behaviors exhibited by the pellet and clad are not specific to any reactor design, they are applicable to all $UO_2$ fueled plants."*

       The conclusion to Section 4.2.2, which states that "fuel geometry changes with depletion do not need to be explicitly modeled," is based on a limited study of a single Westinghouse 15x15 fuel design. Provide adequate justification supporting the applicability of the study results to other fuel assembly designs.

    b. The study presented in Section 4.2.2 is based on PAD code results. Is PAD validated and approved for this application? Are the PAD results best-estimate or are they conservatively biased to meet some other safety analysis need? For example, if PAD is designed to conservatively estimate maximum rod internal gas pressure or maximum fuel temperatures, the fuel density, pellet diameter, and clad dimension results may be non-conservative when applied to criticality analyses.

    c. How do the study results compare to irradiated fuel rod measurements (e.g., fuel assembly or fuel rod growth measurements)?

d. What depletion model assumptions (e.g., fuel enrichment, infinite lattice, assembly in reactor, water density, soluble boron concentration, etc.) were used for the Δk values provided in Table 4-1? For the criticality model, was the assembly modeled in a fuel storage rack? Provide a more comprehensive set of results considering more reactor depletion parameters.

19. Section 4.2.3, "PWR Depletion Uncertainty."

a. The section appears to be mixing "5% Δk" and "5% of the reactivity decrement," which are not the same. Revise this section clarifying the intent and provide appropriate justification and guidance to ensure consistent implementation.

b. The text in the first paragraph indicates that the referenced analyses "confirm" that a zero bias is appropriate. From a review of the data in Figure 3.1-2 in the cited Reference 34, there does appear to be a bias, as indicated by 70 percent of the data being above the 0 percent value. Additionally, the cited references are not independent as claimed in the report. Revise the text to eliminate the discussion of bias or provide a better justification for the statement.

20. Provide clear guidance in Section 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.

b. Define and provide guidance for the standard method(s) used for BWR spent fuel pool 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.

21. The next to last bullet in Section 4.3.1 states:

> *"All BWR criticality calculations should ensure a conservative reactivity is analyzed in the storage configuration with consideration given to possible cooling and discharge times."*

Change this to a requirement rather than a recommendation.

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.

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.

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. 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 percent 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.

**The following comments are provided on Section 5 of NEI 12-16:**

24. Revise Section 5 to include guidance on the standard method for modeling fuel assemblies. The guidance should describe and justify modeling simplifications and approximations.

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, 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).

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.

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. These include dimensions, axial location, pellet density, neutron absorber loading, and local variation of neutron absorber loading.

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

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

   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

27. Section 5.1.2 introduces the concept of insignificant uncertainties. The text includes the following:

   *"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."*

   From the text, it is not clear exactly what is proposed.

   Is the guidance to calculate all 95/95 uncertainties and then discard uncertainties that are less than 10 percent of the total? Or, is the guidance to ignore 95/95 uncertainties that are expected to be less than 10 percent of the total? If so, which uncertainties are to be ignored and under what circumstances? What calculations over what range of parameters were used to determine that these ignorable uncertainties are small enough? In this case, the justification currently provided in Section 5.1.2 is inadequate because the guidance does not provide or reference studies that look at variation with fuel assembly design, storage rack design, storage configuration, and environmental conditions. Revise Section 5.1.2 accordingly.

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.

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 in a checkerboard pattern with spent fuel or accident conditions with fresh fuel placed next to spent fuel.

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.

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 assemblies designs that vary significantly from those present in the database used to generate the NUREG/CR-6801 profiles.

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, and 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.

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

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.

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 Oak Ridge National Laboratory (ORNL) report, Reference 21, merely reported information from another reference and did not endorse use of the 2 percent value.

a. Revise the basis for using a value of 5 percent 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."*

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

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.

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.

    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.

    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.

    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.

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

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

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

    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.

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, sufficient calculations with detailed rack models should be performed to justify the simplified model as conservative or a conservative bias term should be used to cover the effects of the simplification.

    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.

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

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.

34. Regarding Section 5.2.2:

    a. expand the list of fuel storage rack uncertainties to be considered to include:

        - Poison panel length, width, thickness, and location.
        - Poison panel wrapper (sometimes referred to as poison panel sheath) thickness
        - Poison panel gap thickness
        - Rack poison insert thickness, width, length, poison loading, and location

    b. Add guidance requiring evaluation of tolerances/uncertainties over the range of fuel assembly designs and properties, storage configurations, and environmental conditions.

    c. While NEI 12-16, Reference 31, did provide some analysis of uncertainties associated with absorber sheath tolerance, the analysis was too limited (2D analysis of only two fuel assembly designs and two rack designs with no consideration of variation with water density, soluble boron concentration, etc.) to draw general conclusions related to ignoring manufacturing tolerances on sheath thickness. Consequently, provide additional justification for ignoring manufacturing tolerance on sheath thickness.

35. The last sentence in Section 5.2.2.1 claims that it is not necessary to evaluate temperatures (and presumably density) at any temperature other than the maximum and minimum allowed temperatures, citing NEI 12-16, Reference 31, as justification.

    While NEI 12-16, Reference 31, did provide some analysis of the variation of $k_{eff}$ with temperature, the analysis was too limited (2D analysis, only two fuel assembly designs, only two rack designs, etc.) to support a general conclusion that it is always acceptable to evaluate only the maximum and minimum temperatures.

    It seems that the basis for the general conclusion that it is acceptable to evaluate only the maximum and minimum temperatures comes from the fact that Doppler feedback in the fuel is always negative as NEI 12-16, Reference 31, points out. However, there is also a temperature effect on the hydrogen scattering cross sections. In some analyses, an increase in moderator temperature has a larger effect, increasing $k_{eff}$ more than the reduction in $k_{eff}$ associated with Doppler feedback in the fuel. This effect can be isolated and observed by simply changing the moderator and fuel temperatures while holding the moderator density constant.

Optimum moderation may be achieved for some rack designs with some fuel assembly designs at temperatures between the maximum and minimum temperature/density. Without further analysis, providing generic guidance that allows the analyst to simply assume there is no need to evaluate temperatures between minimum and maximum is not appropriate. Provide guidance to perform sensitivity studies to identify the most reactive moderator and fuel temperature and how to perform this analysis for the various fuel, rack, and storage configurations that require approval.

36. Section 5.2.2.3.1, “Boron Content,” discusses the modeling of the boron-10 ($^{10}B$) content of SFP neutron absorbing materials. The NRC staff disagrees with the guidance in this section. The standard practice should be that the $^{10}B$ content of the neutron absorbing material modeled in SFP NCS analyses is less than the minimum as-built $^{10}B$ content. Allowing use of the minimum as-built $^{10}B$ content is not necessarily conservative and provides no margin to allow for degradation or uncertainties in the monitoring program. Revise the guidance accordingly.

    a. Section 5.2.2.3 contains the phrase, “These racks incorporated neutron absorbers (typically containing boron)…”. However, there is no guidance on any neutron absorber other than $^{10}B$. Revise the guidance to be exclusive to $^{10}B$ or provide guidance for other neutron absorbers.

    b. Revise the text in Section 5.2.2.3.2 to include consideration of poison panel shrinkage, gap-formation, edge-erosion, bubble formation or anything else that may impact the poison panel’s ability to meet the credited criticality control function.

    c. Is modeling of degraded or modified poison panels considered to be a standard method? If not, clearly state as much. If so, provide guidance for modeling degraded/modified poison panels. Such guidance should describe and justify modeling assumptions, simplifications, and approximations used to model the degraded poison panel and address associated uncertainties.

37. Provide guidance that the criticality analysis report content is to include descriptions of models used for fuel depletion and for NFSV and SFP criticality calculations. Descriptions should include sketches or figures and cover dimensions and materials in enough detail for NRC staff to perform necessary confirmatory calculations.

**The following comments are provided on Section 6 of NEI 12-16:**

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.

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

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.

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.

39. Consider the following comments regarding Section 6.2, "Interfaces," and revise the report accordingly:

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

    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.

    c. Provide a definition for "storage configuration."

    d. The second sentence in the third paragraph of Section 6.2 specifies that the resulting $k_{eff}$ of the interface cannot be less than the $k_{eff}$ of the two individual storage configurations. However, the first bulleted criterion under that paragraph is that the calculated $k_{eff}$ for the interface is less than the $k_{eff}$ 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 $k_{eff}$ 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.

    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.

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.

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.

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.

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.

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.

**The following comments are provided on Section 7.**

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.

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

    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.

**The following comments are provided on Section 8 of NEI 12-16:**

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

43. In the list of biases, include:
    - Lattice code bias
    - Modeling simplification biases
    - Safety margin covering validation gaps/deficiencies
    - Other biases

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

**The following comments are provided on Section 9 of NEI 12-16:**

45. Revise Section 9.5, "Neutron Absorber Monitoring Programs," to clearly identify the key attributes of a neutron-absorbing material monitoring program and describe the actions licensees need to take to ensure the neutron-absorbing material monitoring program is effective in identifying degradation.

**The following comments are provided on Appendix Section A.1 of NEI 12-16:**

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.

47. Based on the content of Section A.1 and the lack of a separate section on spent fuel criticality code validation, rename Section A.1 to “Criticality Code Validation.”

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.

49. Revise this section to make it clear that the Mixed Oxide (MOX) experiments being referenced are necessary to validate the plutonium isotopes in depleted fuel and are not intended as a methodology for validating MOX fuel. As currently writing NEI 12-16 does not provide a justification for applicability to MOX fuel. Whether NEI 12-16 can be extended to MOX fuel will be determined at a future time if the need arises.

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

51. Provide additional guidance for selection of critical experiments to clarify that critical experiments from multiple facilities and experiment series should be selected in order to avoid a facility-specific or experiment series-specific systematic bias.

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

53. Section A.1.3 covers critical experiment modeling. Add text to this section to make it clear that the analyst is responsible for the accuracy of the critical experiment models and may not assume they are accurate, even if the models are obtained from a reputable source.

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

**The following comments are provided on Appendix Section A.2 of NEI 12-16:**

55. The section appears to be mixing “5% Δk” and “5% of the reactivity decrement,” which are not the same. Revise this section clarifying the intent and provide appropriate justification and guidance to ensure consistent implementation.

56. Section A.2.1.2 characterizes the larger relative uncertainty at low burnups as being due to the burnup binning strategy utilized by the Monte Carlo sampling approach in NUREG/CR-7108. This trend is not a result of the burnup binning strategy; rather, it is a result of a relative lack of destructive assay data at low burnups. Rewrite the discussion to clarify the source of the higher uncertainty.

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 percent bias for all nuclides, even though this bias was determined using a more limited set of nuclides.

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 Am-241 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 NEI 12-16 accordingly.

59. The final paragraph in Section A.2.1.2 suggests that, when ENDF/B-V or VII cross sections are used, the validation study for burned fuel does not need to include MOX experiments. This is not acceptable. Revise the text to remove this assertion.

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.

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 $k_{eff}$ value and its uncertainty;

   b. Definition of the benchmark models;

   c. Generation of the calculated $k_{eff}$ 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 percent probability and with a 95 percent confidence level.

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

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.

    b. Code-to-code comparaison 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.

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

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

**Comments and Requests for Additional Information on EPRI Report 3002003073, *Sensitivity Analyses for Spent Fuel Pool Criticality***

EPRI TR 3002003073 is intended to provide analyses supporting guidance in NEI 12-16.

1. If approved, TR 3002003073 would be part of the technical underpinning of guidance for performing safety related nuclear critical safety analyses for years to come. Describe the quality control measures taken to ensure that the analyses performed are reliable.

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 percent of the $UO_2$ maximum theoretical density (i.e., 10.5216 g $UO_2/cm^3$). Thus, the fresh fuel calculations had 1.8 percent 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.

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).

   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.

3. EPRI TR 3002003073 is intended to support application of certain simplifications as general practice for criticality safety analyses. However, the sensitivity studies were performed using a limited set of calculations that do not cover the full range of parameters encountered by analysts, such as the following:
   - Fuel assembly design specific characteristics, such as H/U ratio, uranium weight per assembly, pin pitches, fuel rod dimensions, etc.
   - Burnable absorber combinations
   - Axial variation in fuel assembly characteristics
   - Rack geometries and compositions
   - Fuel storage configurations, including fuel loading arrangements and use of reactivity control components other than in-rack neutron absorber panels

   - BWR specific considerations
   - Depletion parameters

   Expand the scope of the sensitivity studies to address significant differences that may exist in these parameters relative to the studies that were performed. Whenever general recommendations are made, provide a technical justification for the applicability of the sensitivity study results or limit the range of applicability for the recommendation.

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.

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 percent power for final 15 days). Describe how this model compares with actual end-of-cycle operations. Provide the in-rack $\Delta k_{eff}$ 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.

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.

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.

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?

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

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.

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?

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.

    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.

    c. No reference is provided for the CE clad inner diameter tolerance. Reference 15 states a tolerance on the outer diameter 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.

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.

b. The text notes that the "selected" grid volume is 2 percent 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 percent of volume an assembly average value or a local average at the grid locations?

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 percent grid volume assumption.

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.

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.

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.

    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.

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.

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.

15. Section 9 discusses the impact of eccentric (i.e., non-centered) fuel placement in fuel storage racks. This section presents studies and recommendations related to treatment of the reactivity impact due to eccentric fuel placement. In several places, the report suggests the possibility of treating part of the reactivity effect due to eccentric positioning as an uncertainty. No methodology is presented, or is currently approved, that demonstrates if the 95/95 threshold required by 10 CFR 50.68 is met for a minimum number of eccentrically positioned fuel assemblies. Until such a methodology is developed, the eccentric positioning should be treated as a bias. Please rewrite the discussion to make this clear.

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.

    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.

    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.

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

    a. Revise the $2^{nd}$ 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.

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 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.

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.

    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.

    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.

    d. The 6th bullet discusses the conclusion related to the study of small sets of assemblies. As discussed in RAI Question No. 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.

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(f)(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 8th 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 $k_{eff}$. 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 <u>Criteria</u> 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 <u>Burnup</u> Confirmation.” Include the ORNL report number, which is ORNL/TM-2007/229.

j. Reference 23 – Revise the title from “Human Reliability...” to “<u>Preliminary, Qualitative</u> 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.