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

January 25, 2018

Mr. Joseph W. Shea, Vice President
Nuclear Regulatory Affairs
and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 5.5.12 "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NOS. MG0113, MG0114, AND MG0115; EPID L-2017-LLA-0292)

Dear Mr. Shea:

By letter dated August 15, 2017, Tennessee Valley Authority (TVA or the licensee) submitted a license amendment for Browns Ferry Nuclear Plant Units 1, 2, and 3. The proposed amendment would revise Browns Ferry's Technical Specification 5.5.12 "Primary Containment Leakage Rate Testing Program," by adopting Nuclear Energy Institute (NEI) 94-01, Revision 3 A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," as the implementation document for the performance-based Option B of 10 CFR Part 50, Appendix J. The proposed changes would allow the licensee to extend the Type A containment integrated leak rate testing interval from 10 to 15 years and the Type C local leakage rate testing intervals from 60 to 75 months.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the licensee's submittal and determined that additional information is needed. On December 11 and 12, 2017, the NRC staff forwarded, by electronic mails, draft requests for additional information (RAIs) to TVA. On December 21, 2017, the NRC staff held two conference calls to provide the licensee with the opportunity to clarify any portion of the draft RAIs and discuss the timeframe for which TVA would provide the requested information. The finalized NRC staff's RAIs were as discussed with the TVA staff are provided in Enclosures 1 and 2.

By an email dated December 27, 2017, Mr. Thomas Hess of your staff provided TVA's proposed response dates to the NRC staff's RAIs (Enclosure 3). TVA proposed to provide its responses to certain RAIs by March 27, 2018, which is 3 months (90 calendar days) after the date of its email. Specifically, TVA staff stated that additional time is required for those RAI responses due to significant additional analysis. The NRC staff agreed with the TVA proposed dates, noting that no extension to these dates will be accepted by the staff.

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:

1. RAI from Containment and Plant Systems Branch
2. RAI from Probability Risk Assessment Licensing Branch
3. TVA Proposed Response Dates

cc w/enclosure: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 5.5.12

"PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (BFN-TS-497)

TENNESSEE VALLEY AUTHORITY

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

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

By letter dated August 15, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17228A490), Tennessee Valley Authority (TVA or the licensee) submitted a license amendment request (LAR) for Browns Ferry Nuclear Plant (BFN or Browns Ferry) Units 1, 2, and 3. The proposed amendment would revise Browns Ferry's Technical Specification (TS) 5.5.12 "Primary Containment Leakage Rate Testing Program," by adopting Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," as the implementation document for the performance-based Option B of 10 CFR Part 50, Appendix J. The proposed changes would allow the licensee to extend the Type A containment Integrated leakage rate testing (ILRT) interval from 10 to 15 years and the Type C local leakage rate testing (LLRT) interval from 60 to 75 months.

The U.S. Nuclear Regulatory Commission (NRC) staff from the Office of Nuclear Reactor Regulation, Division of Safety Systems, Containment and Plant Systems Branch (SCPB) reviewed the information provided by TVA and determined that additional information as discussed below is needed to complete its review.

SCPB RAI-1

All historical ILRT leakage rate values in the table of the LAR Section 4.2 "Integrated Leak Rate Test History" (Enclosure 1, page E1-12 of 39) are below the limits of BFN TS 5.5.12. However, the test pressure values associated with the BFN historical ILRT as-found leakage rates were not included in the LAR.

Section 4.0 "Limitations and Conditions" of the NRC staff's Safety Evaluation (SE) associated with NEI 94-01, Revision 2-A (ADAMS Accession No. ML100620847) established the requirements for extending the ILRT interval beyond 10 years as will be implemented with the NRC approval of the August 15, 2017, LAR. Section 3.1.1.1 "Type A Performance Leakage Rate" of the NRC staff SE states, in part:

> Acceptable performance history is defined as successful completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 L_a [the maximum allowable Type A test leakage rate at P_a, where P_a equals the calculated peak containment internal pressure related to the design-basis loss-of-coolant accident].

Section 3.2.11, "Type A Test Pressure," of American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-1994, "American National Standard for Containment System Leakage Testing Requirements" (ADAMS Accession No. ML11327A024), states, in part:

> The Type A test pressure shall not be less than $0.96P_{ac}$ [calculated peak accident containment internal pressure, also defined as P_a above] nor exceed P_d [containment design pressure].

Provide the test pressure values for the two most recent as-found Type A tests for BFN and state if these values satisfy the requirements of NEI 94-01, Revision 2-A, and Section 3.2.11 of ANSI/ANS 56.8-1994. Also explain how these test pressure values relate to the BFN revised P_a values associated with License Amendment Nos. 299, 323, and 283 (ADAMS Accession No. ML17032A120) pertaining to the extended power uprate LAR.

<u>SCPB RAI-2</u>

Both NEI 94-01 Revision 0 (current license basis) and Revision 3-A (LAR requested license basis) Sections 10.2.1.4, and 10.2.3.4 address corrective actions for unacceptable Type B and Type C test results, respectively. It is stated in these documents, in part:

> . . . a cause determination should be performed and corrective actions identified that focus on those activities that can eliminate the identified cause of failure with appropriate steps to eliminate recurrence.

This information is not provided in the Section 4.3 "Type B and Type C Testing Programs," of the LAR dated August 15, 2017. Please identify and provide details regarding "worst case" repetitive failures of administrative limits for LLRTs associated with any Type B or Type C penetration test results. Also, identify and provide the details of all corrective actions performed to prevent recurrence.

a. For BFN Unit 1, since the last ILRT of 2010

b. For BFN Unit 2, since the last ILRT of 2009

c. For BFN Unit 3, since the last ILRT of 2012

<u>SCPB RAI-3</u>

LAR Section 4.3 "Type B and Type C Testing Programs" states, in part:

> Each unit has two airlock doors, 30 individual bellows tests, 31-33 electrical penetrations, 16 resilient seal and hatch type penetrations, and 20 piping flanges that are LLRT (Type B) tested. Additionally, the entire HPCI [high-pressure coolant injection] and RCIC [reactor core isolation coolant] exhaust vacuum breaker network lines for all three units are Type B tested. BFN Unit 1 has 43 penetration / pathways with 99 isolation valves, and BFN Units 2 and 3 have 45 penetration / pathways with 102 isolation valves that are LLRT (Type C). Currently, approximately 9.4% of the components (all units combined) are tested at the 30-month nominal interval due to performance.

For an established Appendix J, Option B, LLRT program with a sufficient historical base, the percentage of Type B or Type C components on repetitive frequencies can indicate the quality of the maintenance program and corrective action process. Provide the following information for BFN Units 1, 2, and 3:

a. The total number (i.e., population) and percentage of the total number of BFN Type B tested components currently on a 120-month extended performance-based test interval,

b. The total number (i.e., population) and percentage of that total number of BFN Type C Containment Isolation Valves (CIVs) currently on a 60-month extended performance-based test interval,

c. Discuss how the percentages reported in a. and b. above support an extended test interval of up to 75 months for Unit 1, Unit 2, and Unit 3 Type C tested CIVs in accordance with the guidance of NEI 94-01, Revision 3-A.

<u>SCPB RAI-4</u>

Item 4 under "Limitation/Condition" of Enclosure 1 (page E1-6 of 39) to the LAR is derived from Sections 3.1.4 and 4.1 of the NRC SE associated with NEI 94-01, Revision 2-A, which stipulates that the licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable.

This "Limitation/Condition" is intended to verify any <u>past</u> major modification, maintenance, or repair of the primary containment since the last ILRT has been appropriately accompanied by either a structural integrity test (SIT) or ILRT and that any plans for future major modifications also include appropriate pressure testing.

However the "TVA Response" states:

> "Any future containment modifications will be addressed by the site design change process including any containment post-modification testing as required by Section 3.1.4 of the NRC staff SE for NEI 94 01, Revision 2."

The above TVA response is forward looking with respect to plans for any future BFN containment modifications.

Identify (i.e., provide a synopsis of) any containment major modification and post-modification testing performed since the most recent ILRT for each unit. This synopsis should demonstrate a consistency with the guidance of NEI 94-01, Revision 2, NRC staff SE Section 3.1.4.

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 5.5.12

"PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (BFN-TS-497)

TENNESSEE VALLEY AUTHORITY

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

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

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The U.S. Nuclear Regulatory Commission (NRC) staff from the Office of Nuclear Reactor Regulation, Division of Risk Assessment, Probabilistic Risk Assessment (PRA) Licensing Branch A (APLA) reviewed the information provided by TVA and determined that additional information as discussed below is needed to complete its review.

APLA RAI-01

In the LAR dated August 15, 2017, the licensee specified that the risk assessment evaluation performed to support the proposed change follows the guidelines of NEI 94-01, Revision 3-A "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, dated July 2012 (ADAMS Accession No. ML12221A202) and the Electric Power Research Institute (EPRI) Report 1018243, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," Revision 2-A of EPRI 1009325, dated August 1994.

a. EPRI Report 1018243 described Class 2 sequences as large containment isolation failures. This group consists of all core damage accident progression bins for which a pre-existing leakage due to failure to isolate the containment occurs. These sequences are dominated by failure to close of large (larger than 2 inches [5.1 cm] in diameter) containment isolation valves.

   The licensee, in Table 3 of the LAR, describes Class 2 as "Dependent failure modes, or common cause failures" and provides further interpretation for the BFN assigning of Class 2 sequences as isolation faults that are related to a loss of power or other isolation failure mode that is not a direct failure of an isolation component. Confirm that Class 2

Large Containment Isolation Failures, as described in Section 8.1.2 of the LAR, is consistent with the EPRI Report 1018243.

b. Section 4.2.7, External Events, of the EPRI Report 1018243 states in part,

> If the external event analysis is not of sufficient quality or detail to allow direct application of the methodology provided in this document, the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed. This assessment can be taken from existing, previously submitted and approved analyses or another alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

To estimate the external events (EEs) Large Early Release Frequency (LERF), the licensee applied (in Sections 10.2 and 10.3 of the LAR) the internal events (IEs) LERF to core damage frequency (CDF) ratio. Since external events can subject plants to common-cause failures of multiple structures, systems, and components, it is possible that the LERF to CDF ratio for the EEs will be higher than for the IEs. Provide justification for assuming that the IEs' CDF to LERF ratio also applies to the seismic CDF to LERF ratio for EEs. In addition, explain the basis for using the IEs' LERF to CDF ratio to the EEs, specifically for seismic and high winds.

APLA RAI-02 [Fire Probabilistic Risk Assessment (FPRA)]

Section 2.5.5, "Comparisons with Acceptance Guidelines," of Regulatory Guide (RG) 1.174 states that when the contributions from the contributors modeled in the PRA are close to the risk acceptance guidelines, the argument that the contribution from the missing items is not significant must be convincing and in some cases may require additional PRA analyses (e.g., bounding analyses, detailed analyses, or by a demonstration that the change has no impact on the unmodeled contributors to risk). When the margin is significant, a qualitative argument may be sufficient. In addition, Section 2.5.3, "Model Uncertainty," of RG 1.174 states that the impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments.

Table 38, "Peer Reviews" of the LAR outlines Fire (Focused Scope) and Internal Events 2009 Fact and Observation (F&O) Resolution Review (Focused Scope) that occurred in May 2015 and July 2015, respectively. Furthermore, the licensee states that the 2015 Fire PRA (FPRA) peer review focused on specific aspects of the FPRA that had changed including updated PRA methodologies/approaches. However, there have been numerous changes to FPRA methodology since the staff's review of the BFN FPRA National Fire Protection Association Standard 805 (NFPA 805) LAR and issuance of the safety evaluation that may be relevant for the FPRA to include such as the following:

- The NRC issued a letter, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires'" (ADAMS Accession No. ML12171A583), June 21, 2012, providing staff positions on 1) frequencies for cable fires initiated by welding and cutting, 2) clarifications for transient fires, 3) alignment factor for pump oil fires, 4) electrical cabinet fire treatment refinement details, and 5) the EPRI 1022993 report.

- The NRC published NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 2, which is supported by a letter from the NRC to NEI, "Supplemental Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis" (ADAMS Accession Nos. ML14086A165 and ML14017A135).

- The NRC published NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009" (ADAMS Accession No. ML15016A069).

- Guidance on the credit taken for very early warning fire detection system is available in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)" (ADAMS Accession No. ML16343A058). The guidance provided in Frequently Asked Question 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML093220426), has been rescinded.

In review of the BFN LAR, the calculated total LERF that includes EE is close to the RG 1.174 risk acceptance criteria. However, the integration of NRC-accepted FPRA methods and studies described above that are relevant to this submittal could potentially result in an exceedance of the risk acceptance guidelines. For example, previous risk-informed LARs have shown that integration of NRC-approved methods can lead to a calculated risk increase of up to approximately a factor of 3 in some cases. Therefore, in accordance with Section 2.5.5 of RG 1.174, additional analysis is necessary to ensure that contributions from this influence would not change the conclusions of the LAR.

Provide a detailed justification for why the integration of NRC-approved FPRA methods and studies would not change the conclusions of the LAR. As part of this justification, identify the FPRA methodologies used in the FPRA that have not been formally accepted by the NRC staff, provide technical justification for their use and evaluate the significance of their use on the risk metrics for the application (total LERF, ΔLERF, Population Dose Rate (PDR), and conditional containment failure probability (CCFP)). Provide updated tables that include the increase in total LERF (IEs and EEs), ΔLERF, PDR, and CCFP for each unit to assess the risk impact as appropriate.

<u>APLA RAI-03</u> [Internal Events and FPRA: Facts and Observations (F&Os)]

LAR, Tables 50 and 51, "Internal Events PRA F&O Resolution" and "FPRA F&O Resolution," provide resolution of the peer review F&Os and their impacts on the application for the IEs and FPRAs. Address the following:

a. F&O 1-17 related to Supporting Requirement (SR) DA-C6 identified that post-maintenance testing (PMT) demands were not excluded from the count of plant-specific demands on standby components. The SR states that additional demands from post-maintenance testing should not be counted because they are part of the successful renewal. In resolution to this F&O the licensee stated that it performed an analysis to quantify the effect that removal of potential PMT would have on the results by analyzing seven scenarios. It further states that "the results show that even with an extremely unrealistic number of PMTs the data is not significantly skewed by the inclusion of the PMT data." Describe how the analysis was performed and provide the results. In addition, explain how the treatment of PMT demands contributes to under- or over-estimation of failure probabilities, the CDF and LERF estimates, and the subsequent risk metrics for ILRT acceptance (i.e., total LERF, ΔLERF, CCFP).

b. F&O 4-18, associated with SR HR-G2 identified that for some operator actions the execution failure probability is assumed to be zero. SR HR-G2 states to use an approach to estimate Human Error Probabilities (HEPs) that addresses failure in cognition as well as failure to execute. The resolution states that "TVA staff considered plant data and judged that the most recent history is most applicable of the current as-operated plant. It is justifiable to screen "break-in period" events from the history of a stably operating plant. BFN justification is judged adequate and appropriate." In support of NRC review:

   1. Provide a summary of the data considered and explain how exclusion of the Human Failure Events (HFE) execution was determined to be adequate and appropriate for the HEP and its impact to the ILRT risk metrics for acceptability.

   2. In addition, if determined that there exists potential impact on the application, provide a sensitivity evaluation that demonstrates the effect of exclusion of the HFE execution is negligible relative to the ILRT risk metrics.

c. F&O IFNS-A1-01 identified that a screening value of 0.1 was used for the failure of the door to the air conditioning equipment room to perform its intended function, without proper justification. In resolution to this F&O the licensee stated that it revised the analysis to include supporting information pertaining to the air equipment room door design characteristics and physical location that would describe the likely failure of the door in the event of flood accumulation in the room. A screening factor of 0.1 was determined to be conservative and used for the double glass double door emergency exit.

   To validate the use of the screening value of 0.1 from the expert judgement applied, confirm that use of this value (0.1) does not screen out any scenarios that could potentially be non-negligible risk contributors for IEs. For any identified scenarios provide justification why the screening value is appropriate and confirm that the risk metrics for the application are still met.

d. F&O 6-8 related to SR IE-C8 identified that the loss of Raw Cooling Water (RCW) initiating event appeared to be reduced by an incorrectly calculated factor and for combinations where it is potentially not valid. BFN provided a resolution that described the RCW success criteria and stated that the frequency is calculated by summing 1) all combinations of a failure of three pumps and 2) all combinations of the failure of a single running pump and the failure of two additional pumps included in the system initiating event model.

   RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (ADAMS Accession No. ML090410014) describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. RG 1.200 references

American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," which was issued as at-power Level 1 and limited Level 2 PRA standard. RG 1.200 refers to this standard as one acceptable approach to demonstrate conformance with Regulatory Position 1, which is to use a national consensus PRA standard or standards that address the scope of the PRA used in the decision-making. ASME/ANS RA-Sa-2009 provides both process and technical requirements for an at-power Level 1 and limited Level 2 PRA for IEs, internal flood, internal fire, seismic, wind, external flood and other EEs.

Specifically SR IE-C8, in Table 2-2.1-4(c) of the ASME/ANS RA-Sa-2009 PRA Standard, states that "some initiating events are amenable to fault-tree modeling as the appropriate way to quantify them. These initiating events, which usually support system failure events, are highly dependent upon plant-specific design features. If fault-tree modeling is used for initiating events, USE the applicable systems-analysis requirements for fault-tree modeling found in Systems Analysis (2-2.4)."

Confirm what factor was used for the BFN RCW initiating event frequency, and provide justification to support why the factor used is appropriate.

e. Fire F&O 2-50, associated with SR HRA-C1 identified that, for instances where cues for human actions involve multiple individual instrumentation devices, they are modeled in the PRA as multiple inputs to a single AND gate. In this model, the availability of any single instrument, with the majority of the other instruments failed, would not disable the human action. The peer review team determined that the licensee did not consider or confirm the development of operator guidance that would allow operators to discern which instrument is not impacted by the fire. In resolution to this F&O the licensee stated that the AND gate was maintained with an assumption that the fire procedures will include the impacted instrumentation for fires in the respective area. Therefore, as long as one instrument is available and the operators can determine from the applicable fire procedure which instrument is available, that instrument can be credited. Retaining this modeling as an AND gate for instrumentation failures, given the development of procedures that will guide the operator regarding which, if any, instrument remains functional, appears non-conservative. Perform a sensitivity analysis to confirm that the assumption and the risk metrics for this ILRT application continue to be met by either (1) using an OR gate or (2) including the probability of human failure to choose the functioning instrument if the AND logic is retained.

f. For FPRA F&Os 4-12, 4-21, 9-2, 2-38, 2-39, and 2-50 listed in Table 51 of the LAR the licensee states "the evaluation used the FPRA model that will represent BFN at the time this ILRT application is applied. Therefore, the HFEs that will be in place will no longer be a strategy employed by BFN for fire hazards." Provide clarification confirming that the HFEs that have been proposed to be modeled in the FPRA will be representative of the strategy employed (i.e., as built, as operated) upon completion of all NFPA 805 milestones.

g. F&O 6-10 associated with SR IE-C8 identified that common cause failure (CCF) for battery chargers is not included in the initiating event fault tree for loss of two direct current buses, other than for the standby chargers. The licensee's resolution to this F&O addresses the exclusion of the CCF for battery chargers beyond the 24-hour mission time and states that the data used for modeling the individual buses is so conservative that it would be overly conservative to include the CCF. Table 2-2.4-3(b), "Supporting Requirements for HLR-SY-B," of the ASME/ANS RA-Sa-2009 as it pertains to systems analysis states, in part, that "MODEL intra-system common-cause failures when supported by generic or plant-specific data (an acceptable model is the screening approach of NUREG/CR5485, which is consistent with DA-D5) or SHOW that they do not impact the results."

Provide a quantitative basis for the conclusion that modeling the CCF battery chargers would be over-conservative.

APLA RAI-04

Throughout the LAR inconsistencies in values were identified by the NRC staff across tables, and some editorial context could not be understood or validated. Please address (correct) each of the below excerpts from the LAR and confirm that there was no impact on the application as a result of any changes made.

a. Sections 7.1.1, 7.1.2, and 7.1.4 in Table 4, of the LAR do not correspond to the Equation/Section in the EPRI Report.

b. Table 8, for Collapsed Accident Progression Bin #7, the population dose factor is 0.488. This value does not correspond to the calculated value for Browns Ferry 50-Mile Dose of 3.81E+06.

c. Table 12 identifies class 3b as the population dose-rate increase due to extending the ILRT interval, and is inconsistent with the description in Section 8.3.2, Unit-1 Population Dose-Rate Calculations.

d. Table 31 identifies the external events contribution for Fire CDF to be 3.70E-06, 5.40E-06, 5.40 E-06 and LERF to be 6.73E-07, 1.04E-06, and 1.01E-06 for Units 1, 2, and 3, respectively. The table also identifies the Seismic CDF to be 5.03E-05, 5.64E-05, 5.92E-05 and LERF 5.47E-06, 5.37E-06, and 5.02E-06 for Units 1, 2, and 3, respectively. This is inconsistent with the values provided in Section 10.1, Seismic Discussion.

e. In Table 32, confirm the LERF Increase value for Unit 3. This is inconsistent with Table 33.

f. In Table 33, the values of 4.60E-02 and 4.51E-02 for Units 2 and 3, respectively, are inconsistent with the values in Tables 18 and 22, which are 0.0417 and 0.0387, respectively.

g. Section A-1.0 states in part,

> A discussion of the TVA model update process, model history, peer reviews performed on the Browns Ferry models, the results of those peer

reviews and the potential impact of peer review findings on the containment ILRT extension risk assessment are provided in...

Provide a statement to complete the last sentence in the above paragraph.

h. With respect to Table 31, of the LAR, the staff acknowledges the discrepancy between the labeling of the seismic row for the fire values and vice-versa. Using those values for fire, the LERF fractions relative to CDF for Units 1, 2, and 3 are 0.109, 0.0952, and 0.0848, respectively. NRC staff determined that the values identified in Table 51 of the submittal for resolution of F&O 4-28 are inconsistent. Provide an explanation for the discrepancies in the values and confirm the values in Table 31 are the updated and correct values.

APLA RAI-05

For SR HR-G7 in Table 50 of the LAR, the licensee states that "the MOR [model of record] uses a minimum joint HEP threshold of 1E-07." During review of the BFN NFPA 805 LAR, the NRC staff requested that the licensee provide justification with respect to the establishment of acceptable minimum (or "floor") values for HEP combinations. In its response to NPFA 805 PRA RAI 01.v and PRA RAI 24 (ADAMS Accession No. ML14363A057), the licensee indicated that it updated the FPRA to apply a floor value of 1.0E-05 to all HEP combinations that do not include long-term decay heat removal HFEs for FPRA CDF or those HFEs that are cued and guided by severe accident mitigation guideline (SAMG) procedures for FPRA LERF. For the remaining combinations, the licensee stated that the FPRA applied a floor value of 1.0E-06, given that a low dependency exists between long-term decay heat removal and SAMG actions and other earlier actions. Further in response to PRA RAI-24, the licensee indicated that the revised floor values were incorporated in the integrated analysis. Specific to the NFPA 805 application, the NRC staff concluded that this issue was resolved because the FPRA included the use of floor values consistent with guidance contained in NUREG-1921. To support review of the ILRT LAR:

a. Confirm that the licensee's justification with respect to the establishment of acceptable minimum (or "floor") values for joint HEPs as described above for resolution in the FPRA for NFPA 805, remains the same and unchanged for the ILRT application. Subsequently, if the values used in the FPRA depart (are lower than) from the values provided in the NFPA 805 PRA RAI-01.v response, perform an updated sensitivity analysis that applies a floor value of 1.0E-05 to all joint HEP combinations and provide the results. Provide justification for any joint HEPs in which a value lower than the floor value (1.0E-05) has been determined to be acceptable for use.

b. For the IEs PRA, identify if any of the joint HEPs uses a value less than the floor value of 1.0E-06.

    1. If so, perform a sensitivity analysis for all the joint HFEs using a floor value of 1.0E-06 and provide the results, and

    2. Provide justification for joint HEPs in which a value lower than the floor value (1.0E-06) was determined to be acceptable for use.

APLA RAI-06 [Browns Ferry NFPA 805 License Condition, and Implementation Actions]

In Section 4.6.2, "PRA Technical Adequacy" of the LAR the licensee states in part, "The external events analyses include the current FPRA model which represents the plant after all modifications are implemented in support of transition to NFPA 805. This work is scheduled for completion in 2019. The FPRA model will represent the as-built, as-operated plant in the time period for the proposed extended containment ILRT interval." Provide the following information to address and confirm that the results of the licensee risk evaluation performed to support the requested extension for the ILRT in the LAR, will continue to meet RG 1.174 risk metrics, and the specific risk metrics for acceptance of an ILRT extension outlined in the NRC final safety evaluation for NEI 94-01 (ADAMS Accession No. ML081140105), after the scheduled work is due to be completed and prior to the next scheduled ILRTs on October 10, 2020, March 2021, and March 2020 for Units 1, 2, and 3, respectively.

a. Provide the status of all implementation listed in Table S-3, Implementation Items," of TVA letter, dated September 8, 2015 (ADAMS Accession No. ML15251A598), for transition to full compliance with 10 CFR 50.48(c) and delineated as Transition License Condition (3) in the staff safety evaluation issued for approval of BFN transition to a risk-informed performance-based fire protection program in accordance with 10 CFR 50.48(c) (ADAMS Accession No. ML15212A796).

b. For modifications that are listed in Table S-2 of TVA letter, dated September 8, 2015, as delineated in Transition License Condition (2) of the above safety evaluation, and credited in the as-built as-operated FPRA model:

   EITHER

   1. Perform a sensitivity analysis that excludes the incomplete modification(s) to assess if the acceptance criteria for the ILRT risk metrics continue to be met. This sensitivity analysis should also incorporate the NRC-approved fire methods from APLA RAI-02 and provide updated tables that include total LERF, ΔLERF, PDR, and CCFP for each unit to assess the risk impact.

   OR

   2. Propose a license condition requiring that for the requested ILRT extensions, all modifications that are credited in the as-built as-operated FPRA model listed in Tables S-2 of the TVA letters dated September 8, 2015 and October 20, 2015 will be completed immediately following the first outage of occurrence, for the next scheduled ILRT for each respective unit.

TENNESSEE VALLEY AUTHORITY PROPOSED RESPONSE DATES

Tennessee Valley Authority (TVA) proposed by an email dated December 27, 2017, the following schedule for responding to the Nuclear Regulatory Commission (NRC) staff's draft request for additional information (RAI), which were emailed to TVA on December 11 and 12, 2017, related the license amendment request to extend containment integrated and local leakage rate testing intervals at Browns Ferry Units 1, 2, and 3. TVA stated that additional time is required for APLA RAI-3b, APLA RAI-3e, APLA RAI-5, and APLA RAI-2 due to significant additional analysis.

Item	Final Response Date
SCPB[1] RAI-1	2/5/18
SCPB RAI-2	2/5/18
SCPB RAI-3	2/5/18
SCPB RAI-4	2/5/18
APLA[2] RAI-1	2/5/18
APLA RAI-3a	2/5/18
APLA RAI-3c	2/5/18
APLA RAI-3d	2/5/18
APLA RAI-3f	2/5/18
APLA RAI-3g	2/5/18
APLA RAI-4	2/5/18
APLA RAI-6	2/5/18
APLA RAI-3b	3/27/18
APLA RAI-3e	3/27/18
APLA RAI-5	3/27/18
APLA RAI-2	3/27/18

[1] RAIs by NRC staff from Containment and Plant Systems Branch (SCPB).
[2] RAIs by NRC staff from Probability Risk Assessment Licensing Branch A (APLA).

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 5.5.12 "PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NOS. MG0113, MG0114, AND MG0115; EPID L-2017-LLA-0292) DATED JANUARY 25, 2018

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ADAMS Accession No. ML18010B055 * via email ** via memo

OFFICE	DORL/LPLII-2/PM	DORL/LPLII-2/LA	DSS/SCPB/BC*
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DATE	01/23/18	01/23/18	01/22/18
OFFICE	DRA/APLA/BC**	DORL/LPLII-2/BC	DORL/LPLII-2/PM
NAME	SRosenberg	UShoop (PBuckberg for)	FSaba
DATE	12/01/18	01/24/18	01/25/18

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