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

June 13, 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, 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 letter dated February 18, 2016, the U.S. Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI). The licensee, by letter dated March 28, 2016, responded to the requested information.

The NRC staff reviewed the licensee's submittals and determined that additional information is needed. On March 28, 2016, the NRC staff forwarded, by electronic mail, a draft RAI to TVA. On April 8, 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, from May 3, to May 5, 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 deleted one of the draft RAIs and revised two for clarification. As agreed to by the NRC and TVA staffs, TVA will respond to the RAI in Enclosure 1 according to the proposed dates presented in Enclosure 2. The TVA 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
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Office of Nuclear Reactor Regulation

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

Enclosures:
1. Request for Additional Information
2. Proposed TVA Response Schedule

cc w/enclosures: Distribution via Listserv

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 (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 (CLTP) of 3,458 MWt.

By letter dated February 18, 2016, the U.S. Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI) (ADAMS Accession No. ML16041A307). The licensee, by letter dated March 28, 2016 (ADAMS Accession No. ML16089A054), responded to the requested information.

The NRC staff from the Reactor System Branch, previously from Containment and Ventilation Branch (SCVB), 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.

SCVB-RAI 2[1]

In Reference 2 (Power Uprate Safety Analysis Report (PUSAR)), Table 2.6-5, Note 3 states:

> The larger drywell volume of 171,000 ft^3 [cubic feet] (compared to the minimum DW [drywell] volume of 159,000 ft^3 (See Table 2.6-2a) results in a larger initial drywell non-condensable gas mass and more non-condensable gas transferred to the wetwell during a LOCA [loss-of-coolant accident]. This maximizes the wetwell and drywell pressure and is conservative.

The above statement provides a qualitative justification for a higher drywell pressure response for a larger drywell volume. For the short term drywell pressure response, it seems that using the larger drywell volume of 171,000 ft^3 would result in a less limiting pressure response than using the minimum volume of 159,000 ft^3. Confirm that maximum drywell and wetwell pressure response mentioned above (using the larger drywell volume of 171,000 ft^3) was obtained by a

[1] NRC letter dated February 18, 2016, contained SCVB-RAI 1.

sensitivity analysis performed using the LAMB and M3CPT computer codes for drywell volumes of 171,000 ft^3 and 159,000 ft^3 using same assumptions and same remaining input parameters.

SCVB-RAI 3

Reference 1, Enclosure, Section 3.2 states:

> The elimination of CAP [Containment Accident Pressure] credit from the licensing basis is accomplished through system modifications and analytical assumption changes that are factored into the safety analyses.
>
> TVA is proposing a modification to increase the isotopic B-10 [boron-10] enrichment provided by the SLC [Standby Liquid Control] System. Raising the boron-10 enrichment for EPU [extended power uprate] increases the rate of negative reactivity inserted by the SLC system and results in a faster shut down of the reactor during the ATWS [Anticipated Transient Without Scram] event. This results in a reduced heat load input into the suppression pool; therefore, the suppression pool temperature is lower.

Describe all system modifications, in particular system hardware [emphasis added] modifications, performed for elimination of CAP credit for the available Net Positive Suction Head (NPSH) analysis for the Residual Heat Removal (RHR) and Core Spray (CS) pumps other than the change in the RHR heat exchanger K-values described in Reference 5, and the increase in isotopic B-10 enrichment provided by the SLC system.

SCVB-RAI 4

Table 2.6-2a of PUSAR does not provide the assumptions for the initiation of feedwater (FW) flow isolation time and the valve closure time for the suppression pool temperature response analysis.

Updated Final Safety Analysis Report (UFSAR) Section 14.11.3.3.1, item f provides the FW flow assumption for the current Recirculation Suction Line Break (RSLB) Design Basis (DB) LOCA licensing basis analysis as follows:

> The feedwater flow was assumed to stop instantaneously at time zero. This conservatism is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby reducing the discharge of steam and water into the primary containment.

Section 14.11.5.1.1 of UFSAR, item j provides the FW flow assumption for the current main steamline break LOCA licensing basis analysis as follows:

> Feedwater flow is assumed to decrease linearly to zero over the first four seconds to account for the slowing down of the turbine-driven feed pumps in response to the rise in reactor vessel water level.

State the assumption for the EPU analysis that was used for feedwater isolation for the short and long term suppression pool temperature response analyses and how it differs from the current licensing basis analysis. Provide justification in case the conservatism in the EPU analysis is reduced.

SCVB-RAI 5

Refer to Section 2.6.3.1.1 of PUSAR third paragraph; explain the basis for assuming the LOCA signal (high drywell pressure concurrent with low reactor pressure vessel (RPV) pressure) occurs in the accident unit 10 minutes after the accident initiation in the small steamline break (SSLB) accident analysis.

SCVB-RAI 6

Section 2.6.5.1 of PUSAR under the heading "Suppression Pool Temperature Response – recirculation discharge line break (RDLB) LOCA" states: "(break area 4.2 ft^2 [square feet] for the RSLB versus 1.94 ft^2 for the RDLB)." The RSLB area does not match with the break flow area given in Table 2.6-5; explain.

SCVB-RAI 7

Section 2.6.5.1 of PUSAR, third paragraph under heading "Suppression Pool Temperature Response – Small Steam [Steamline] Break LOCA" does not distinctly state the assumed sequence of events and operator actions for the following analyses: (a) 0.01 ft^2 SSLB with high pressure coolant injection (HPCI) available, (b) 0.01 ft^2 SSLB with HPCI not available, (c) greater than 0.01 ft^2 SSLB with HPCI available, and (d) greater than 0.01 ft^2 SSLB with HPCI not available. State the assumed sequence including the timing of operator actions for each of these analyses. Include answers to the following question in the response.

a. At what time from the accident initiation would the low pressure emergency core cooling system (ECCS) (i.e., RHR and CS) pumps automatically start operating, and would they operate in a low bypass flow return to the suppression pool, and for how long?

b. The starting of HPCI (for HPCI-available case), CS, and RHR systems is automatic according to their start logics and loading sequences. HPCI being a high pressure system would be replenishing RPV inventory earlier. State at what time in the sequence would the RHR and CS pumps supply water to the RPV? In the analysis, do HPCI, RHR and CS systems supply water simultaneously for some period, if so, for how much time?

c. What are the containment and RPV conditions when the operator would initiate RHR system in the drywell and wetwell spray mode?

d. For breaks greater than 0.01 ft^2, from what point in time and sequence in the event the drywell and wetwell spray is delayed by 1200 seconds to address the concern related to ECCS interruption?

e. For breaks greater than 0.01 ft^2, explain how the delaying of drywell and wetwell spray initiation by up to 20 minutes will address the concern related to ECCS interruption caused by a subsequent LOCA signal (activated on high drywell pressure concurrent with low RPV pressure).

f. Are the sequence of events assumed in the analyses consistent with the emergency operating procedures for breaks less than and greater than 0.01 ft^2.

g. Since the sequence of events assumed in the analysis differs for breaks less than and greater than 0.01 ft^2, are there different emergency operating procedures for less than and greater than 0.01 ft^2 breaks? How would the operator know whether the break is less than or greater than 0.01 ft^2.

<u>SCVB-RAI 8</u>

Section 2.6.5.2 of PUSAR fourth and fifth paragraphs under heading "Large Break LOCA Short-Term Phase ECCS NPSH [Net Positive Suction Head]," states:

> As stated above, the actual delivered pump flow rate will be between the safety analysis flow rate (where there is positive NPSH margin) and the pump run-out flow rate (where there may be negative NPSH margin). Because it is assumed that the operators take actions to control the RHR and CS pumps at ten minutes (see following subsection for evaluation of Large Break LOCA Long-Term Phase ECCS NPSH), this condition, should it occur, would exist for no more than ten minutes.
>
> During this ten minute period it is prudent to address two aspects of pump operation at these conditions: (1) whether the pump(s) could actually be operating with less than NPSHr [NPSH required]$_{3\%}$; and (2) whether the pump(s) could sustain any damage during this ten minute period. The SECY-11-0014 guidance (Reference 97) addresses these two concerns and the Boiling Water Reactors Owners Group (BWROG) provided in-depth assessments in two BWROG reports for the Browns Ferry RHR pumps: "Pump Operation at Reduced NPSHa [NPSH available] Conditions" (Reference 100) and "BWROG CVIC Report Task 4, Operation in Maximum Erosion Rate Zone" (Reference 101).

The above paragraphs gives the impression that the NPSHa is less than $NPSHr_{3\%}$ (i.e., pumps may be operating with a negative NPSH margin during the short-term). This is in conflict with data in Tables 2.6-4a and the graphs presented in Figures 2.6-11a and 2.6-12a, which show that both RHR and CS pumps have positive margin with respect to effective required NPSH ($NPSHr_{eff}$). In case the above tables and the figures are correct, it is not clear why the assessment in the BWR Owners' Group report "Containment Accident Pressure Committee (344), Task 3 – Pump Operation at Reduced NPSHa [available NPSH] Conditions (Sulzer Model CVIC Pump)," BWROG-TP-13-009, Revision 0, June 2013 (ADAMS Accession No. ML14077A097), is used to justify pump operation with a negative NPSH margin during the short-term. Confirm that in the short-term, with runout flow rates, the RHR and CS pumps operate with positive NPSH margin with respect to $NPSHr_{eff}$.

SCVB-RAI 9

Section 2.6.5.2 of PUSAR under heading "ECCS NPSH Summary" states:

> The ECCS strainer design debris load, which was used as an input to the strainer design, is documented Reference 38 [NEDC-32721-P-A]. The quantity and characterization of the strainer debris loading is based on the methodology in Reference 102.

GE Hitachi Nuclear Energy (GEH) letter (MFN-08-286) dated March 24, 2008 "Notification of Inaccurate Correlation in GE Hitachi Nuclear Energy (GEH) Licensing Topical Report NEDC-32721P-A, 'Application Methodology for the General Electric Stacked Disk ECCS Suction Strainer'" (ADAMS Accession No. ML080850242), reported inaccurate correlation for the strainer debris-bed head loss in the NEDC-32721-P-A.

Describe the methodology, correlation, and basis for the strainer debris-bed head loss calculation for available NPSH calculation during accident and special events.

SCVB-RAI 10

Section 2.6.5.2 of PUSAR, second paragraph under heading "Loss of RHR SDC [Shutdown Colling] ECCS NPSH" states:

> The assumption of HPCI operation is conservative for the determination of peak suppression pool temperature. However, HPCI pump suction from the suppression pool is limited to suppression pool temperatures below 140 °F, the maximum allowed temperature at Browns Ferry for HPCI operation.

In the suppression pool temperature response analysis for the loss of RHR SDC event, the HPCI suction source is assumed to be a suppression pool that its temperature varies from 95 °F to 140 °F during the event till the HPCI is isolated at its RPV isolation pressure or the pool temperature of 140 °F. The alternate suction source for the HPCI pump suction is condensate storage tank (CST) whose temperature is fixed at 130 °F (PUSAR, Table 2.6-2b). It does not appear that the assumption of HPCI operation is conservative for determination of the peak suppression pool temperature. Justify results of suppression pool temperature response for the case with HPCI suction from suppression pool and the case with HPCI suction CST. Note that the same comment applies to suppression pool temperature response analysis for the stuck open relief valve with RPV isolation event in which HPCI suction source is assumed to be suppression pool instead of CST, which is at 130 °F.

SCVB-RAI 11

Section 2.6.5.2 of PUSAR, last paragraph under heading "Loss of RHR SDC ECCS NPSH" states that the suppression pool temperature response analysis for the loss of RHR SDC event is also applicable for a small liquid break LOCA. Provide a description of the suppression pool temperature response analysis of the small liquid line break LOCA performed, including the liquid line and its size inside the drywell that would be most limiting. Explain why the suppression pool temperature response for the small liquid break LOCA, the RHR and CS NPSHa and NPSH margins would be bounded by the same parameters for the small steam break LOCA reported in Table 2.6-4 of PUSAR.

SCVB-RAI 12

Section 2.6.5.2 of PUSAR, third paragraph under heading "Small Break LOCA ECCS NPSH" states:

> As stated in Section 2.6.5.1, the suppression pool temperature response was evaluated for cases where HPCI was assumed not available and where HPCI was conservatively assumed available with suction from the suppression pool for the entire duration of the small break event. Because HPCI is only qualified for a suction temperature of up to 140 °F, the assumption of HPCI available with suction from the suppression pool during the entire event is not realistic.

Section 2.6.5.1 of PUSAR, second paragraph under heading "Suppression Pool Temperature Response –Small Steam [Steamline] Break LOCA" states:

> If HPCI is conservatively assumed available, HPCI will provide reactor inventory makeup until the reactor pressure decreases below the HPCI isolation pressure, after which low-pressure ECCS provides reactor inventory makeup.

During any SSLB LOCA (not necessarily the most limiting for peak suppression pool temperature), in case the suppression pool temperature exceeds 140 °F before the HPCI isolation pressure is reached, how is the HPCI operation prevented at higher pool temperatures?

SCVB-RAI 13

Section 2.6.5.2 of PUSAR, last sentence in second paragraph under heading "Fire Event ECCS NPSH" mentions the analysis performed for a sensitivity case to show increased NPSH margin during the fire event. The third paragraph states:

> The sensitivity case involved an analysis where a postulated 1,000 hp [horsepower] electric-driven Emergency High Pressure Makeup Pump (EHPMP) could be used as defense-in-depth to inject water from the CST through the FW piping and into the RPV while the RHR pump was operating in ASDC [Alternate Shutdown Cooling] mode. This effectively provides a means of pumping CST inventory through the RPV and into the torus to increase the suppression pool mass, providing more mass to accept the heat input from the RPV while at the same time increasing the suppression pool level which would increase the RHR pump NPSHa.

The statement does not confirm that the equipment is already installed or will be installed before EPU implementation for water addition to the suppression pool. Confirm that all equipment, piping, and connections on which the sensitivity case analysis is based are currently, or will be, installed in all BFN Units prior to EPU implementation; and state the plant system the equipment belongs to, or would belong to.

SCVB-RAI 14

Refer to Section 2.6.2 of PUSAR regarding the postulated break in a 4-inch jet pump instrument line nozzle at EPU conditions, and UFSAR Section 12.2.2.6.

UFSAR Section 12.2.2.6, eighth paragraph describes the effect of jet forces resulting from the 4-inch break line as follows:

> An analysis has also been performed of the effects of jet forces resulting from a double-ended break of the 4-inch line, assuming the jet forces from the break were to impinge directly on the removable plug. The resulting load would be 11 kips [kilo pounds], which is less than the capability of the locking bars and hinges, the capability of the shield wall, and the capability of the reactor vessel and its support skirt.

Provide a description of the analysis and results of the effect of the jet forces on the removable plug from a double-ended break of the 4-inch line under the most limiting thermal-hydraulic conditions in the reactor at EPU conditions. Justify that the reactor conditions (such as fluid density, jet velocity, etc.) assumed in the analysis are most limiting. What is the load capability of the locking bars and hinges, the capability of the shield wall, and the capability of the reactor vessel and its support skirt? Confirm these load capabilities bound the loads resulting from the jet forces from the double-ended break of the 4-inch line under EPU conditions.

SCVB-RAI 15

Refer to Section 2.6.2 of PUSAR and Section 12.2.2.6 of UFSAR,

Section 12.2.2.6 of UFSAR, 14th paragraph describes the effect of jet reaction force on the sacrificial shield wall (SSW) and its supports as follows:

> The jet load used in the design is the worst condition of either a clean break of the 26-inch main steam reactor pressure vessel penetration resulting in a jet reaction force of 595 kips, or a clean break of the 28-inch recirculating loop outlet penetration resulting in a jet reaction force of 658 kips.

a. What is meant by a clean break as compared to a double-ended break?

b. What is the load carrying capability of the SSW and its supports?

Section 2.6.2 of PUSAR does not address the jet reaction forces from the 26-inch main steam reactor pressure vessel penetration and the 28-inch recirculating loop outlet penetration on the SSW and its supports at EPU condition. Provide a description of the most limiting analysis and results of the jet reaction forces from a clean break and double-ended break of the same 26-inch main steam RPV penetration, the clean and double-ended break of the 28-inch recirculation loop outlet penetration, and the breaks of FW line under EPU conditions.

Justify that the conditions (such as fluid density, jet velocity, etc.) assumed in the analysis are most limiting. What is the capability of the SSW and its supports for carrying these loads?

SCVB-RAI 16

Section 2.6.1.5 of PUSAR states:

> The Browns Ferry analysis of record at CLTP shows that the recirculation line break DBA-LOCA [design basis loss-of-coolant accident] event causes boiling in the drywell cooling coils sooner than for steamline breaks at CLTP conditions. A comparison of the drywell temperature profiles during a DBA-LOCA for CLTP and EPU conditions shows a minor difference in temperature (less than 2 °F) and lasting only seconds. Based on the minor differences in the drywell temperature profiles and the margin between the RBCCW [reactor building closed cooling water] pump start (40 seconds) and calculated time to boil (61 seconds), it is concluded that voiding will not occur under EPU conditions prior to the restart of a RBCCW pump.

Table 2.6-1 of PUSAR shows peak drywell temperature of 297 °F for Units 2 and 3, and 295.2 °F for Unit 1 in the current licensing basis analysis. Table 2.6-1 shows 336.9 °F EPU long-term drywell gas temperature for the SSLB. In the current response to Generic Letter (GL) 96-06, the boiling time in the drywell cooling coils of 61 seconds should be based on the short-term drywell gas temperatures.

In the current response to GL 96-06, what are the drywell gas and the RBCCW temperatures used in the analysis to determine the boiling time of 61 seconds in the drywell cooling coils? State the basis for these temperatures and confirm that the both are limiting for the boiling time of 61 seconds in the RBCCW piping and the drywell cooler cooling coils. Provide the same temperature under the EPU conditions with justification that these are most limiting for the boiling time determination in the drywell cooling coils.

SCVB-RAI 17

Section 2.6.1.5 of PUSAR provides the current and proposed responses to GL 96-06 for the demineralized water system, drywell floor drain sump discharge, drywell equipment drain sump discharge, and reactor water sampling system. For these systems, provide the penetration numbers in UFSAR Table 5.2.2 for the piping between which containment isolation valves are under consideration for GL 96-06.

SCVB-RAI 18

Section 2.6.1.5 of PUSAR states that the drywell floor and drywell equipment drain sump discharge lines are acceptable because a 0.06-inch diameter orifice has been drilled in each discharge check valve. In UFSAR Table 5.2-2, the containment isolation valves listed for the drywell floor and equipment drain sump discharge are 3-inch ball valves (penetration Nos. X-18, and X-19). Describe the location of the check valve in relation to the 3-inch ball valves and describe how the 0.06-inch orifice in the check valve would relieve the pressure between the containment isolation valves (3-inch ball valves).

SCVB-RAI 19

Refer to Section 2.6.1.5 of PUSAR; confirm that all system lines for the containment isolation valves listed in UFSAR Table 5.2-2 have been evaluated for the GL 96-06 over-pressurization issue under EPU conditions.

SCVB-RAI 20

Section 2.6.1.5 of PUSAR states that thermally induced over-pressurization of the reactor sampling system at the current power using a constant drywell temperature of 336 °F following a LOCA reaches approximately 2,546 pound per square inch gauge (psig), which will lift the inboard globe isolation valve disc.

a. What is the design pressure of these valves and the interfacing piping on both sides?

b. Are these valves designed to lift at 2,546 psig?

SCVB-RAI 21

PUSAR, Section 2.6.1.5, describe the controls in place that will also be used under EPU conditions for draining the demineralized water system header prior to power operation in each cycle.

SCVB-RAI 22

Section 2.6.1.1.2 of PUSAR provides a discussion of the impact of drywell-to-wetwell steam bypass and concluded that EPU requires no change to the existing Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.1.2 that is to detect flow paths between the drywell and wetwell whose total capacity is equal to or greater than the capacity of a 1-inch diameter plate orifice (a 1-inch plate orifice has an effective area capability, $A/\sqrt{K}$, of approximately 0.0033 ft^2).

As per NUREG-0800 (Standard Review Plan), Section 6.2.1.1.C, Revision 7, Appendix A, Section B, item 2.c, the acceptance criterion for Mark I containments with regard to steam bypass leakage, is that the measured leakage should not be greater than the leakage that would result from a 1-inch diameter opening ($A/\sqrt{K}$ approximately 0.0033 ft^2). What is the value of $A/\sqrt{K}$ derived from the SR 3.6.1.1.2, which reads as follows?

> Verify drywell to suppression chamber differential pressure does not decrease at a rate > 0.25 inch water gauge per minute over a 10 minute period at an initial differential pressure of 1 psid [pounds per square inch differential].

SCVB-RAI 23

Section 2.6.1.2.1 of PUSAR provides an evaluation of increase fatigue usage factor of wetwell components due to an increased duration in chugging cycles from 900 seconds to 1200 seconds. Clarify that this evaluation is also applicable to the EPU conditions.

SCVB-RAI 24

Section 2.5.3.2 of PUSAR under heading "Browns Ferry Current Licensing Basis" states:

> Browns Ferry's current licensing basis regarding GL 89-13 is discussed in TVA's response to the NRC by letter dated March 16, 1990, "Response to Generic Letter 89-13 Service Water Problems Affecting Safety-Related Equipment.

In the TVA response letter to GL 89-13, dated March 16, 1990, the response to NRC recommended action II states:

> Inspect and clean the cooling water side of the RHR heat exchangers at least once per cycle.

In Reference 7, response to SCVB-RAI 1(b), under title "Frequency of Monitoring - Performance Testing," states:

> Each RHR heat exchanger will have been performance tested at least once and will be tested periodically at an interval that initially will not exceed five years.

In Reference 7, response to SCVB-RAI 1(b), under title "Frequency of Monitoring - Visual Inspection and Cleaning" for EPU implementation states:

> Each RHR heat exchanger will be cleaned once every 8-years or more frequently if the trended fouling rate indicates the need to take corrective actions in order to maintain the heat exchanger condition within the fouling resistance acceptance criteria.

As stated above, the inspection and cleaning interval of the RHR heat exchangers in TVA's response to GL 89-13 was at least once every cycle (i.e., at least 2 years) as committed in TVA letter dated March 16, 1990. Currently the inspection and cleaning is interval is 4 years. The inspection and cleaning interval after EPU implementations is being proposed to increase to 8 years.

a. Provide the basis for increasing the inspection and cleaning interval from 2 years to the current interval of 4-years, and

b. Provide the basis for the proposed increase in interval from 4 years to 8 years after EPU implementation. Justify the increase in intervals based on the trending history of the previous as-found inspection and cleanliness condition results of the RHR heat exchangers, such as the as-found number of partial and wholly plugged tubes, and/or any other parameter that reflected the cleanliness of the as-found heat exchanger.

SCVB-RAI 25

In Reference 4, the implementation Item 49, which is a license condition states:

> Revise the program that monitors BFN Residual Heat Removal (RHR) heat exchanger performance for consistency with the assumptions of the NFPA [National Fire Protection Association] 805 Net Positive Suction Head (NPSH) analysis. The monitoring program shall include verification that the tested worst fouling resistance, with measurement uncertainty added, of all BFN Units 1, 2, and 3 RHR heat exchangers is less than the design value of 0.001517 hr-ft^2 - °F/BTU and the worst tube plugging is less than 4.57 percent.

Provide the proposed revision to the above licensee condition for EPU implementation including the frequency of cleaning and testing of the RHR heat exchangers.

SCVB-RAI 26

Provide the proposed revision to the GL 89-13 response applicable to the RHR heat exchanger for EPU implementation.

SCVB-RAI 27

Section 2.6.6 of the PUSAR, under "Technical Evaluation," states:

> Because the maximum dome pressure is also not changed for EPU, there is no effect to the ability of secondary containment to contain mass and energy released to it. There is no increase in mass and energy released to secondary containment for EPU.

a. Explain, what is meant by "ability of secondary containment to contain mass and energy released to it."

b. Which break mass and energy released in the secondary containment is being referred to in the above statement?

c. What is the relationship between the reactor dome pressure and the mass and energy released to the secondary containment?

SCVB-RAI 28

Table 2.6-1 of PUSAR states peak drywell temperature of 297 °F for Units 2 and 3, and 295.2 °F for Unit 1 in the current licensing basis analysis. Table 2.6-1 shows 336.9 °F for the EPU long-term drywell gas temperature for the SSLB accident, which is a significant change from the current drywell peak temperatures. The statement in Section 2.6.6 of PUSAR "The secondary containment temperature and pressure are not evaluated further in the Constant Pressure Power Uprate Licensing Topical Report because there is no effect as a result of EPU" does not appear to be correct because of large change in the drywell temperature.

a. Provide the secondary containment pressure and temperature response under EPU conditions. State all the assumptions and analysis inputs that differ from the current licensing basis.

b. Provide for the Standby Gas Treatment System drawdown time, flow capacity, and iodine removal capacity based on the secondary containment conditions at EPU.

SCVB-RAI 29

In Reference 6 under the Introduction section states:

> The PUSAR uses GEH GE14 fuel as the principal reference fuel type for the evaluation of the impact of EPU. However, the BFN units will utilize AREVA ATRIUM 10XM fuel, with the potential for some legacy ATRIUM 10 fuel, under EPU conditions.

The various containment analysis presented in Section 2.6 of PUSAR are based on the sensible and decay heat from GE14 fuel. There is no quantitative comparison provided between the fuel characteristics for GE14 and the ATRIUM 10XM or ATRIUM 10 fuel from a containment analysis standpoint.

Considering the differences between GE14 and ATRIUM 10 XM or ATRIUM 10 fuel such as mass, material properties, core flow, decay heat, heat transfer coefficients between the core and the coolant, and any other variations, justify the LOCA mass and energy release in the containment based on GE14 fuel is bounding and the various containment analysis based on GE14 fuel mass and energy release presented in Section 2.6 of PUSAR will remain bounding.

SCVB-RAI 30

As per Electric Power Research Institute (EPRI) Report 107397, in the nuclear industry, test uncertainty is often determined with 95 percent coverage. Per American Society of Mechanical Engineers Operation and Maintenance (ASME OM)-21, "A 95% confidence level shall be applied to the calculated result for the purpose of comparing the testing or monitoring results to the acceptance criteria."

a. Provide a detailed discussion of how the uncertainty analysis approach described in Section 3.9.2 of EPRI Report 107397 was applied in determining the fouling resistance results for the 2C, 3A, 3C, 2C, and 2A RHR heat exchangers given Table 4 of Reference 6.

b. The fouling resistance result that should be less than the acceptance criteria is defined as [fouling resistance result = measured fouling factor + test and measurement uncertainty]. Provide the values of test and measurement uncertainties that were added to the measured fouling resistances for determining the fouling resistances of the RHR heat exchangers given in Table 4 of Reference 6.

SCVB-RAI 31

In PUSAR Section 3, "References," the Reference 88, "GE Nuclear Energy, 'Mark I Containment Program - Full Scale Test Program, Final Report,'" NED0-24569, August 1979 does not appear to be a correct reference. Is it NEDO-24539 instead of NEDO-24569? Clarify or correct.

SCVB-RAI 32

TVA, in response to SCVB RAI-1(e), states that the performance testing will be conducted consistent with the guidance described in EPRI 3002005340, "Service Water Heat Exchanger Testing Guidelines," May 2015.

The response to SCVB RAI-1(j) states:

> The data analysis and preparation of vendor test reports are performed by a vendor also operating under a 10 CFR 50 Appendix B Quality Assurance Program in accordance with approved procedures that include steps to compare process and tube side heat transfer rates and to statistically evaluate test data such that results conservatively account for the uncertainties associated with each test. The method of calculation follows the EPRI guidelines (see response (e), above) in terms of determining the uncertainty contributors of precision and bias errors for thermal performance test evaluations. The uncertainty analysis methodology in PROTO-HX determines the sensitivity coefficients through a numerical approach using the central differencing method (i.e., symmetric uncertainties). The EPRI guideline (see response (e), above) provides an overview of this approach. The variables considered are the test data (flow rates and temperatures) and the film coefficients.

Provide the following information:

a. Describe the statistical evaluation method of the test data with discussion of how the uncertainty analysis methodology described in Chapter 4 of EPRI 3002005340 is implemented for evaluating the Random Standard Uncertainty and Systematic Standard Uncertainty combined together.

b. Justify that the Combined Standard Uncertainty (combination of Random Standard Uncertainty and Systematic Standard Uncertainty) in the test result is conservative as mentioned in the above response.

SCVB-RAI 33

Refer to the following assumption in response to SCVB RAI-1(l), regarding the method of as-found heat-exchanger inspection for determining the number of plugged tubes and the resulting effective heat transfer area.

> It is assumed that all tubes were unobstructed during the test (i.e., none of the tubes were plugged by macro fouling during the test). It is an inherent part of the

PROTO-HX method of analysis to distribute the tube-side flow equally to all tubes and to use the specified heat transfer area in the fouling calculation. This assumption is acceptable since lost area due to unknown macrofouling will show up as extra fouling resistance.

a. According to the above assumption, since the number of plugged tubes is unknown, the PROTO-HX analysis would use the cross-sectional area of all tubes, which is obviously greater than the actual cross-sectional flow area during the test because of plugged tubes. The tube flow velocity calculated by PROTO-HX would, therefore, be less than the actual tube flow velocity in the test. Explain how the difference in tube velocity is accounted for in determining the tube side heat transfer coefficient.

b. According to the above assumption, since the number of plugged tubes is unknown, the PROTO-HX analysis would use the clean heat exchanger maximum design heat transfer area of all tubes, which is obviously greater than the actual heat transfer area during the test. Justify quantitatively the equivalency [emphasis added] of lost heat transfer area due to unknown macrofouling (unknown number of plugged tubes) with showing up of extra fouling resistance so that the heat exchanger overall heat transfer coefficient (UA) output from the PROTO-HX analysis is conservative (minimum) and bounded by the actual UA of the tested heat exchanger.

SCVB-RAI 34

Provide the following latest as-found test results of all BFN Units 1, 2, and 3 RHR heat exchangers:

Parameter	**UNIT 1**				**Unit 2**				**UNIT 3**			
	1A	**1B**	**1C**	**1D**	**2A**	**2B**	**2C**	**2D**	**3A**	**3B**	**3C**	**3D**
Test date												
Number of years in operation after last cleaning												
Hot side inlet temperature (°F)												
Hot side outlet temperature (°F)												
Hot side flow rate (pound mass/hour)												

Parameter	UNIT 1				Unit 2				UNIT 3			
	1A	1B	1C	1D	2A	2B	2C	2D	3A	3B	3C	3D
Cold side inlet temperature (°F)												
Cold side outlet temperature (°F)												
Cold side flow rate (pound mass/hour)												
Test heat transfer rate (BTU/hr)												
Test U (BTU/ hr-ft^2-°F)												
Test UA (BTU/hr-°F)												
Test fouling resistance (hr-ft^2-°F/BTU)												
Uncertainty in fouling (hr-ft^2-°F/BTU) resistance (hr-ft^2-°F/BTU)												
Acceptance criteria for fouling resistance (hr-ft^2-°F/BTU)												
Margin in fouling resistance (hr-ft^2-°F/BTU)												
If inspected, number of fully blocked tubes												
If inspected, number of partially blocked tubes												

Parameter	UNIT 1				Unit 2				UNIT 3			
	1A	1B	1C	1D	2A	2B	2C	2D	3A	3B	3C	3D
Acceptance criteria for blocked tubes												

REFERENCES

1. Letter from TVA to NRC dated September 21, 2015, "Proposed Technical Specifications Change TS-505 – Request for License Amendments – Extended Power Uprate" (ADAMS Accession No. ML15282A152).

2. PUSAR - Attachment 6 to Reference 1, "NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate (proprietary [non-public])" ADAMS Accession No. ML15282A264. Attachment 7 (ADAMS Accession No. ML15282A181) contains public version of Attachment 6.

3. Letter from TVA to NRC dated June 13, 2014, "Response to NRC Request for Additional Information Regarding the License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (TAC Nos. MF1185, MF1186, and MF1187) - Attachment X and Fire Modeling" (ADAMS Accession No. ML14167A175).

4. Letter from TVA to NRC dated October 20, 2015, "Update to License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (TAC Nos. MF1185, MF1186, and MF1187) - Revised Implementation Item 49" (ADAMS Accession No. ML15293A527).

5. Attachment 39 to Reference 1, "Heat Exchanger K Values Utilized in EPU Containment Analyses" (ADAMS Accession No. ML15282A235).

6. Attachment 8 to Reference 1, "ANP-3403P, Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3 (Proprietary [non-public])" ADAMS Accession No. ML15282A268). Attachment 9 (ADAMS Accession No. ML15282A182) contains public version of Attachment 8.

7. TVA Letter to NRC dated March 28, 2016, "Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU) - Supplement 8, Response to Request for Additional Information" (ADAMS Accession No. ML16089A054).

TVA PROPOSED RESPONSE SCHEDULE

Request for Additional Information (RAI) Question Number	Due Date
SCVB-RAI 1, Revision 1[1]	July 25, 2016
SCVB-RAI 2	June 24, 2016
SCVB-RAI 3	June 24, 2016
SCVB-RAI 4	June 24, 2016
SCVB-RAI 5	June 24, 2016
SCVB-RAI 6	June 24, 2016
SCVB-RAI 7	June 24, 2016
SCVB-RAI 8	July 25, 2016
SCVB-RAI 9	June 24, 2016
SCVB-RAI 10	June 24, 2016
SCVB-RAI 11	June 24, 2016
SCVB-RAI 12	June 24, 2016
SCVB-RAI 13	June 24, 2016
SCVB-RAI 14	July 25, 2016
SCVB-RAI 15	June 24, 2016
SCVB-RAI 16	June 24, 2016
SCVB-RAI 17	June 24, 2016
SCVB-RAI 18	June 24, 2016
SCVB-RAI 19	June 24, 2016
SCVB-RAI 20	June 24, 2016

[1] TVA revises its response to SCVB-RAI 1.

Request for Additional Information (RAI) Question Number	**Due Date**
SCVB-RAI 21	June 24, 2016
SCVB-RAI 22	June 24, 2016
SCVB-RAI 23	June 24, 2016
SCVB-RAI 24	July 25, 2016
SCVB-RAI 25	July 25, 2016
SCVB-RAI 26	July 25, 2016
SCVB-RAI 27	June 24, 2016
SCVB-RAI 28	June 24, 2016
SCVB-RAI 29	June 24, 2016
SCVB-RAI 30	July 25, 2016
SCVB-RAI 31	June 24, 2016
SCVB-RAI 32	July 25, 2016
SCVB-RAI 33	June 24, 2016
SCVB-RAI 34	July 25, 2016
CAP Credit Elimination LAR Supplement (PUSAR and Attachment 39)	July 25, 2016

Summary

June 24, 2016: SCVB-RAI 2, 3, 4, 5, 6, 7, 9, 10, 11, 12, 13, 15, 16, 17, 18, 19, 20, 21, 22, 23, 27, 28, 29, 31, 33

July 25, 2016: SCVB- RAI 1, 8, 14, 24, 25, 26, 30, 32, 34, CAP Credit Elimination LAR Supplement

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
Enclosures:
1. Request for Additional Information
2. Proposed TVA Response Schedule

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