Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-18-011

February 5, 2018

10 CFR 50.4
10 CFR 50 90

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, 50-296, and 72-052

Subject: Tennessee Valley Authority Response to NRC Request for Additional Information related to BFN Application to Revise Technical Specification 5.5.12 "Primary Containment Leakage Rate Testing Program" (BFN-TS-497)

References:

1. Letter from TVA to NRC, CNL-17-056, "Application to Revise Technical Specification 5.5.12 "Primary Containment Leakage Rate Testing Program" (BFN-TS-497),'" dated August 15, 2017 (ML 17228A490)

2. Letter from NRC to TVA, "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 (ML18010B055)

By letter dated August 15, 2017 (Reference 1), Tennessee Valley Authority (TVA) submitted a License Amendment Request (LAR) to the Nuclear Regulatory Commission (NRC) to revise Browns Ferry Nuclear Plant Units 1, 2, and 3 (BFN) Technical Specification 5.5.12 "Primary Containment Leakage Rate Testing [ILRT] Program," to adopt Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix J," as the implementation document for the performance-based Option B of 10 CFR Part 50, Appendix J.

In Reference 2, the NRC transmitted a Request for Additional Information (RAI) related to the TVA LAR. As described In the reference 2 letter, TVA will provide responses to selected RAIs by February 5, 2018. Per email on February 5, 2018, the NRC granted permission to defer the response to APLA RAI 3a to March 27, 2018. The remaining RAI responses will be provided by March 27, 2018. Enclosure 1 to this letter contains TVA's response to the RAIs due by February 5, 2018. The errors identified in the Reference 2 letter have been documented in TVA's Corrective Action Program.

Consistent with the standards set forth in 10 CFR 50.92(c), TVA has determined that the additional information, as provided in this letter, does not affect the no significant hazards determination associated with the request provided in Reference 1.

There are no new regulatory commitments contained in this submittal. If you have any questions concerning this submittal, please contact Ed Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 5th day of February 2018.

Sincerely,

Joseph W. Shea
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosure:

TVA Response to NRC Request for Additional Information

cc (Enclosures):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
NRC Project Manager - Browns Ferry Nuclear Plant
State Health Officer, Alabama State Department of Public Health

ENCLOSURE 1

TVA Response to NRC Request for Additional Information

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 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.

<u>SCPB RAI-1</u>

All historical ILRT leakage rate values in the table of 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 Safety Evaluation (SE) associated with NEI 94-01, Revision 2-A (ADAMS Accession No. ML100620847) established the requirements for extending the ILRT interval beyond ten (10) years as will be implemented with staff approval of the subject 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 La [the maximum allowable Type A test leakage rate at Pa, where Pa 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 ANSI/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 0.96Pac [calculated peak accident containment internal pressure, also defined as Pa above] nor exceed Pd [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 ANS 56.8-1994. Also explain how these test pressure values relate to the BFN revised Pa values associated with the Amendment Nos. 299, 323, and 283 (ADAMS Accession No. ML17032A120) pertaining to the Extended Power Uprate LAR.

<u>TVA Response</u>

This table provides the results of the two most recent Type A test performed on Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3.

Type A Test Results						
	Test Date	**P_a**	**$0.96P_a$**	**P_t**	**P_d**	**Acceptable**
BFN Unit 1	3/8/2007	48.5	46.56	49.82	56	Yes
	11/18/2010	48.5	46.56	51.16	56	Yes
BFN Unit 2	11/06/1994	49.6	47.62	50.28	56	Yes
	5/27/2009	50.6	48.58	51.05	56	Yes
BFN Unit 3	10/10/1998	50.6	48.58	51.2	56	Yes
	05/09/2012	50.6	48.58	50.83	56	Yes

P_d - Containment Design Pressure (psig)
P_a - Calculated Peak Accident Containment Internal Pressure (psig)
P_t - Final Test Pressure (psig)
The Acceptance Criteria is $0.96P_a<P_t<P_d$)

Extended Power Uprate (EPU) License Amendments 299, 323, and 283 (ML17032A120) for BFN Units 1, 2, and 3, respectively, revised the calculated peak accident containment internal pressure (P_a) to 49.1 psig. The table below evaluates the above results using the revised EPU P_a value.

Type A Test Results using EPU P_a						
	Test Date	**P_a**	**$0.96P_a$**	**P_t**	**P_d**	**Acceptable**
BFN Unit 1	3/8/2007	49.1	47.14	49.82	56	Yes
	11/18/2010	49.1	47.14	51.16	56	Yes
BFN Unit 2	11/06/1994	49.1	47.14	50.28	56	Yes
	5/27/2009	49.1	47.14	51.05	56	Yes
BFN Unit 3	10/10/1998	49.1	47.14	51.2	56	Yes
	05/09/2012	49.1	47.14	50.83	56	Yes

P_d - Containment Design Pressure (psig)
P_a - Calculated Peak Accident Containment Internal Pressure (psig)
P_t - Final Test Pressure (psig)
The Acceptance Criteria is $0.96P_a<P_t<P_d$)

As indicated above, the last two Type A test performed on each unit met the ANSI/ANS 56.8-1994 acceptance criteria of not less than 0.96 P_a and less than P_d for both current and post EPU P_a values.

<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. Both state 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 penetrations test results Also, identify and provide the details of all corrective actions performed to prevent reoccurrence.

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>TVA Response</u>

a. There have been two repeat failures in BFN Unit 1 testing since the last ILRT in 2010.

   1-FCV-1-56 has an Administrative limit of 6 standard cubic feet per hour (scfh). The refueling cycle (RF) 9 as-found test result was 44.696 scfh. As a result of the RF9 failure, the valve seat was re-furbished. The RF9 as-left PMT test result was 0.197 scfh. The RF10 as-found test result was 8.80 scfh. The cause of the RF10 failure was seat wear in a specific disc location that was alleviated by limit switch adjustment to change the seating of the double disc into a new seat location. Following limit switch adjustment, the as-left PMT test result was 0.0 scfh.

   1-SHV-71-0014 and 1-CKV-71-0580 are tested simultaneously with an administrative limit of 30 scfh. The as-found RF10 test result was 124.07 scfh. Internal inspection of 1-SHV-71-0014 revealed a scored seat, and the valve seat was replaced. The RF10 as-left test result of 29.601 scfh. Additionally, because there was very little margin in the RF10 as-left result, a contingency work order was prepared for RF11. The RF11 as-found result was 81.361 scfh. Following the RF11 failure, 1-SHV-71-0014 was re-furbished and the seat was replaced. The RF11 as-left result was 9.885 scfh. A review of test results for the same components of the same design in BFN Units 2 and 3 have not shown similar results. The upcoming RF12 test result in the fall of 2018 will determine if the action to repair the seat after the previous failure is sufficient to prevent re-occurrence.

b. Based on a review of all BFN Unit 2 administrative limit failures since the last ILRT in 2009, there have been no repetitive as-found failures (i.e., failed consecutive tests).

c. There have been two repeat failures in BFN Unit 3 testing since the last ILRT in 2012.

   3-FCV-77-2B has an Administrative limit of 6 scfh. The RF16 as-found test result was 10.85 scfh. The valve was disassembled, cleaned and inspected. The RF16 as-left result was 1.91 scfh. The RF17 as-found result was 6.69 scfh. Troubleshooting and evaluation of the RF17 failure determined the cause to be boundary valve leakage. The boundary leakage was quantified and as-found leak rate adjusted to 2.549 scfh, which is below the administrative limit. Maintenance is scheduled in RF18 to repair leaking boundary valve 3-SHV-77-601 to prevent re-occurrence.

   3-FCV-64-31/34/139/140 and 3-FCV-84-20 are tested simultaneously with an administrative limit of 6 scfh. As-found RF16 test result was 13.34 scfh. Troubleshooting performed during RF16 determined the major leakage to be from 3-FCV-64-140. 3-FCV-64-140 was refurbished during RF16. The RF16 as-left test result was 2.53 scfh. The RF17 as-found test result was 10.72 scfh. Troubleshooting performed during RF17 determined the major leakage to be from 3-FCV-64-139. Air Operated Valve (AOV) diagnostics were performed and backstop adjustments were made. The RF17 as-left test result was 2.62 scfh. Although this was a repeat failure of a composite group of valves tested, the failures were do to different valves with different failure mechanisms. No single valve failure was repeated. Because there was no repeat failure at the component level, no cause analysis with preventative recurrence controls were performed for this configuration.

SCPB RAI-3

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 and RCIC 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.*

TVA Response

a. BFN only utilizes the 120 month interval allowance for the 186 Mechanical Bellows and Electrical Penetrations. 100% of these components are on 120 month intervals. For all other Type B components, BFN utilizes more conservative extended intervals (i.e., 60 or 75 months).

   The following table provides a summary for all Type B components.

Type B Components	
Total Number of Type B Components	393
Not Eligible	3(1)
Insufficient Data	15(2)
Total Number of components eligible for extended test interval (i.e., 60, 75, 120 months)	375
Recent Performance Data does not support extended test interval	4
Total Number of eligible components on extended test interval	371
Percentage of eligible components on extended test interval	98.9%
Notes: (1) 30 month test interval is required by NEI 94-01. (2) Due to recent modifications or maintenance, these components have not completed two in-service test intervals.	

b. The following table provides a summary of all Type C components

Type C Components	
Total Number of Type C Components	291
Not Eligible	81(1)
Tested every outage	39(2)(3)(4)
Total Number of components eligible for extended test interval	171
Recent Performance Data does not support extended test interval	20
Total Number of eligible components that meet extended test interval criteria	151
Percentage of eligible components that meet extended test interval criteria	88.3%
Notes: (1) 30 month test interval is required by RG 1.163. (2) 15 Components tested every refuel cycle to satisfy Inservice Testing Program Pressure Isolation Valve Leakage Criteria. (3) Nine Components tested every refuel cycle due to only having a single isolation. (4) 15 Components tested at 30 months to satisfy Inservice Remote Position Indication Verification.	

c. NEI 94-01,Revision 3-A, states that Past Component Performance is one of the factors to be considered when establishing test intervals for Type B and C components. As indicated above, 98.3% of Type B components and 88.3% of Type C components meet the extended test interval performance criteria. TVA's judgement is that these percentages provide reasonable confidence that the historical performance of BFN containment isolation components is acceptable.

A majority of containment isolation components that are subject to performance based Type B or Type C tests are on or could be on an extended test interval, which demonstrates good performance of Type B and Type C penetrations. This also demonstrates that the current maintenance strategy on these components is sufficient to maintain long term component isolation health.

SCPB RAI-4

The 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 past major modification or maintenance repair of the primary since the last ILRT has been appropriately accompanied by a either a structural integrity test (SIT) or ILRT and that any plans for future major modification also includes 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 ILRTs 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.

<u>TVA Response</u>

As described in Section 3.1.4 of the NRC Safety Evaluation for NEI 94-01, Revision 2, the NRC staff considers a major modification as any modification, that would require the performance of an ILRT or NRC approved ILRT relief. For example, the cutting of a large hole in the containment for replacement of steam generators or reactor vessel heads, replacement of large penetrations would be considered a major modification. TVA is using this definition of a major modification to respond to this RAI.

There have been no major containment structure modifications made to the BFN Unit 1 containment since last ILRT performed on 11/8/2010.

There have been no major containment structure modifications made to the BFN Unit 2 containment since last ILRT performed on 5/27/2009.

There have been no major containment structure modifications made to the BFN Unit 3 containment since last ILRT performed on 5/9/2012.

The U.S. Nuclear Regulatory Commission (NRC) staff from 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 license amendment request (LAR) dated August 15, 2017 (ML 17228A490) 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 (ML12221A202) and the EPRI Report 1018243,"Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 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 (>2 inches [5.1 cm] in diameter) containment isolation valves.*

   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 (EE) Large Early Release Frequency (LERF), Sections 10.2 and 10.3 of the LAR applied the internal events LERF to CDF ratio. Since external events can subject plants to common-cause failures of multiple SSCs, it is possible for the LERF to CDF ratio to be higher than for internal events. Provide justification for assuming that the internal events CDF to LERF ratio also applies to the seismic CDF to LERF ratio. In addition, explain the basis for using the internal events LERF to CDF ratio to the external events, specifically for seismic and high winds.

TVA Response

a. Table 3 (EPRI Release Classes) of Enclosure 2 of the License Amendment Request (LAR) dated August 15, 2017, contains an editorial error in the description and the interpretation of EPRI Class 2. The description of Class 2 should have read "Large Containment Isolation Failures." The interpretation of Class 2 for assigning the BFN release category should read "failure to close of large (>2" diameter) containment isolation valves."

   Table 1 of Enclosure 2 of the August 15, 2017 LAR describes EPRI Class 2 correctly and is consistent with EPRI 1018243 Table 4-1. TVA confirmed that the data from Section 8.1.2 of Enclosure 2 of the August 15, 2017 LAR "Class 2 - Large Containment Isolation Failures" is consistent with EPRI 1018243 Section 4.3 "EPRI Accident Class Descriptions." This error is strictly editorial and has no affect on the analysis.

b. TVA concurs that common cause failures could cause the LERF / CDF ratio to be greater for seismic risk than for internal events (IE) risk. To resolve this issue, TVA used the results from an industry peer with two BWR-4 units similar to BFN, but having a seismic PRA model, to determine the effect of increasing the LERF / CDF ratio. The data for BFN and the reference plant are contained in the following table:

Row		Ref-1	Ref-2	BFN-1	BFN-2	BFN-3
1	LERF / CDF Ratio - Internal Events	0.145	0.132	0.182	0.192	0.188
2	LERF / CDF Ratio - Seismic Events	0.204	0.216	0.298	0.314	0.308
3	Seismic LERF % Increase	40.6%	63.6%	63.6%	63.6%	63.6%
4	Seismic LERF			**1.10E-06**	**1.70E-06**	**1.66E-06**

   As shown in the above table, the seismic PRA LERF / CDF ratio for the reference plant is 40.6% greater than the Internal Events (IE) ratio for one unit, and 63.6% greater for the other unit. Conservatively, the larger of the two values is used for the BFN analysis. Accordingly, the BFN IE LERF / CDF ratios (Row 1, taken from equation 34 of Enclosure 2 of the August 15, 2017 LAR) are increased by 63.6% resulting in the calculated seismic ratio (Row 2). The seismic ratio is then multiplied by the seismic CDF values from Table 31 of the submittal to estimates the LERF contribution for seismic events (Row 4). These values are used in the repopulation of Table 31, below.

   With respect to high winds, the conservative LERF ratios from the seismic calculation (Row 2 above) are applied against the high winds CDF. This treatment is conservative because the high winds initiator would not subject the plant on the same order of magnitude as seismic events in terms of common cause failures. In Enclosure 2 of the

August 15, 2017 LAR, TVA explained that the High Winds CDF was conservatively given a 1.0E-06/yr frequency; therefore, resulting in a LERF contribution of 2.98E-07/yr, 3.14E-07/yr, and 3.08E-07/yr for Units 1, 2 and 3, respectively.

Table 31 from Enclosure 2 of the August 15, 2017 LAR is repopulated with the new LERF values as indicated below:

External Hazard	CDF/yr			LERF/yr		
	Unit 1	Unit 2	Unit 3	Unit 1	Unit 2	Unit 3
Seismic	3.70E-06	5.40E-06	5.40E-06	1.10E-06	1.70E-06	1.66E-06
Fire	5.03E-05	5.64E-05	5.92E-05	5.47E-06	5.37E-06	5.02E-06
High Winds	1.00E-06	1.00E-06	1.00E-06	2.98E-07	3.14E-07	3.08E-07
Other Hazards	Screened	Screened	Screened	Screened	Screened	Screened
External Events - Total	5.50E-05	6.28E-05	6.56E-05	6.87E-06	7.38E-06	6.99E-06
Internal Events	6.93E-06	6.29E-06	7.72E-06	1.26E-06	1.21E-06	1.45E-06
External + Internal Events	6.19E-05	6.91E-06	7.33E-05	8.13E-06	8.59E-06	8.44E-06

As can be seen from the change in the total LERF values, all three BFN units remain below the acceptance criteria of Total LERF less than 1.0E-05/yr.

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

Section 2.5.5, "Comparisons with Acceptance Guidelines," of 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 BFN LAR outlines Fire (Focused Scope) and Internal Events 2009 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 fire PRA methodology since review of BFN FPRA NFPA 805 staff review and issuance of the SE 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 (VEWFDS) 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 Nos. ML16343A058). The guidance provided in FAQ 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 (ML 17228A490), the calculated total LERF that includes EE is close to the RG 1.174 risk acceptance criteria. However, the integration of NRC-accepted fire PRA 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 fire PRA methods and studies would not change the conclusions of the LAR. As part of this justification, identify the fire PRA methodologies used in the fire PRA 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 CCFP). Provide updated tables that include the increase in total LERF (IE and EE), total LERF (IE and EE), ΔLERF, PDR, and CCFP for each unit to assess the risk impact as appropriate.

TVA Response

TVA will provide the response to this request by March 27, 2018.

APLA RAI 03 [Internal Events and Fire PRA: FACTS AND OBSERVATIONS]

Tables 50 and 51, Internal Events PRA F&O Resolution, and Fire PRA F&O Resolution of the LAR, provide resolution of the peer review facts and observations (F&Os) and their impacts on the application for the internal events and fire PRAs. 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).

TVA Response

TVA will provide the response to this request by March 27, 2018.

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. The SR HR-G2 states to use an approach to estimation of 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. In addition,*
2. *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.*

TVA Response

TVA will provide the response to this request by March 27, 2018.

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 Internal Events. For any identified scenarios provide justification why the screening value is appropriate and confirm that the risk metrics for the application are still met.

TVA Response

The basic event IF-CB593-DOOR uses an assumed probability of 0.1 for fails to open. The reason for this basic event is to identify any potential flood scenarios that may result if this non-watertight door failing to release internal flood waters. The average test and maintenance model (BFN PRA CAFTA Model Rev. 7) was used to estimate risk of failing Door 593. This is the same model revision used for the BFN CILRT submittal.

The model was re-solved with the probability of this basic event set to 1.0 and the change in CDF and LERF was calculated. The percent change increase ranged from 0.03% to 0.17% for CDF, which is in the low E-08/yr range, and and 0.15% to 0.77% for LERF which is in the low E-08/yr range.

The change in CDF and LERF is very small and non-risk significant. The negligible increase in CDF and LERF indicates that no significant scenarios result in applying the screening factor. Therefore; the screening value of 0.1 is reasonable and appropriate for identifying potential internal flood scenarios for this plant configuration. The impact on the CILRT results are insignificant.

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 which 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 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 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 RCW initiating event frequency, and provide justification to support why the factor used is appropriate.

TVA Response

TVA concurs that the response to the 2009 F&O 6-8 used an incorrectly calculated factor and for combinations where it is potentially not valid. The 2015 peer view team noted this error while reviewing the BFN F&O responses. Subsequent to the 2015 peer review, TVA removed the basic event (SR IE-C8) from the Rev. 7 Model of Record (MOR), which was used in the Containment ILRT LAR. The Containment ILRT LAR provided the original 2009 disposition of F&O 6-8 in error, and did not describe the change to the model of record for this issue.

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.

TVA Response

TVA will provide the response to this request by March 27, 2018.

f. *For fire PRA 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 that "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.*

TVA Response

The FPRA credited actions have been developed to the extent possible to make these HRAs represent those proposed actions. The final fire procedures are not available to complete and verify the fire HRAs. Therefore, the FPRA model assumes that these actions will be in the final procedures as currently proposed. Before the FPRA recovery actions can be considered complete, they will have to be reevaluated when the fire procedures are approved and ready to be implemented in the post 805 transition. This License Condition is listed as Implementation Item 33 in the current version of Table S-3, Implementation Items" which was provided in Enclosure 1 of TVA letter to NRC, CNL-17-130, "Update to Response to NRC Request for Additional Information for License Amendment Request to Revise Modifications and an Implementation Item Related to 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) - Revision to Table S-3," dated October 23, 2017 (ML17297A039).

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 DC 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. Section 2-2.4, "Supporting Requirements for HLR-SY-B," of the ASME/ANS PRA 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.

TVA Response

An evaluation was performed whereby the intra-system CCF for the battery chargers was added to the fault tree logic and the model re-solved. The conclusion is that modeling of the battery charger CCF would not be over-conservative. While performing this evaluation, errors were identified in BFN PRA CAFTA Model Rev. 7 that needed to be resolved prior to performing the sensitivity. The probability of Battery Charger A under the support tree logic for Loss of both DC Buses initiator (BCHFR0CHGA2480000AIE) was calculated from an inadvertently assigned mission time of 24 hours. The probability for BCHFR0CHGA2480000AIE was therefore updated to show a one-year mission time instead of 24 hours (probability of failure increased from 1.22E-04 to 4.35E-02). Additionally, common cause variables were reviewed and updated throughout the fault tree to ensure the correct CCF probabilities were used.

The results of this evaluation indicates that the CDF increase (1E-07/yr range) was less than 4% for all three units, and the LERF increase (1E-06/yr range) was less than 10% for all three units. A review of the cutsets showed that the change in CDF and LERF was from the errors that were identified and corrected. The battery charger CCF had a negligible contribution to the increase. As such, the additional contribution to LERF was added to the total LERF values and all three units remained under the acceptance criteria threshold of 1.0E-05/yr.

APLA RAI 04

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

a. *In Table 4, 'Section 7.1.1', 'Section 7.1.2', and 'Section 7.1.4' does 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. *Table 32, confirm the LERF Increase value for Unit 3. This is inconsistent with Table 33.*

f. *Table 33, for Units 2, and 3 the values of 4.60E-02, and 4.51E-02 respectively, are inconsistent with the values from Tables 18, and 22, which lists 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 BFN ILRT submittal, 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 50 of the submittal for resolution of F&O 5-1 are inconsistent. Provide explanation for the discrepancies in the values and confirm the values in Table 31 are the updated and correct values.*

<u>TVA Response</u>

a. The Equation/Section referred to in Table 4 of Enclosure 2 of the LAR for the three cited examples are typographical errors. They should have been 8.1.1 (Class 1), 8.1.2 (Class 2) and 8.1.4 (Class 8). TVA has confirmed that there is no effect on the application because there is no change to the data provided by these references.

b. The Population Dose Factor for collapsed accident progression bin #7 should have been 1.950, rather than 0.488. The calculated value of 3.81E06 is correct for the BFN 50-mile Dose (Person-REM) as given in Table 8 for the 1.950 factor. This error was due to a transcription mistake while populating Table 8. As stated, the calculated dose given in the table is correct. Therefore, TVA has confirmed that there is no impact on the results.

c. As the title indicates, Table 12 represents the data for a population dose increase for Class 3a due to the extended CILRT intervals. The '3b' labels in the far left column of Table 12 should read '3a.' The data for Table 12 was verified to be that of Class 3a. Therefore, TVA has confirmed that there is no change to the data, and thus, no change to the results.

d. The title (external hazard) for the first two rows of Table 31 have been transposed. The first row should be 'seismic' and the second row 'fire.' The data for 'fire' is taken from the quantification of the Fire PRA model. The total external events contribution is determined by summing the contributors. Therefore, TVA has confirmed that there is no change to the data, and no change to the results.

e. The combined delta-LERF/yr for Unit 3 of 2.84E-07 given in Table 33 is correct. The value (3.84E-07) given in Table 32 is a typographical error. The values provided in Table 33 are those that are compared against the acceptance criteria for this metric. TVA has confirmed that there is no change to the Table 33 value and no change to the results

f. TVA agrees that the Table 33 values for BFN Units 2 and 3 for the Internal Events ΔPerson-REM/yr due to the proposed extension to the CILRT are inconsistent with those from Tables 18 and 22 for Units 2 and 3, respectively. The Table 33 values are populated directly from Tables 28 and 29 for Units 2 and 3, respectively.

   Two issues were determined to be at play for this data. The Unit 2 and 3 values for the delta population dose-rate included containment liner corrosion. To be consistent with the data for Unit 1, the Unit 2 and 3 values should have been those without corrosion. However, when considering the non-corrosion data (Unit 2 3.92E-02, Unit 3 4.11E-02), the values continue to be inconsistent with Tables 18 (4.17E-02) and 22 (3.87E-02), Units 2 and 3 Total Population Dose-Rate, respectively.

   For Unit 1, the population dose-rate values for each EPRI Class from Table 10 was used to propagate Table 27. For Units 2 and 3, the data was entered manually and truncated to the second decimal point. Table 33 receives its data as a direct input from Tables 27, 28 and 29 for Units 1, 2 and 3, respectively. Tables 28 and 29 were updated (i.e., a link to the data from Table 10) which removed the rounding, and the total dose changed to become consistent with Tables 18 and 22.

g. Section A-10 of enclosure 2 of the LAR should read:

   > 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 Table 50.

h. The closure statement for F&O 4-28 (Table 51) Fire PRA gave LERF to CDF ratios of 0.034, 0.029 and 0.035 for Units 1, 2 and 3, respectively. Those values were based on calculation NDN0009992012000012 Revision 0 "Fire Risk Quantification," which was issued on 12/5/14. The information from that calculation was used in the F&O response at a time that preceded the CILRT submittal. The CILRT analysis used Revision 4 of the Fire Risk Quantification, which was issued on 4/20/15. TVA has confirmed that the values in Table 31 are the correct values.

APLA RAI 05

For SR HR-G7 in Table 50 of the BFN ILRT submittal, the licensee states that "the MOR uses a minimum joint HEP threshold of 1E-07." During further review of the BFN NFPA-805 SE dated Oct 2015 (ML 15212A796) 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 (ML14363A057), and PRA RAI 24 (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 (DHR) 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 DHR and SAMG actions and other earlier actions. In its response to NFPA-805 PRA RAI 24 (ML14363A057), 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 is resolved because the FPRA includes the use of floor values consistent with guidance contained in NUREG-1921. To support review:

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.

TVA Response

TVA will provide the response to this request by March 27, 2018.

b. For the internal events 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.*

TVA Response

TVA will provide the response to this request by March 27, 2018.

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

In Section 4.6.2, "PRA Technical Adequacy" of the submittal the licensee states in part, "The external events analyses include the current Fire PRA (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 submittal dated, August 15, 2017, 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 (ML 081140105), 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 items for Units 1, 2, and 3 from Table S-3, Implementation Items," of Tennessee Valley Authority letter CNL-15-191, dated September 8, 2015 (ML 15251A598), to complete 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) (ML15212A796).*

TVA Response

The current version of Table S-3, Implementation Items" was provided in Enclosure 1 of TVA letter to NRC, CNL-17-130, "Update to Response to NRC Request for Additional Information for License Amendment Request to Revise Modifications and an Implementation Item Related to 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) - Revision to Table S-3," dated October 23, 2017 (ML17297A039). As stated in that letter, there are no additional updates or revisions to Table S-3.

b. *For modifications that are listed in Table S-2, 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 that excludes the incomplete modification(s) to assess if the acceptance criteria for the ILRT risk metrics continue to be met. This sensitivity should also incorporate the approved-NRC fire methods from RAI 02.a 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.*

<u>TVA Response</u>

TVA proposes the following license condition:

> *Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.*

Proposed mark-ups of the Operating License for BFN Units 1, 2, and 3 are provided in the attachment.

ENCLOSURE 1

TVA Response to NRC Request for Additional Information

Attachment

Proposed mark-ups of the Operating License for BFN Units 1, 2, and 3

Following Implementation:

(a) The first performance of SR 3.7.4.4, in accordance with TS 5.5.13.c.(i), shall be within a specific frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from November 10, 2003, the date of the most recent successful tracer gas test.

(b) The first performance of the periodic assessment of the Control Room Envelope (CRE) Habitability, Technical Specification 5.5.13.c.(ii), shall be within 9 months following the initial implementation of the TS Change. The next performance of the periodic assessment will be in a period specified by the CRE Program. That is 3 years from the last successful performance of the Technical Specification 5.5.13.c.(ii) tracer gas test.

(c) The first performance of the periodic measurement of CRE pressure, TS 5.5.13.d, shall be within 24 months, plus 180 days allowed by SR 3.0.2 as measured from the date of the most recent successful pressure measurement test.

(17) The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, Revision 0,” (i.e., TS 5.6.5.b.11) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined. This license condition will be effective upon the implementation of Amendment 285.

(18) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, “Plant Modifications,” of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than December 20, 2013, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage

Following implementation:

(a) The first performance of SR 3.7.4.4, in accordance with TS 5.5.13.c(i), shall be within a specific frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from November 10, 2003, the date of the most recent successful tracer gas test.

(b) The first performance of the periodic assessment of the Control Room Envelope (CRE) Habitability, Technical Specification 5.5.13.c(ii), shall be within 9 months following the initial implementation of the TS change. The next performance of the periodic assessment will be in a period specified by the CRE Program. That is 3 years from the last successful performance of the Technical Specification 5.5.13.c(ii) tracer gas test.

(c) The first performance of the periodic measurement of CRE pressure, TS 5.5.13.d, shall be with 24 months, plus 180 days allowed by SR 3.0.2 as measured from the date of the most recent successful pressure measurement test.

(17) The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, Revision 0," (i.e., TS 5.6.5.b.10) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined. This license condition will be effective upon the implementation of Amendment No. 311.

(18) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than June 28, 2014, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the

Following Implementation:

(a) The first performance of SR 3.7.4.4, in accordance with TS 5.5.13.c.(i), shall be within a specific frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from November 10, 2003, the date of the most recent successful tracer gas test.

(b) The first performance of the periodic assessment of the Control Room Envelope (CRE) Habitability, Technical Specification 5.5.13.c.(ii), shall be within 9 months following the initial implementation of the TS Change. The next performance of the periodic assessment will be in a period specified by the CRE Program. That is 3 years from the last successful performance of the Technical Specification 5.5.13.c.(ii) tracer gas test.

(c) The first performance of the periodic measurement of CRE pressure, TS 5.5.13.d, shall be within 24 months, plus 180 days allowed by SR 3.0.2 as measured from the date of the most recent successful pressure measurement test.

(d) For License Amendment 268, the licensee shall implement changes to BFN, Unit 3 TSs 5.6.5 and 3.3.1.1 within 60 days of approval. The remaining BFN, Unit 3, changes will be implemented upon completion of required supporting modification work and prior to entering Mode 3 (i.e., Hot Shutdown) from the spring 2014 refueling outage.

(13) The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, Revision 0," (i.e., TS 5.6.5.b.10) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined. This license condition will be effective upon the implementation of Amendment No. 270.

(14) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-17-024, dated June 7, 2017.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than July 2, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.