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

August 10, 2017

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 NOS. 2 AND 3 – REQUEST FOR ADDITIONAL INFORMATION REGARDING INTER-UNIT TRANSFER OF SPENT FUEL LICENSE AMENDMENT REQUEST (CAC NOS. MF8991 AND MF8992)

Dear Sir or Madam:

By letter dated December 14, 2016, as supplemented by letter dated April 19, 2017, Entergy Nuclear Operations, Inc. (the licensee) submitted a license amendment request to revise the Appendix C Technical Specification (TS) Limiting Condition for Operation (LCO) 3.1.2 for Indian Point Nuclear Generating Unit Nos. 2 and 3 (Indian Point 2 or Indian Point 3), and Appendix A TS LCO 3.7.13 for Indian Point 2 in order to increase the population of Indian Point 3 fuel eligible for transfer to the Indian Point 2 spent fuel pit and maintain full core offload capability for Indian Point 3.

The U.S. Nuclear Regulatory Commission staff is reviewing the licensee's 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). The RAI was discussed with your staff during a clarification call on August 3, 2017. Based on our discussions, we understand that a response to the RAI will be provided by September 18, 2017, which is 45 days from the date of the call.

Please contact me at (301) 415-1030 or Richard.Guzman@nrc.gov if you have any questions regarding this request.

Sincerely,

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-247 and 50-286

Enclosure:
Request for Additional Information

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST REGARDING INTER-UNIT TRANFSER OF SPENT FUEL

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3

DOCKET NOS. 50-247 AND 50-286

By letter dated December 14, 2016 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML16355A066), as supplemented by letter dated April 19, 2017 (ADAMS Accession No. ML17114A467), Entergy Nuclear Operations, Inc. (the licensee) submitted a license amendment request (LAR) to revise the Appendix C Technical Specification (TS) Limiting Condition for Operation (LCO) 3.1.2 for Indian Point Nuclear Generation Unit Nos. 2 and 3 (Indian Point 2 and Indian Point 3), and Appendix A TS LCO 3.7.13 for Indian Point 2 in order to increase the population of Indian Point 3 fuel eligible for transfer to the Indian Point 2 spent fuel pit and maintain full core offload capability for Indian Point 3. The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the licensee's submittals and has determined that additional information, as listed below, is needed to complete its review.

RAI-1 (CSTB-Thermal)

Clarify how the Appendix C TS LCO 3.1.2.b assures the shielded transfer canister (STC) total heat load is not exceeded. LCO 3.1.2.b, as shown in Attachments 2 and 4 of the LAR, specifies the fuel assembly maximum decay heat. However, it is not clear how the STC total heat load is met when only Type 2 fuel is loaded. Additional assurance may be provided by also specifying the STC total decay heat in LCO 3.1.2.b, as it is done in LCO 3.1.2.a.3.

Regulatory Basis: This information is needed to determine compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Sections 72.122 and 72.128.

RAI-2 (CSTB-Thermal)

Explain how the per-cell decay heat and total per-region decay heat would meet the STC total decay heat requirement specified in the LAR. Table 3.2, "Loading Scenarios Adopted for Thermal Evaluations of Holtec Report HI-2084146," in Enclosure 3 of your LAR provides a representative heat load pattern used to perform sensitivity analyses to show the effect on predicted temperatures and pressures. However, it is not clear from the values shown in this table how the STC total decay heat requirements are met. For example, adding the total heat load for Regions 1 and 2 of Loading Scenario 2 would result in a value that would exceed the specified limit for the STC.

Regulatory Basis: This information is needed to determine compliance with 10 CFR 72.122 and 10 CFR 72.128.

RAI-3 (CSRB-Shielding/Radiation Protection)

Confirm the following:

a. The dose rate and dose information for the Independent Spent Fuel Storage Installation (ISFSI) in the safety analysis report (SAR) analyses continue to bound the dose contributions of the ISFSI, modifying the SAR analyses, as needed. More storage casks have likely been added to the ISFSI since the original amendment to add the STC operations, which could increase the ISFSI contributions to doses. This, in turn, could affect the SAR dose analyses.

b. The activity values shown in Table 7.2.9 are the total burnable poison rod assembly (BPRA) activities that are used in the shielding analysis. If they are in-core activities only, also provide the total BPRA activities and ensure the analyses use the total BPRA activities.

c. Regarding the purpose and use of Appendix G of the shielding calculation package, this appendix was not part of the calculation package submitted with the previous amendment application dated July 8, 2009 (ADAMS Accession No. ML091940176), for the STC operations. Its purpose and use in terms of the shielding analysis is not clear and should be explained.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 10 CFR 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-4 (CSRB-Shielding/Radiation Protection)

Confirm the Indian Point 2 and Indian Point 3 TSs in Appendix C for allowable minimum cooling time and maximum burnup for BPRAs will remain unchanged for the proposed new STC loading patterns.

The shielding analysis in the SAR and the supporting shielding calculation package assume that the cooling time for the BPRAs will be the same as the fuel assembly in which the BPRA is loaded. The analysis also uses burnups of 40 gigawatt-days per metric ton of uranium (GWd/MTU) for those BPRAs in fuel assemblies with burnups not exceeding 40 GWd/MTU, and 60 GWd/MTU for those BPRAs in fuel assemblies with burnups in excess of 40 GWd/MTU. It appears that these parameters are intended to specify the allowable BPRA contents for the proposed new STC loading patterns. However, the TSs related to BPRAs (see Table 3.1.2-2 in Appendix C of the licenses) have not been changed and do not allow for loading of BPRAs with burnup and cooling times that fit the above-stated descriptions for three of the loading patterns. In particular, the BPRAs loaded into STC basket cells 1 through 4 of loading patterns 8, 11, and 12 are limited to 50 GWd/MTU, 20 GWd/MTU, and 40 GWd/MTU, respectively. Even for loading patterns 7 and 10, BPRAs in the same four basket cells would be limited to 50 GWd/MTU versus the 60 GWd/MTU considered per the SAR descriptions. A BPRA with a burnup of 60 GWd/MTU must have a minimum cooling time of not less than 11 years.

Additionally, per the parameters used in the shielding analysis, BPRAs in cells 1 through 4 of loading pattern 11 could have a burnup up to 60 GWd/MTU. This would appear to be unacceptable since the maximum cobalt-60 activity for such a BPRA with only 6 years of decay would exceed the maximum activity level previously analyzed and used in the shielding and radiation protection analyses. Any proposed changes to the TSs should be supported or bounded by the analysis in the application and clearly confined to the proposed new loading patterns.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 10 CFR 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-5 (CSRB-Shielding/Radiation Protection)

Clarify the cobalt impurity levels used in the modified analysis for:

a. The Inconel components in the spent fuel assembly's hardware and

b. The steel and Inconel components of the non-fuel hardware to be loaded with the spent fuel assemblies.

The modified analysis describes a change to the cobalt impurity level in the steel components of the fuel assembly hardware. However, it is not clear if the analysis changed the impurity level for Inconel components of the assembly hardware. Also, it is not clear that the analysis uses the same impurity level for steel and Inconel components of the non-fuel hardware such as BPRAs. The SAR describes crediting different decay times for the new BPRA analysis for the difference in BPRA cobalt-60 activities. However, the decay times in Table 7.2.9 overlap with decay times in Table 7.2.8 for which the cobalt-60 activity was previously evaluated to be 895 curies. Thus, the decay time does not account for the difference in cobalt-60 activity at the decay times listed in Table 7.2.9, since the activity is smaller at shorter decay times than was previously analyzed. While the SAR continues to state that the cobalt in the non-fuel hardware is taken to be 1.2 grams to kilograms (g/kg) in steel and 4.7 g/kg in Inconel, the analysis information does not appear to be consistent with this statement. Note that the differences in activity level for BPRAs are not consistent with an 0.5 g/kg impurity level either.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-6 (CSRB-Shielding/Radiation Protection)

Modify the analyses to address the uncertainties associated with calculations of source terms for high burnup fuel.

The licensee uses an older version of SCALE (version 4.3) and a module of that code, SAS2H, which is no longer supported by the code developer (Oak Ridge National Laboratory). The NRC staff accepted the use of this code version and code module in the previous analyses for the STC and the proposed contents, which included a limited amount of high burnup spent fuel. The NRC staff's safety evaluation for that license amendment dated July 13, 2012 (ADAMS Accession No. ML121230011), however, indicates that it accepted the analysis with code version and code module, in part, because the burnups did not extend significantly into the high burnup regime. Also, high burnup fuel was limited to the inner STC basket locations of only two loading patterns. Burnups for the proposed additional loading patterns extend further into the high burnup regime, and high burnup fuel may be loaded into inner or outer STC basket locations for multiple loading patterns. Thus, the licensee should address the uncertainties in the analyses for calculating high burnup fuel source terms with the code version and code module used in the analyses. One possible approach would be to evaluate the uncertainties and adjust the analyses in a manner similar to what has been done for the certified HI-STORM 100 dry storage system. Uncertainties in terms of radiation source terms and decay heats should also be addressed.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-7 (CSRB-Shielding/Radiation Protection)

Justify the use of proposed contents burnup, enrichment, and cooling times that result in decay heats that exceed the proposed decay heat limits, adjusting the proposed burnup, enrichment, and cooling time specifications, as needed.

Based on Table 7.1.1, specifications for fuel assemblies in the inner four STC basket locations in loading pattern 12 result in assembly decay heats that exceed the allowed decay heat limits for those basket locations. The proposed burnup, enrichment, and cooling time specifications should also ensure the decay heat limit is met, considering the contribution from non-fuel hardware, as well as the uncertainties in calculations of decay heat and radiation source terms for high burnup fuel (see preceding request for additional information (RAI)). This concern should similarly be addressed in terms of the limit for the total STC decay heat load since all of the new proposed loading patterns result in decay heats that exceed this limit. An alternative approach would be for the licensee to modify the operations descriptions in Chapter 10 of the SAR to ensure that verification of acceptability of assemblies and non-fuel hardware for loading in the STC include verification that the individual decay heats and the cumulative basket decay heat meet their respective limits. In other words, the verification of the acceptability of assemblies, as described in Chapter 10 of the SAR, should include more than just verification of the assembly burnup, enrichment, and cooling time. The operations should include verification of decay heat of each assembly, as well as the entire STC.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-8 (CSRB-Shielding/Radiation Protection)

Provide information to demonstrate that the current HI-TRAC dose rates are bounding for the additional loading configurations for both normal conditions and accident conditions.

Based on information in the SAR and shielding calculation package (e.g., Tables 23 through 29 of the calculation package), neutron dose rates are a significant or dominant component of the total dose rates for the HI-TRAC for both normal and accident conditions. Additionally, with higher burnups such as those in the proposed loading configurations 7 through 12 (particularly those with high burnup fuel in the outer basket locations), the neutron source term increases significantly versus the source terms in the configurations currently identified as yielding bounding dose rates. Thus, it would appear that the HI-TRAC dose rates should be updated to address the new loading patterns and the evaluations versus the regulatory limits revised. Sample input files may also be useful in responding to this question.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-9 (CSRB-Shielding/Radiation Protection)

Confirm that dose rates from BPRAs are calculated correctly for the proposed loading configurations, revising the calculations, as needed.

Based on the information in the SAR and shielding calculation package, the source terms for BPRAs for all the new proposed loading patterns should be for exposures of 60 GWd/MTU since all of the proposed loading pattern limits are for fuel with burnups exceeding 40 GWd/MTU. Thus, the dose rates due to BPRAs for the new loading patterns should vary from the BPRA dose rates in the design-basis calculations in proportion to the ratio of the cobalt-60 activities determined in Table 7.2.9 of the SAR versus the design-basis activity of 895 curies. Based on the information in Appendix A of the calculation package and Table 7.4.2 of the SAR, the variation in BPRA dose rates at 1 meter from the STC top surface is consistent with this expected change. However, the BPRA dose rates at 1 meter from the radial and bottom surfaces are not consistent with this expectation; they are substantially below values that would be consistent with this expectation (about 1 percent or less of the expected values). Thus, it appears there is an error in the analyses for BPRA dose rates, at least for the noted locations, and possibly for others. Sample input files for the analyses for the previously approved STC amendment and sample input files for the proposed, modified analyses may be helpful in responding to this question.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-10 (CSRB-Shielding/Radiation Protection)

Provide information to demonstrate that the proposed change to the analysis for control rod assemblies (RCCAs) is adequately bounding for actual operations of RCCAs that are not typical, revising the dose rate calculations, as necessary.

The licensee proposed to modify the source term calculation for RCCAs based on typical operating parameters for RCCAs at Indian Point 3. However, the analysis should be bounding for RCCAs, which may have experienced operations that are not consistent with typical operations at the plant (e.g., operated with some insertion into the core for one or more periods of time or use as regulating rods), if any. Since the information in the SAR only discusses typical RCCA operations, it is not clear how the proposed analysis change considers RCCAs that have experienced non-typical operations.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-11 (CSRB-Shielding/Radiation Protection)

Modify the evaluation of measured dose rates versus calculated dose rates for the two STC loads described in Appendix I of the shielding calculation package to address the following:

a. Calculate dose rates using burnup, enrichment, and cooling time values for the assemblies that are closest to the measurement location for which the dose rate comparison is made.

b. Compare dose rates for locations that are adequately defined to know where on the STC the measurement was made and where the dose rates should be calculated for accurate comparison.

c. Calculate dose rates for burnups and cooling times of the non-fuel hardware that are consistent with the burnups and cooling times of the hardware in the loaded STC.

d. Clarify the descriptions of the measurements to (1) identify the assemblies that were closest to the measurement locations, (2) identify the location of the measurement at the STC base (e.g., at the center of the base or the relative location versus the center or the inner region of the STC basket), (3) confirm the measurements are surface measurements (versus at distance from the STC surfaces), and (4) provide the (estimated) uncertainties in the measurements.

The comparisons in Appendix I of the calculation package are used to demonstrate the degree of conservatism that remains, even with the changes to the source terms for the spent fuel assemblies and non-fuel hardware specifications for the proposed loading patterns. However, it is not clear that the current evaluation is adequate for that purpose. Based on the information provided in the appendix, it is not clear that the calculations are for the same locations on the STC as the measurements. At least in one instance, it is not clear where the measurement was taken that is used in the comparison (see Table I.2.B). Significant variations in dose rates are expected, depending upon the measurement locations.

Also, the conservative nature of the analysis method should be demonstrated with calculations that use the actual burnups, enrichments, and cooling times for the contents (both the spent fuel and the non-fuel hardware). Differences in burnup, enrichment, and cooling time of assemblies versus the region average used in the calculation can influence the determination of the degree of conservatism that may or may not be present in the analysis method. This may be particularly true for the assemblies in the outer STC basket region, especially for STC #1. Thus, the licensee should use the burnups, enrichments, and cooling times of the assemblies that are closest to the measurement locations to determine dose rates for comparison with measured dose rates. Alternatively, the licensee may represent the assemblies in the region using the assembly that was loaded in that region of the STC and has burnup, enrichment, and cooling time parameters that result in the weakest radiation source term (and thus the lowest dose rates). In addition, for the non-fuel hardware, the source term should be derived from burnups and cooling times of the loaded hardware and not the design-basis source terms. If design-basis non-fuel hardware were loaded in the STC (as allowed by the TSs), then use of the design-basis hardware source term would be appropriate for this analysis.

Regulatory Basis: This information is needed to confirm compliance with 10 CFR 20.1101(b) and 20.1301(a), (b), and (e) and the intent of 10 CFR 72.104, 10 CFR 72.106(b), and 10 CFR 72.126(a).

RAI-12 (CSRB-Criticality Safety)

Clarify which part(s) of the Chapter 4 SAR has been changed and provide a revised SAR, if necessary, to clearly mark the changes, and provide an evaluation of the impact of the changes on criticality safety of the STC.

On page S7 of the LAR dated December 14, 2016, the licensee states, "All of Chapter 4 has significant changes due to the change in criticality methodology from Part 50 to Part 71. This is summarized in the preface to chapter 4 responses in the response to the RAI." The licensee further states on page 14 of Attachment 1: "New criticality evaluations for both the STC confirm that operation in accordance with the proposed amendment continues to meet the required subcriticality margins." However, only a very small portion (shown only on pages 4-43 and 4-57) of Chapter 4 of the SAR is marked with a revision bar. As such, the NRC staff was unable to identify the changes in the SAR and new criticality evaluations; consequently, the staff is unable to evaluate the impact on criticality safety. The licensee is requested to provide a revised SAR in which the changes are clearly marked. The licensee is also requested to add an evaluation of the impact of the changes, item by item, on criticality safety of the STC.

Regulatory Basis: This information is needed to determine if the Indian Point spent fuel inter-unit transfer canister STC design meets the regulatory requirements of 10 CFR 72.124(a) and 10 CFR 72.124(b).

RAI-13 (CSRB-Criticality Safety)

Revise Table 3.1.2-3, "Allowable STC Loading Configurations," to include the upper enrichment limit and the required minimal burnup for each of the proposed configurations that takes burnup credit.

On page 3.1.2-7 of the proposed new loading table, Table 3.1.2-3, the licensee specifies minimum enrichment and maximum burnup for each configuration. However, the proposed loading table does not include maximum fuel enrichment limit and minimal burnup requirement for each configuration. Because the shield transfer canister (STC) takes credit for fuel burnup, the loading table must include a maximum enrichment and minimal burnup for each configuration and/or location in the fuel basket that takes burnup credit. The licensee is requested to revise Table 3.1.2-3 to include the maximum enrichment limit and the required minimal burnup for each configuration.

Regulatory Basis: This information is needed to determine if the Indian Point spent fuel inter-unit transfer canister STC design meets the regulatory requirements of 10 CFR 72.124(a).

RAI-14 (CSRB-Criticality Safety)

Clarify if damaged fuel is intended to be transferred by the transfer cask. If so, provide a criticality safety analysis consistent with the fuel conditions and intended configurations. One of the proposed changes to the TSs is to change LCO 3.1.2.c from "Only INTACT FUEL ASSEMBLIES with initial enrichment ≥ 3.2 and ≤ 4.4 wt% U-235 and discharged prior to IP3 Cycle 12 shall be placed in the STC basket" to the various configurations as defined in Table 3.1.2-3 of Attachment 2 to the revised SAR. However, there is no restriction in the configurations presented in Table 3.1.2-3 on whether the fuel shall be intact or not. As such, with the proposed changes, the restriction that all fuel to be transferred by the fuel transfer cask shall be intact would be eliminated from the TSs. However, in Table 10.0.1 on page 10-2 of the

SAR, the licensee states, "Failed fuel assemblies (i.e. assemblies that are not intact) and/or damaged fuel are not permitted for transfer in the STC." In addition, the criticality safety analyses for the STC does not include any analyses for the transfer cask containing damaged fuel. The licensee is requested to clarify whether damaged fuel is the intended content to be transferred from the Indian Point 3 spent fuel pool to the Indian Point 2 spent fuel pool using the STC. If so, provide criticality safety analyses consistent with the fuel conditions and intended configurations.

Regulatory Basis: This information is needed to determine if the Indian Point spent fuel inter-unit transfer canister STC design meets the regulatory requirements of 10 CFR 72.124(a).

RAI-15 (CSRB-Criticality Safety)

Clarify the definition for the term "simulated actinide compositions" and justify the validity of this approach for benchmarking analyses.

On page 4.A-1 of the SAR, the licensee states:

> Since then, an expanded set of critical experiment has been analyzed [L.O], that includes critical experiments with simulated actinide compositions of spent fuel. This results in a bias of -0.0013 with an uncertainty 0.0086 (95/95), i.e. values that are very similar to those used in the HI-STAR 100. Those values were used in all calculation in the Tables in the main part of Chapter 4 of this report where a maximum k_{eff} is reported.

However, the term "simulated actinide compositions of spent fuel" was not defined in the SAR. The licensee is requested to clarify the definition of this term and provide justification for the use of "simulated spent fuel compositions" in code benchmarking analyses for burnup credit application. If the term means data that is not obtained from radiochemical assay (RCA) samples, the licensee is requested to provide information on the extent of the use of data. If a significant amount of surrogate data is used in the code benchmarking analyses, the licensee is requested to provide justification for the validity of the approach.

Regulatory Basis: This information is needed to determine if the Indian Point spent fuel inter-unit transfer canister STC design meets the regulatory requirements of 10 CFR 72.124(a).

RAI-16 (CSRB-Criticality Safety)

Clarify if fresh fuel critical experiments are used for the criticality code benchmarking analyses for STC configurations that take burnup credit. If so, provide justification for the validity of this approach or revise the benchmarking analyses with appropriate critical experiments per the recommendation provided in NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses–Criticality (k_{eff}) Predictions" (ADAMS Accession No. ML12116A128).

On page 4.A-1 of the SAR, the licensee states, "HI-STAR 100 initially use critical experiments with fresh and mixed uranium and plutonium oxides (MOX) fuel that results in a bias of -0.0004 with an uncertainty of 0.0083 (95/95). This is discussed in the appropriate section of this appendix, but was only used for the initial scoping calculation for this project." The licensee further states, "Since then, an expanded set of critical experiment has been analyzed [L.O], that includes critical experiments with simulated actinide compositions of spent fuel." As such, it is not clear if fresh fuel critical experiments were used in the final code benchmarking analyses. Based on the latest research results published in NUREG/CR-7109, fresh fuel critical experiments are not suitable for code benchmarking for burnup credit applications. Specifically, NUREG/CR 7109 states:

> Based on the analyses performed for this report, it is affirmed that criticality analysts should continue to validate BUC [burnup credit] criticality safety evaluations to the extent possible, with the best available critical experiment data. MOX configurations from the IHECSBE [International Handbook of Evaluated Criticality Safety Benchmark Experiments] and the HTC [French Huat Taux De Combustion] experiment configurations, collectively, provide sufficient data for validation of BUC analyses with major actinides and hence should be used for validation. LEU [low-enrichment uranium] critical configurations should not be used in a conventional validation analysis to validate burned fuel systems because they do not include any bias contribution from the plutonium present in burned fuel. The validation statistical analysis should include bias trending analysis as a function of plutonium content, using a trending variable such as plutonium fraction (i.e., gram of Pu per gram of Pu + U).

The licensee is requested to clarify if fresh fuel critical experiments are used for criticality code benchmarking analyses for STC that takes burnup credit. If so, provide justification for the validity of this approach or revise the benchmarking analyses with appropriate critical experiments.

Regulatory Basis: This information is needed to determine if the Indian Point spent fuel inter-unit transfer canister STC design meets the regulatory requirements of 10 CFR 72.124(a).

RAI-17 (CSRB-Criticality Safety)

Provide criticality safety analyses for these configurations corresponding to the revised loading table for the STC fuel loading configurations that take burnup credit.

The licensee states that it performed criticality safety for the STC using the method from HI-STAR 100 criticality safety analysis. However, the NRC staff was unable to find the analyses for the STC, except some general discussion of the applicability of the methodology. Although the approach used for the HI-STAR 100 criticality was accepted by the NRC staff, the licensee is expected to perform specific analyses for the STC with various combinations of enrichment and burnup to support the revised loading table per RAI-13, above, as appropriate. An MCNP (computer code) input or output file containing the material composition data for the most reactive STC model using burnup credit would suffice.

Regulatory Basis: This information is needed to determine if the Indian Point spent fuel inter-unit transfer canister STC design meets the regulatory requirements of 10 CFR 72.124(a).

RAI-18 (CSRB-Criticality Safety)

Provide the fuel material composition for the most reactive STC configuration that takes burnup credit.

The licensee states that it performed criticality safety for the STC using burnup credit. However, the SAR does not include the spent fuel material composition. The NRC staff requests the material composition of spent fuel that is used in the MCNP model to verify the criticality safety analysis for the most reactive STC configuration that takes burnup credit. An input file for the model of the STC that exhibits the maximum reactivity would suffice.

Regulatory Basis: This information is needed to determine if the Indian Point spent fuel inter-unit transfer canister STC design meets the regulatory requirements of 10 CFR 72.124(a).

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NOS. 2 AND 3 – REQUEST FOR ADDITIONAL INFORMATION REGARDING INTER-UNIT TRANSFER OF SPENT FUEL LICENSE AMENDMENT REQUEST (CAC NOS. MF8991 AND MF8992) DATED AUGUST 10, 2017

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