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

March 15, 2016

Mr. Joseph W. Shea
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 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15282A152), as supplemented by letters dated November 13, December 15, and December 18, 2015 (ADAMS Accession Nos. ML15317A361, ML15351A113, and ML15355A413, respectively), 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.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's submittals and determined that additional information is needed. On February 4, 2016, the NRC staff forwarded, by electronic mail, a draft request for additional information (RAI) to TVA. On February 18, 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. As agreed by NRC and TVA staff during the conference call, TVA will respond to the enclosed RAI by April 4, 2016.

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

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REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST REGARDING EXTENDED POWER UPRATE

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, and December 18, 2015 (ADAMS Accession Nos. ML15317A361, ML15351A113, and ML15355A413, respectively), Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for the 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 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.

The U.S. Nuclear Regulatory Commission (NRC) staff from the Reactor Systems Branch (SRXB), Division of Safety Systems, Office of Nuclear Reactor Regulation, reviewed the information the licensee provided and determined that the following additional information is required in order to complete the evaluation.

SRXB-RAI 1 MICROBURN-B2 (MB2) Non-Convergence Issue

The licensee, in Tables 3 and 4 of Enclosure 1, "Browns Ferry EPU [Extended Power Uprate] Low Flow Issues Impact (Proprietary),[1] AREVA FS1-0024528, Revision 1.0," of the letter dated November 13, 2015, provided the revised operating limit minimum critical power ratio (OLMCPR) two recirculation pump trip (2RPT) limits that are calculated with the revised version of MB2. The changes between the original and revised OLMCPR values are significant (almost a factor of 2). However, (1) the delta over initial versus oscillation magnitude curve is essentially unchanged (see Figure 1), and (2) the 2RPT event is initiated from full power, where MB2 did not have any convergence issue. Provide an explanation for the significant changes in OLMCPR.

SRXB-RAI 2 MB2 Void-Quality Correlation

Section 2.2 of FS1-0024528 provides information that appears to be contradictory. The solution presented is to remove the feature implemented in the correlation that was found to cause problems and then review the analyses to ensure that the results are acceptable. However, if the results of analyses are acceptable, removing the low flux cutoff should have had no effect. Provide a more detailed explanation of the analysis performed, including the cause and effect of

[1] Enclosure 2 (FS1-0024530) contains a non-proprietary version of this document.

the changes implemented in the correlation for this analysis. Indicate what conditions and transients were analyzed and the impact on the BFN EPU LAR.

SRXB-RAI 3 Bypass Voids

Figure 7-7 in the enclosure for Attachment 34,[2] "ANP-2860P, Revision 2," to LAR (ADAMS Accession No. ML15282A152) provides the bypass voids at local power range monitor level elevations, but it does not specify the power-flow condition.

a. Provide the bypass void at D-level at the intercept of the backup stability protection region with the high-flow line.

b. If greater than 5 percent, is a correction required for decalibration (i.e., "losing the calibration" that was set without voids in the bypass)?

SRXB-RAI 4 Emergency Operating Procedures (EOPs)

What version of the emergency operating guidelines (emergeny procedure gudelines (EPG)) is implemented in the current BFN EOPs? Provide a comparison between the values of the cycle-specific EOP parameters (e.g., hot shutdown boron weight) pre- and post-EPU.

SRXB-RAI 5 Safety Limit Minimum Critical Power Ratio (SLMCPR) Adders

The NRC requires that the licensee in the EPU applications provide SLMCPR and OLMCPR adders for operation at EPU conditions. Provide a list of the OLMCPR and SLMCPR adders proposed for BFN using AREVA methods and/or a justification for not applying them.

SRXB-RAI 6 Core Flow

The power-flow map in Figure 1.1 of Enclosure 3, "ANP-3404P, Revision 3, Browns Ferry Unit 3 Cycle 19 Representative Reload," of letter dated December 15, 2015, shows Point F with 105 percent flow in the increased core flow region. What is the maximum core flow that can be achieved at BFN? Is this a function of exposure (e.g., bottom-peaked shapes may result in reduced max achievable flow)? Is BFN susceptible to bi-stable flow in the recirculation loops? If so, what is the maximum (or range of) achievable recirculation flow used in normal operation to minimize bi-stable flow concerns?

SRXB-RAI 7 SRV

In Attachment 6[3], "NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate [aka, PUSAR]," to the LAR, it is assumed that one main safety relief valve (SRV) is out of service (OOS); however, the LAR is confusing, as different assumptions are used for different analyses. Has the number of allowed SRVs OOS

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[2] Attachment 35 contains a non-proprietary version of Attachment 34.

[3] Attachment 7, "NEDO-33860, Safety Analysis Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3 Extended Power Uprate," contains a non-proprietary version of Attachment 6.

been changed as a result of the EPU operating domain extension? Is the 3 percent tolerance value supported by testing results?

SRXB-RAI 8 ACE Critical Power Ratio (CPR) Correlation Applicability Range

Section 4 of ANP-2860P, Revision 2, states that approved corrective actions are applied if analyzed conditions in BFN fall outside the ACE CPR correlation range of applicability. The range of applicability is defined in Table 2-1 of ANP-10298P,[4] Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," in terms of flow, pressure, inlet sub-cooling, and local peaking. However, the approved corrective actions are defined in Table 3-1 of Supplement 1P to ANP-10298PA in terms exclusively of local peaking factor. Please confirm that the analyzed conditions (in terms of flow, pressure, and inlet sub-cooling) in BFN fall inside the range of applicability of the ACE correlation, and only the local peaking (mostly of controlled bundles) is outside.

SRXB-RAI 9

ANP-3404, Revision 3, "BFN Unit 3 Cycle 19 Representative Reload Analysis at Extended Power Uprate (Proprietary)[5]" (Enclosure 3 to letter dated December 15, 2015), is submitted to support the EPU operation of all three units.

a. Identify any design and operational differences between the units that affect the EPU analyses, if any.

b. Justify the application of the Unit 3 analysis at EPU for the other two units and explain in detail that the Unit 3 analysis is representative or bounding for the other two units.

SRXB-RAI 10

ANP-3404P, Table 2.1, "Disposition of Events Summary for EPU and AREVA Fuel at Browns Ferry," in the comments column of page 2-8 for the Final Safety Analysis Report, Sections 14.5.4.3 and 14.5.4.4, "Control Rod Removal Error During Refueling," and "Fuel Assembly Insertion Error During Refueling," stated respectively that, "This event is not credible," and, "An unplanned criticality during refueling due to a single fuel assembly insertion error is not credible." The staff does not agree with the statement that these events are not credible. These events, which may be infrequent events, happened in plants, and hence, these events may not be categorized as "incredible events." Provide an evaluation for these infrequent events in BFN.

SRXB-RAI 11

Chapter 14 of the BFN Updated Final Safety Analysis Report addresses loss of a feedwater heater event. Clarify whether any BFN units had any loss of feedwater event during the life of the plant, and if so, what was the maximum feedwater temperature reduction during the event?

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[4] A non-proprietary version of this document is provided at ADAMS Accession No. ML14183A734.

[5] Enclosure 4 (ANP-3404NP) contains a non-proprietary version of this document.

SRXB-RAI 12

ANP-3404P, Table 1.1, "EOD and EOOS Operating Conditions," lists "Extended Operating Domain (EOD) Conditions" and "Equipment Out-Of-Service (EOOS) Operating Conditions." Section 5.3 describes EOOS scenarios. Section 5.3 describes operation scenarios with turbine bypass valve OOS, feed water heater OOS, power load unbalance OOS, and different combinations of these scenarios.

a. Clarify whether TVA is planning to take credit for these analyses when the above specified equipment becomes inoperable. ANP-3404P representative analyses are only for Unit 3, Cycle 19, and hence, may not accurately reflect the actual Unit 1 and Unit 2 plant configuration in future operations.

b. Specify the number of turbine bypass valves, feed water heater OOS assumed to be inoperable.

c. How many power load unbalance devices are in the plant? Are they located in the control room or outside the control room?

d. Minimum critical power ratio limits are given in Tables 8.1 and 8.2 for EOOS. Describe the nominal scram speed and technical specifications scram speed tests performed during the EOOS operation, if they are different from the tests done at normal operation.

e. Note below that Table 1.1 states, "SLO [single-loop operation] may be combined with all of the other EOOS conditions." The NRC staff does not agree with this statement. SLO operation is susceptible to thermal-hydraulic stability, and feedwater heater OOS may increase the susceptibility. Verify that analyses have been performed to confirm the TVA statement.

f. Note below that Table 1.1 mentions two traversing in-core probe (TIP) machines OOS. How many TIP machines are in BFN units?

SRXB-RAI 13

PCT for anticipated transient without scram (ATWS) is not provided in the LAR. Provide a quantitative assessment to justify the TVA position that the PCT calculation is not needed for the ATWS scenario.

SRXB-RAI 14

ANP 3403P, Revision 3, "Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3 (Proprietary)," Enclosure 1[6] to the letter dated December 15, 2015, in Section 2.8.5.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," Technical Evaluation, TVA stated, "The event was analyzed at EPU conditions and resulted in an unblocked delta CPR of 0.27." What is meant by "unblocked"?

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[6] Enclosure 2 contains a non-proprietary version of Enclosure 1.

SRXB-RAI 15

ANP 3403P, Section 2.8.5.4.4, "Spectrum of Rod Drop Accidents," Technical Evaluation (page 95) in the Criteria Table, number of failed rods is given as less than 850. Identify the source from where the criterion for number of failed rods was obtained (less than 850 in this case).

SRXB-RAI 16

TVA stated in page 41, Attachment 8[7], "ANP-3403P, Fuel Uprate Safety Analysis Report [FUSAR] for Browns Ferry Units 1, 2, and 3 (proprietary)," LAR, that the average bundle power increases from 4.53 MW/bundle to 5.17 MW/bundle (approximately 14 percent) from pre-EPU to post-EPU, which corresponds to the same percent increase of total core power from current licensed thermal power (CLTP) to EPU. It is assumed in the constant pressure EPU for BWRs that the additional core power is obtained by raising the average bundle power, and that the peak bundle power should remain approximately the same. However, other plant operations at EPU power level have shown that peak bundle power can increase by a limited amount. Provide the current peak bundle power and compare it with the expected value of peak bundle power for EPU operation in BFN.

SRXB-RAI 17

Stress corrosion cracking (SCC) and pellet-clad-interaction (PCI) phenomena can cause clad perforation, resulting in leaking fuel bundles and resultant increased reactor coolant activity. Provide the following additional information regarding PCI/SCC for BFN at EPU conditions:

a. Describe whether ATRIUM-10 and XM fuel designs have barrier cladding that has built-in PCI resistance.

b. Describe any differences in operating procedures associated with PCI/SCC at EPU conditions versus pre-EPU operations.

c. From the standpoint of PCI/SCC, discuss which of the anticipated operational occurrences (AOOs), if not mitigated, would most affect operational limitations associated with PCI/SCC.

d. For the AOOs in part c. above, discuss the differences between the type of required operator action, if any, and the time to take mitigating actions between pre-EPU and EPU operations.

e. If the EPU core will include fuel designs with non-barrier cladding that have less built-in PCI resistance, demonstrate by plant-specific analyses that the peak clad stresses at EPU conditions will be comparable to those calculated for the current operating conditions.

f. Describe operator training on PCI/SCC operating guidelines.

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[7] Attachment 9 contains a non-proprietary version of Attachment 8.

SRXB-RAI 18

TVA stated in pages 5-6 and 2-14 of ANP-3404P that CASMO-4/MICROBURN-B2 code was used to calculate flow dependent linear heat generation rate operating limits to establish the fuel failure criteria for the recirculation flow control failure – increasing flow event. The analysis of the event assumes recirculation flow increases slowly along the limiting rod line to the maximum flow physically attainable by the equipment. Discuss whether there was any impact of MICROBURN-B2 low flow issues on this slow flow event.

SRXB-RAI 19

TVA stated in page 2-410 of the PUSAR that the standby liquid control system (SLCS) relief valve setpoint margin is 33 pounds per square inch (psi) for EPU. Please provide the following additional information:

a. What is the allowable value (AV) for SLCS relief valve setpoint drift for BFN?

b. Whether the 33 psi AV account for relief valve setpoint drift, and justify that the available margin remains conservative at EPU conditions.

c. The relief valve setpoint margin for CLTP after accounting for the AV setpoint drift.

d. The minimum margin required for the SLCS pump discharge relief valves to remain closed during system injection.

SRXB-RAI 20

In page 107 of the FUSAR, TVA stated that BFN reactors are small-break limited, as determined by AREVA's EXEM BWR-2000 evaluation model, and that this trend in the break spectrum analysis was not impacted by EPU. Provide the following additional information regarding the loss-of-coolant accident (LOCA) analysis performed in support of EPU operation at BFN:

a. In page 109 of the FUSAR, limiting LOCA break characteristics (location, type/size, single failure, axial power shape, and initial state) were provided for EPU condition. Compare these characteristics of EPU with that of pre-EPU condition. Discuss if there was any significant change in these characteristics as a result of power uprate from CLTP to EPU, and explain why.

b. Provide the total number of automatic depressurization system (ADS) valves at BFN, and the number of ADS valves assumed to be OOS for limiting small-break LOCAs for CLTP and EPU. Explain the difference, if any.

c. The limiting PCT for EPU is provided in FUSAR, page 109, as 2,086 degrees Fahrenheit (°F) for ATRIUM-10 and 2008 °F for XM fuels. Provide the limiting PCT for CLTP condition, including the type of fuel, and explain any significant change between the PCTs before and after EPU.

d. Provide the break location, type/size, single failure, axial power shape, initial state, and the PCT at EPU condition for the worst large-break LOCA case.

SRXB-RAI 21

Provide a brief description of how the stability mitigation actions (e.g., immediate water level reduction and early boron injection) are implemented in BFN. Discuss if operation at constant pressure power uprate conditions requires modification of any operator instruction, including the EOP and the EPGs and severe accident guidelines (EPG/SAG) that are applicable to BFN.

SRXB-RAI 22

Table 2.6-3 of PUSAR provides BFN peak suppression pool (SP) temperature for postulated ATWS, station blackout, and Appendix R fire events. Compare peak SP temperatures for these events with that of pre-EPU condition, and explain any significant change.

SRXB-RAI 23

Provide a brief description of the plant training simulator neurotic core model. Provide the schedule as to when the BFN plant training simulator is upgraded for EPU conditions.

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
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