UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 29, 2015

Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE SPENT FUEL POOL CRITICALITY ANALYSIS (TAC NO. MF5282)

Dear Sir or Madam:

By letter dated November 13, 2014, Entergy Nuclear Operations, Inc. (the licensee), submitted NETCO Report NET-300067-01, "Criticality Safety Analysis of the Indian Point 2 Spent Fuel Pool with Credit for Inserted Neutron Absorber Panels" (Agencywide Documents Access and Management System Accession No. ML14329A194), for review and approval by the U.S. Nuclear Regulatory Commission (NRC).

The NRC staff is reviewing the submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). Based on our discussions we understand that a response to the RAI will be provided within 60 days of the date of this letter.

Please contact me at (301) 415-1364 if you have any questions on this issue.

Sincerely,

Douglas V Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosure:
Request for Additional Information

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REQUEST FOR ADDITIONAL INFORMATION

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

SPENT FUEL POOL CRITICALITY ANALYSIS

DOCKET NO. 50-247

**Absorber Panel Design**

1. Section 3.4, "Absorber Panel Design," credits "a double panel at the [Region 1/Region 2] interface." Upon installation of the absorber panel inserts, how will it be ensured that the installed insert orientation at the Region 1/Region 2 interface, and elsewhere, is consistent with the modeled orientation credited in the criticality safety analyses? Furthermore, how will proper panel installation be confirmed given that there are three possible panel types with region-specific dependency?

2. A footnote to Table 3.5, "Absorber Panel Dimensions," regarding alternate absorber panel adjustments states: "Minor adjustments to these specific dimensions and areal densities are acceptable provided that the panel is shown to be as effective in absorbing neutrons as the primary design." This statement implies that a calculation or calculations will be performed to demonstrate the effectiveness of any minor adjustments to the alternate absorber. Provide clarification regarding how the effectiveness of the final alternate absorber panel will be demonstrated if adjustments to the values in Table 3.5 are made. Also, the notes to Table 3.5 indicate that some values will be restricted to a minimum or maximum value. Will the minor adjustments be consistent with these notes?

3. Section 3.4 states that "all of the loading curve calculations were performed with the primary design," and for the alternate design, the minimum areal density had to be increased relative to the primary design "so that the loading requirements would remain the same." Was the Region 2 alternate panel areal density confirmed to be valid over the entire burnup loading curve range? Also, there are missing footnotes in the second to last paragraph of Section 3.4 for the alternate absorber panel areal densities. Provide the referenced footnotes or remove the references if they were unintentional.

**Code Validation**

4. The last sentence of Section 5.1, "Limiting Depletion Parameters – Burnable Absorbers," states: "If gadolinium or erbium is used in the future, then this criticality analysis is valid." How does the current critical experiment benchmarking analysis cover future erbium or gadolinium burnable absorber credit in spent fuel pool (SFP) criticality safety calculations?

5. In Section A.2.5, "Statistical Analysis of the Fresh UO2 Critical Benchmark Results," the equations for the soluble boron and boron areal density (B-10) trend lines are shown to be identical. Please correct the apparent error and confirm that the uncertainty treatment for the B-10 areal density trend is correct since the limiting criticality code validation uncertainty is based on the areal density trend analysis.

Enclosure

6. Referring to lattice characteristics in Table A.7, "Area of Applicability (Benchmark Applicability)," the comment mentions that "the expected range of all fuel types, including both pressurized water reactors (PWRs) and boiling water reactors (BWRs) fuel is covered." Why is BWR fuel included in the table if BWR fuel is not stored at Indian Point 2?

7. The assessment of the mixed uranium/plutonium oxide (MOX) critical experiments in Section A.3.2, "MOX Critical Experiments," concludes that the average uncertainty weighted k-effective of the fresh UO2 critical experiments is less than that of the MOX experiments implying that the fresh fuel critical experiments should be used as the basis for the criticality code bias and bias uncertainty for all the criticality safety analyses. However, the uncertainty for the MOX critical experiments was not reported. Provide the limiting uncertainty from the MOX critical experiment trending analysis, and the uncertainty for the set as a whole, as was done for the fresh fuel critical experiments and confirm that the criticality code bias and bias uncertainty determined from the fresh fuel critical experiments remains bounding for calculations containing spent fuel.

**Depletion Calculations**

8. Section 2.1 discusses a FORTRAN code that is used in addition to SCALE "to interpolate between burnups from the OPUS output and also to decay the isotopic content to the desired cooling time." It is not clear how isotopic content as a function of cooling time is calculated from the discussion in Section 2.1. In Section 6.4, "Interpolation of Isotopics and Cooling Time Verification," more discussion is provided; and it is implied that isotopics are decayed internally within the FORTRAN code without reliance on SCALE/ORIGEN-S.

    a. Is the FORTRAN code managed under a quality assurance program that meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50, Appendix B?

    b. Provide a description of the cooling time model used in the interpolation program and explain why SCALE isn't used directly for isotopic decay calculations.

    c. Section 6.4 provides verification of the cooling time model in the FORTRAN code for only four cases. Explain why this sample of cases provides assurance that the FORTRAN code will conservatively calculate k-effective relative to SCALE for the burnup and enrichment domain covered by the burnup loading curve.

9. Section 5.4, "Limiting Depletion Parameters – Specific Power," states that nominal specific power is used. However, in Section 5.6, "Summary of Depletion Assumptions for Fuel = 3.5 wt percent U-235," Item c. of the first paragraph states that a higher than nominal specific power is used during depletion. Please correct the apparent discrepancy.

10. Section 5.8, "Depletion Analysis Details (Time Steps, etc.)," briefly discusses a few important depletion analysis details.

    a. What type of modeling guidance is used when performing depletion calculations using SCALE/TRITON? For example, NUREG/CR-7041, "SCALE/TRITON Primer: A Primer for Light Water Reactor Lattice Physics Calculations," describes in detail

some best practices regarding use of SCALE/TRITON modules that would be applicable to the use of SCALE/TRITON for criticality safety applications.

b. Provide SFP k-effective comparisons by using depleted fuel isotopics from the SCALE/TRITON module used to support Indian Point 2 criticality safety calculations and a depletion code that has been approved by the NRC for use at Indian Point 2 to provide assurance that SCALE/TRITON depletion modeling is being performed appropriately. Include a range of cases that are representative of the range of fuel depletion conditions at Indian Point. For example, include cases that model control rod, Pyrex, WABA, and IFBA depletion. Since it is stated that gadolinium and erbium may be used in the future, also include comparisons with representative gadolinium and erbium use.

11. Is the use of SCALE, for depletion and criticality calculations, managed under a quality assurance program that meets the requirements of 10 CFR 50, Appendix B?

12. Section 8.4, "Depletion Effect of Hafnium Flux Suppression Inserts," describes specific analyses that define the basis for the penalty to be applied to corresponding minimum burnup requirements for spent fuel storage of assemblies that were depleted with hafnium inserts. There is no discussion regarding why it was assumed that the hafnium inserts are only depleted for the 8 GWd/MTU and not more, why only enrichments greater than 4 percent were considered, and why the two cooling times chosen were considered. Furthermore, hafnium insert usage is not part of assumption verification during the reload design process as indicated by Table 10.6, "Fuel Assembly Operating Requirements for Fuel Enriched > 3.5 wt%." How do the hafnium insert modeling assumptions bound past, present, and future hafnium insert use at Indian Point 2? Discuss any other inserts that have been used at Indian Point 2. Also recognize that the hafnium insert penalty credited in Note (a) and (b) to Table 10.2, "Region 2 Minimum Burnup (GWd/T [gigawatt-days/ton]) Requirements" is based on these very narrow set of depletion conditions and, therefore, it wouldn't be appropriate to apply these penalties to all possible hafnium insert depletion scenarios without further justification; likewise for footnotes to Table 10.3, "Summary of Loading Restrictions," regarding use of hafnium inserts.

**Criticality Calculations**

13. Section 2.1, "Computer Codes," and Section 6.5, "Convergence of Calculations," gives the number of neutron generations and neutrons per generation used, but not the number of skipped generations. How many skipped generations were used and what starting source distribution was used? Include justification for these assumptions.

14. Section 2.1 describes a process by which various nuclide concentrations are manually changed for the first 72 hours of decay after fuel depletion instead of using the SCALE/TRITON module to perform this 72-hour decay for all credited nuclides. Provide verification that manually changing nuclide concentrations is conservative relative to performing the calculation with SCALE/TRITON.

15. Section 3.3, "Fuel Assembly Insert Designs," describes control rod modeling "for the special case of crediting a control rod in a fresh fuel assembly in the pool" where "the Ag-In-Cd content (density) is reduced by 20%."

a. Are the same assumptions applied to all criticality calculations crediting control rods (e.g. burned fuel that doesn't meet the 12 gigawatt-days/metric ton of uranium (GWd/MTU) burnup requirement in Region 1 or Region 2 control rod credit)?

b. Provide more detail describing how the 20 percent reduction bounds manufacturing tolerances, any absorber material loss during operation, and any modeling assumptions or simplifications, given that Table 8.1, "Calculated k's in Region 1," shows that the minimum margin case for Region 1 occurs with a case crediting a control rod in the fuel assembly.

c. Are there different control rod designs or variations in designs available onsite or will there be in the future?

16. Section 6.3, "Axial Burnup Distribution," mentions that "the lower 10 nodes were averaged into one node" when accounting for the axial burnup distribution in criticality calculations. Were calculations performed to confirm that this modeling simplification does not affect the k-effective calculation? If not, provide confirmation that this simplification is conservative.

17. Table 7.1, "Tolerance Reactivity Effects," indicates that only positive fuel pin pitch changes were considered when determining the fuel pin pitch tolerance reactivity effect. Please verify that negative fuel pin pitch changes do not result in larger reactivity effects.

18. Section 8.1 discusses criticality calculations involving fresh fuel crediting integral fuel burnable absorber (IFBA). Due to the large flexibility allowed by SCALE regarding cross-section processing options and geometry modeling, which could significantly impact k-effective estimation, describe how the IFBA is modeled and provide a sample SCALE input deck for a case crediting IFBA.

19. In Table 8.1, explain why the k-effective of 0.9717 doesn't match the corresponding k-effective of 0.97182 in Table 8.10 given that these are supposed to be the same case.

20. The following statements in Section 8.2.1, "Curve Fit," regarding the curve fitting process of the data in Table 8.2, "Minimum Burnup Requirements (GWd/T) in Region 2" are misleading:

    *The coefficients contain an adjustment to ensure that all burnups calculated by the equation are greater than the burnups from the table. Using the curve fit results in a maximum penalty of 0.2 GWd/T for low enrichments and 0.4 GWd/T for high enrichments when compared to the tabulated values shown in Table 8.2.*

    Confirmatory analysis shows that the defined fits can actually be non-conservative between 3 percent and 3.5 percent enrichment for cooling times of zero and one year, and for enrichments between 4.5 percent and 5 percent for cooling times of 15 and 25 years. The confirmatory analysis shows that approximately 10 percent of burnups are non-conservative with respect to the non-fitted loading requirements, defined by linearly interpolating between points in Table 8.2, up to a maximum of approximately 0.4 GWd/MTU. Please revise the discussion and/or update the fits accordingly.

21. Section 8.6, "Volatile Fission Gases," estimates the reactivity effect associated with a 10 percent release of volatile fission gases from the spent fuel citing Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," as the basis for using 10 percent. However, the SFP environment is not the focus of RG 1.183, and fuel in the SFP can be stored there for several decades or more. Consequently, it is not clear that the 10 percent value taken from RG 1.183 would be applicable to long term storage in a SFP environment. Please re-consider the volatile fission gas release fraction assumed in the criticality safety analyses taking into account long term storage of spent fuel in the SFP.

22. In Section 8.10.2, "Results of Reduced Periphery and Region 1/Region 2 Interface Analysis," regarding the pool reflector sensitivity studies, it is stated that "the concrete used is a conservative mixture created by Oak Ridge -- named as *orconcrete*." Explain why this concrete is considered to be a conservative mixture.

23. In Section 8.11, "Failed Fuel Canisters," an analysis is described where "36 fuel pins could be loaded into each failed fuel container with no criticality concern." This conclusion is valid given strict control on the configuration that was analyzed. What controls will ensure that any failed fuel canisters will remain in the geometry analyzed? Alternatively, what will be the actual range of variation of fuel pin geometry allowed in the failed fuel canister and why does the analysis presented bound all potential fuel pin geometry variations?

24. Provide more discussion regarding Section 8.12, "Fuel Rod Storage Basket," to address the following:

    a. How is geometry controlled?

    b. Are there spaces where other fuel rods could be placed other than the analyzed configuration?

    c. Why isn't a missing fuel rod analysis necessary?

    d. Is it possible that another movable fuel rod storage basket design could be introduced at the plant in the future?

25. In Section 8.13, "Assemblies with Missing Fuel Rods," it was not stated if an analysis was done to verify that the burnup worth associated with 4 GWd/MTU burnup adder to cover reactivity increases for an assembly with any number of missing rods is appropriate, nor was it stated what burnup the modeled reconstituted fuel assemblies were analyzed at. Consequently, provide justification for the 4 GWd/MTU burnup adder to cover reactivity increases for an assembly with any number of missing rods at any burnup, given that burnup worth changes as a function of burnup.

26. Section 8.13 implies that the 4 GWd/MTU adder will only apply to reconstituted fuel assemblies that do not install stainless steel rods -- this means that an additional check would be necessary to indicate that the 4 GWd/MTU adder is needed in some cases, but not for others, creating implementation complexity. Consequently, what controls ensure that the 4 GWd/MTU will be added to the fuel assembly burnup for applicable fuel assemblies before comparison to minimum burnup requirement? That is, how will it be ensured that a reconstituted fuel assembly where stainless steel rods are not installed (i.e.

requiring a burnup adder) can be distinguished from a reconstituted fuel assembly where stainless steel rods are installed (i.e. not requiring a burnup adder), and then be ensured that the fuel assembly burnup is adjusted appropriately for comparison to the corresponding minimum burnup requirement?

27. The missing fuel rod sensitivity study discussed in Section 8.13 does not seem to have analyzed the sensitivity to missing fuel rod orientation for the various cases analyzing different numbers of missing fuel rods. Provide justification that the selected orientations are bounding relative to other potential orientations.

**Accidents**

28. The criticality safety analysis needs to be consistent with existing administrative controls credited in future license amendment requests. Currently, there is no control rod removal accident analyzed. If there are no explicit controls for fuel assemblies crediting the presence of control rods, this accident should be considered for both Region 1 and Region 2. Missing panels are also part of normal operation as shown in Section 8, "Results." Consistent with DSS-IGS-2010-01, "accidents should be considered with respect to all normal conditions, e.g. fuel inspections and fuel reconstitution." If there are no explicit controls for missing panels, then all accident analyses should be updated to account for the maximum number of missing panels allowed during normal operation in both Region 1 and Region 2.

    Provide justification for not including a control rod removal accident, which could include more than one control rod for a cluster of assemblies in either Region 1 or Region 2. Include a discussion of controls that are, or will be, in place at Indian Point 2 to ensure that credited control rods cannot be removed. Alternatively, model the maximum number of allowed missing panels, a normal condition of operation, as a base condition for a control rod removal accident.

29. The claim that no checkerboarding is credited as part of the justification for why a multiple misload is unlikely in Section 9.4, "Multiple Misloads," isn't accurate given that Footnote (f) to Table 10.1 states: "...a Region 2 cell which does not contain an absorber panel does not affect the loading requirements of any other cell in Region 2, so long as the cell which is missing an absorber panel does not contain a fuel assembly...". This operational flexibility would allow for checkerboarding of fuel and also one-out-of-four storage with the empty cells containing no absorber panels. Please revise the statement in Section 9.4 and/or other related statements accordingly.

**Reactor Operating Limits and Allowable Fuel Loading Checks**

30. Note (b) to Table 10.2 and the footnotes to Table 10.3 are confusing as written. An analysis was performed in Section 8.4 accounting for fuel assembly depletion with hafnium inserts and it states that: "For simplicity and to provide margin, a 2 GWd/T increase in the loading curve is required *for any assembly having any burnup with a hafnium insert* [emphasis added]." Based on this statement, if a fuel assembly contained a hafnium insert at any time, why does the amount of burnup a fuel assembly has achieved factor into the application of the 2 GWd/MTU burnup requirement penalty as specified in Note (b) to Table 10.2 and the footnotes to Table 10.3?

31. In Section 10.5, "Reactor Operation Limits," there is a footnote that states: "If fuel less than or equal to 3.5 wt% is ever used in the future, the same requirements shown in Table 10.6 apply. Table 10.6 should be used for all future fuel regardless of enrichment." However, the burnup requirements in Table 10.2 are based on different sets of depletion conditions that depend on enrichment. Therefore, why is it acceptable to treat all lower enrichment fuel, specifically newer fuel, with depletion conditions specified in Table 10.6 given that Table 10.2 doesn't account for the Table 10.6 depletion conditions for lower enrichment fuel?

June 29, 2015

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***/RA/***
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
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