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

June 21, 2016

Mr. Joseph W. Shea
Corporate Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST REGARDING EXTENDED POWER UPRATE (CAC NOS. MF6741, MF6742, AND MF6743)

Dear Mr. Shea:

By letter dated September 21, 2015, as supplemented by letters dated November 13, December 15 (2 letters), and December 18, 2015, Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for the Browns Ferry Nuclear Plant, Units 1, 2, and 3. The proposed amendment would increase the authorized maximum steady-state reactor core power level for each unit from 3,458 megawatts thermal (MWt) to 3,952 MWt. This LAR represents an increase of approximately 20 percent above the original licensed thermal power level of 3,293 MWt, and an increase of approximately 14.3 percent above the current licensed thermal power level of 3,458 MWt.

In addition, by letters dated January 15 and January 28, 2016, the U.S. Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI). The licensee, by letters dated February 18 and March 8, 2016, responded to the requested information.

The NRC staff reviewed the licensee's submittals and determined that additional information is needed. On March 22, 2016, the NRC staff forwarded, by electronic mail, two sets of draft RAIs to TVA. On April 7, 2016, the NRC staff held a conference call to provide the licensee with an opportunity to clarify any portion of the draft RAI and discuss the timeframe for which TVA may provide the requested information.

In addition, on May 10 and 11, 2016, the NRC staff conducted an audit of the licensee's draft responses and the supporting documents to support the review of the extended power uprate LAR regarding the adequacy of containment accident pressure. Subsequent to the audit, the NRC staff added another RAI to the draft RAIs and provided clarification on some of the draft RAIs. As agreed to by the NRC and TVA staff, TVA will respond to the enclosed RAI by July 29, 2016. TVA staff confirmed that the enclosed RAIs do not contain any sensitive information.

If you have any questions, please contact me at 301-415-1447 or farideh.saba@nrc.gov.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

Enclosure:
Request for Additional Information

cc w/enclosure: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST RELATED TO

EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

By letter dated September 21, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15282A152), as supplemented by letters dated November 13, December 15 (2 letters), and December 18, 2015 (ADAMS Accession Nos. ML15317A361, ML15351A097, ML15351A113, and ML15355A413, respectively), Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3. The proposed amendment would increase the authorized maximum steady-state reactor core power level for each unit from 3,458 megawatt thermal (MWt) to 3,952 MWt. This LAR represents an increase of approximately 20 percent above the original licensed thermal power level of 3,293 MWt, and an increase of approximately 14.3 percent above the current licensed thermal power level of 3,458 MWt.

In addition, by letters dated January 28 (2 letters), 2016 (ADAMS Accession Nos. ML16020A111 and ML16019A283), the U.S. Nuclear Regulatory Commission (NRC) issued two sets of requests for additional information (RAIs). The licensee, by letters dated February 16 and March 8, 2016 (ADAMS Accession Nos. ML16049A248 and ML16069A142, respectively), responded to the requested information.

The NRC staff from Steam Generator Tube Integrity and Chemical Engineering Branch (ESGB), Division of Engineering and staff from the Nuclear Performance and Code Review Branch, Division of Safety Systems, Office of Nuclear Reactor Regulation, have reviewed the information the licensee provided and determined that the following additional information associated with Boral plates and spent fuel pool (SFP) criticality is required in order to complete the review.

RAIs Regarding Boral Plates:

ESGB-RAI 2[1]

In NUREG-1801, Revision 2 (published December 2010) Generic Aging Lessons Learned Report, Aging Management Program XI.M40, the staff has established a 10-year maximum surveillance interval for the inspection and testing of a neutron absorbing material other than Boraflex in the SFP. This report also specifies that the periodic testing should include measurement of areal density via coupons or in situ techniques. The program described in your

[1] ESGB-RAI 2 and 3 followup ESGB-RAI 1 in an NRC letter dated January 28, 2016.

RAI response, dated February 16, 2016, is consistent with the staff's expected time interval for testing of your coupons, however you state that neutron attenuation testing to verify Boron-10 (B-10) areal density is not included in the program. You further state that neutron attenuation testing has not been performed on the coupons at BFN since 1985. Please describe your plans to perform neutron attenuation testing in the future. Include a discussion of any modifications to your current program to require attenuation testing.

ESGB-RAI 3

The Boral coupon surveillance program at BFN relies on coupons in the Unit 3 SFP to represent the Boral in all three units. Describe why the Unit 3 coupons are representative or bounding of the service conditions for the Units 1 and 2 Boral. Please discuss the cumulative time that the actual Boral panels have been in service in each unit and compare that time to the Unit 3 coupons. The response should address similarities and differences in the temperature, flow, water chemistry, and irradiation levels between the SFPs as well as similarities and differences in the materials and manufacturing processes of the Boral panels between the SFPs.

RAIs Regarding SFP Criticality:

Regulatory Requirements

The applicable Section 50.68 of Title 10 of *Code of Federal* Regulations (10 CFR) requirement is that the k-effective of the SFP storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95-percent probability, 95-percent confidence level, if flooded with unborated water. AREVA report ANP-3160, Revision 1, documents a criticality analysis performed to demonstrate that this regulatory limit is met. The staff has identified some instances where it is not clear if the reactivity impact due to specific conditions was adequately addressed in the criticality analysis. The potential reactivity impacts may be positive, so the staff needs additional information to verify the regulatory limit will not be challenged by these potential impacts.

SFP-RAI 3

The SFP criticality analysis documented in ANP-3160 used a 2x2 infinite array to determine the k-effective for normal conditions, then analyzed various accident conditions to determine the maximum increase in reactivity. This reactivity increase was then applied as an adder to the k-effective for normal conditions. Since the interface between two SFP rack modules includes face adjacent cells that are separated by stainless steel closure plates with no Boral, the 2x2 infinite array does not accurately capture the local configuration in these locations.

As part of the normal condition analysis, the reactivity for nominal rack module interface spacing was demonstrated to be less than the 2x2 infinite array. As part of the accident condition analysis, a configuration where the rack module interface spacing was significantly reduced was determined to result in an increase in reactivity. The staff notes that if there are no controls in place to ensure that the rack module spacing is equal to or greater than the nominal spacing, then a reduction in rack module spacing may become part of the normal condition. Since a sufficiently large reduction in the rack module spacing would result in a more reactive configuration, provide the following:

a. A discussion of the controls in place at the plant to ensure that the nominal rack module spacing is maintained for normal operating conditions, or

b. If no such controls exist, a discussion of the reactivity impact of potential reductions in rack module spacing on the normal and accident conditions. For example, an accident configuration postulating a single missing Boral plate assumes that the missing plate is located on the interior of a SFP rack module. Other possible configurations involving missing Boral plates do not appear to have been analyzed. For example, a missing Boral plate might be located at the periphery of a SFP rack module, in such a location that results in three face adjacent fuel assemblies without Boral plates between them.

SFP-RAI 4

A number of calculations, including the final criticality analysis used to demonstrate compliance with the 10 CFR 50.68 requirement, are performed using the KENO V.a module in the SCALE 4.4a code system. ANP-3160 discusses how the hardware and software configuration is controlled to ensure that the code system used to run the calculations is appropriately qualified. However, no information is provided to describe how the licensee verified that each calculation converged to an appropriate solution. Appropriate code convergence must be verified to have reasonable assurance in the calculated k-effective results. Describe and justify the method used to verify appropriate convergence. Include the following factors: initial source distribution, calculational parameters controlling the use of histories in computing the final calculated k-effective, convergence checks performed, and their acceptance criteria.

SFP-RAI 5

ANP-3160 states that the Boral is modeled using the design minimum B-10 areal density. The staff was unable to locate information describing how the Boral installed in the BFN SFP racks was verified to meet this minimum value, including any applicable measurement uncertainties. Provide an explanation of how the minimum B-10 areal density used in the SFP criticality analysis bounds all of the Boral plates installed in the BFN SFP racks.

SFP-RAI 6

The spacer and channel growth due to burnup was accounted for in ANP-3160 by statistically combining the reactivity worth of a conservative estimate of the fuel rod pitch change and channel growth with other uncertainties (root of sum of squares method). This treatment is appropriate for a randomly distributed factor that is normally distributed around nominal dimensions, but spacer and channel growth would be expected to be normally distributed around an off-nominal value. Provide a justification for the treatment of the spacer and channel growth solely as an uncertainty.

SFP-RAI 7

ANP-3160 states that pellet deformation with respect to burnup can be ignored, because it does not change the material content of the fuel. Fuel rod geometry changes as a result of irradiation may result in a change in the cladding outer diameter, which would change the fuel-moderator

ratio. The results from the manufacturing tolerance analyses demonstrate that a change in the cladding outer diameter can have a positive reactivity impact. Provide further discussion on how fuel rod geometry changes might be expected to affect the criticality analysis.

SFP-RAI 8

In ANP-3160, Table 2.1, the second option to satisfy the fuel design limitations for enriched lattices involves use of a CASMO-4 storage rack model provided in Appendix A to evaluate each enriched lattice of a fuel assembly. The criticality analyses were performed using commercial grade uranium (CGU) fuel to conservatively bound blended low enriched uranium (BLEU) fuel, and the CASMO-4 model in Appendix A reflects this assumption. However, the previously manufactured BLEU fuel was modeled explicitly as BLEU fuel in the Appendix B evaluation of legacy fuel. Consequently, it is not clear if modeling a fuel lattice as CGU with the same uranium-235 (U-235) enrichment is required as part of satisfying the Table 2.1 limitations, or if the Appendix A inputs may be modified to explicitly model the BLEU fuel. Provide clarification on the intended application of this criterion. If explicit modeling of BLEU is allowed in meeting this criterion, then discuss how BLEU specific uncertainties are addressed, such as manufacturing tolerances, depletion uncertainties, and code validation uncertainties.

SFP-RAI 9

ANP-3160 states that the gadolinia manufacturing uncertainty is evaluated with a combination of KENO V.a and CASMO-4. Since the final criticality analysis utilizes a REBOL lattice with no gadolinia, it is not clear how the gadolinia manufacturing uncertainty was determined using KENO V.a. CASMO-4 appears to be used primarily to compute a depletion based adder. Provide a description of the model used in KENO V.a to evaluate the gadolinia manufacturing uncertainty.

SFP-RAI 10

ANP-3160 discusses a series of calculations in which the radial location of the fuel lattice within the SFP cell was varied to account for fuel assembly lean. Provide additional detail on what configurations were analyzed.

SFP-RAI 11[2]

The response to SFP-RAI 1 (ADAMS Accession No. ML16049A248) discusses requirements given in the criticality safety analysis (CSA) report (AREVA Report ANP-3160, Revision 1, attachment to ADAMS Accession No. ML15351A097) that must be met for fuel stored in the BFN SFPs. Fuel stored in the BFN SFPs may meet either one of two sets of requirements. The first set of requirements impose limits on the U-235 and gadolinia loading for the top and bottom lattices. The second set of requirements represent an in-rack k-infinity limit for the top and bottom lattices. The first set implicitly confirms that the second set is met, since the criticality analysis demonstrates that the maximum U-235 loadings combined with the minimum gadolinia loadings will result in an in-rack k-infinity that meets the provided limits. The NRC staff has determined that some of the margin to the regulatory limit as calculated in the CSA report will

[2] SFP-RAI 11 is follow up to SFP-RAI 1 in the NRC letter dated January 28, 2016.

need to be credited to address potential nonconservatisms. Therefore, for the NRC staff to make a safety finding, the upper limit of the reactivity of the fuel lattices used in the CSA must be maintained. The Standard Technical Specifications list a limit on the k-infinity as calculated in the standard cold core geometry for fuel stored in SFPs, which is an example of one way of meeting this condition.

Propose a means of ensuring that the maximum reactivity of fuel assemblies loaded in the SFP will be limited such that the NRC safety finding for this CSA will continue to be valid for future fuel assemblies stored in the BFN SFPs.

SFP-RAI 12[3]

In response to SFP-RAI 2, the licensee indicated a review of plant records showed that the Boral verification testing was inconclusive for some cell locations on six BFN SFP storage modules. In order to evaluate these cell locations, a statistical analysis was performed to determine the probability that no more than one Boral plate is missing. The staff notes that if a single Boral plate is missing, then this would become part of the normal condition and should be evaluated as such. Since such a normal condition could result in a more limiting accident condition, provide the following:

a. The locations of the cells for which the Boral verification testing was inconclusive.

b. The statistical analysis performed to evaluate the cell locations for which the Boral verification testing was inconclusive.

c. A technical justification for the assumption that the normal BFN SFP rack condition does not include any missing Boral plates.

d. As part of an audit performed May 10 - 11, 2016, TVA provided draft documentation that included information to address one missing Boral panel as part of the normal condition. In addition to submitting this information on the docket, the NRC staff requests the following information:

    i. A discussion of the applicability of the Edwin I. Hatch Nuclear Power Plant (Hatch) testing results to the BFN SFPs, more specifically addressing how the licensee has determined that the manufacturing process used to fabricate the SFP racks delivered to Hatch is the same as those delivered to BFN, and

    ii. Information about any similar testing results from other sites that the licensee may be aware of.

[3] SFP-RAI 12 is follow up to SFP-RAI 2 in the NRC letter dated January 28, 2016.

If you have any questions, please contact me at 301-415-1447 or farideh.saba@nrc.gov.

Sincerely,

/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

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Request for Additional Information

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