1101 Market Street, Chattanooga, Tennessee 37402

CNL-16-082

May 25, 2016

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, and 50-296

Subject: **Response to NRC Request for Additional Information Related to License Amendment Request for Adding New Specifications to Technical Specification 3.3.8.3 (BFN-TS-486) (CAC Nos. MF6738, MF6739, and MF6740) - Letter 4**

References:

1. Letter from TVA to NRC, CNL-15-073, "Application to Modify the Browns Ferry Nuclear Plant, Units 1, and 2 Technical Specifications by Adding New Specification TS 3.3.8.3, 'Emergency Core Cooling System Preferred Pump Logic, Common Accident Signal (CAS) Logic, and Unit Priority Re-Trip Logic,' and Unit 3 TS by adding New Specification TS 3.3.8.3, 'Common Accident Signal (CAS) Logic, and Unit Priority Re-Trip Logic,' (BFN-TS-486)," dated September 16, 2015 (ML15260B125)
2. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information Related to License Amendment Request for Adding New Specifications to Technical Specification 3.3.8.3 (CAC Nos. MF6738, MF6739, and MF6740)," dated March 21, 2016 (ML16074A126)
3. Letter from TVA to NRC, CNL-16-066, "Response to NRC Request for Additional Information Related to License Amendment Request for Adding New Specifications to Technical Specification 3.3.8.3 (BFN-TS-486) (CAC Nos. MF6738, MF6739, and MF6740) - Letter 1," dated April 15, 2016 (ML16106A323)

By letter dated September 16, 2015 (Reference 1), Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3, to revise the BFN, Units 1 and 2, Technical Specifications (TS) by adding a new specification governing the safety functions for the Emergency Core Cooling System (ECCS) Preferred Pump Logic, Common Accident Signal Logic, and the Unit Priority Re-Trip Logic. In addition, the LAR relocated the BFN, Unit 3 requirements for Common Accident Signal Logic and Unit Priority Re-trip Logic to a new specification governing the safety functions for the Common Accident Signal Logic, and the Unit Priority Re-Trip Logic for consistency with the changes to the BFN, Units 1 and 2 TS.

By letter dated March 21, 2016 (Reference 2), the Nuclear Regulatory Commission (NRC) requested additional information to support the review of the LAR. The required dates for responding to the requests for additional information varied from April 15, 2016, to May 25, 2016.

Enclosure 1 provides the fourth set of TVA responses to some of the requests for additional information (RAIs) identified in the Reference 2 letter. The due dates for the RAIs were revised from the Reference 2 letter and detailed in the Reference 3 letter. As stated in the Reference 3 letter, the responses provided in Enclosure 1 to this letter are due by May 25, 2016. Enclosure 2 provides a listing of the RAIs contained in the Reference 2 letter and the date of the TVA response to each of the RAIs.

Consistent with the standards set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.92(c), TVA has determined that the additional information, as provided in this letter, does not affect the no significant hazards consideration associated with the proposed application previously provided in Reference 1.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Mr. Edward D. Schrull at (423) 751 3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 25th day of May 2016.

Respectfully,

J. W. Shea

Digitally signed by J. W. Shea
DN: cn=J. W. Shea, o=Tennessee Valley Authority, ou=Nuclear Licensing, email=jwshea@tva.gov, c=US
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J. W. Shea
Vice President, Nuclear Licensing

Enclosures

cc: See page 3

Enclosures: 1. TVA Responses to NRC Request for Additional Information: Set 4
2. Summary of BFN Request for Additional Information Response Dates

Enclosure
cc (Enclosure):

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

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3

TVA Responses to NRC Request for Additional Information: Set 4

Probabilistic Risk Assessment (PRA) Licensing Branch (APLA) Request for Additional Information (RAI) 12

The License Amendment Request (LAR) states that at the date of the submittal, there were no outstanding plant changes that necessitate a change in the PRA, except for the modifications submitted as part of the transition to NFPA 805. Discuss whether the internal events PRA, fire PRA, and other PRA or external events risk assessments reflect the as-built, as-operated plant in that only completed modifications are credited. If non-completed modifications are credited, remove the credit and provide the updated results for the LAR as part of APLA RAI 14.

Tennessee Valley Authority (TVA) Response

Internal Events (IE) Probabilistic Risk Assessment (PRA)

Revision 6 of the Browns Ferry Nuclear Plant (BFN) IE PRA model was used to evaluate the risk-informed changes proposed in the Tennessee Valley Authority (TVA) LAR) dated September 16, 2015 (ML15260B125). The BFN IE PRA was updated to Revision 7 on March 31, 2016. A summary of the revision to the BFN IE PRA is provided in the TVA response to APLA RAI 11 provided in TVA letter dated April 29, 2016 (ML16123A071). Revision 7 was a minor revision to the model and represents the as-operated plant including all design changes up to the evaluation cutoff of September 2015. The changes in risk from Revision 6 to Revision 7 were dominated by changes in component data. None of the changes between Revision 6 and Revision 7 would change the results of the risk insights provided in the original submittal. The following table provides a comparison of the Core Damage Frequency (CDF) and Large Early Release Fraction (LERF) for the Revision 6 PRA model and the Revision 7 PRA model for each BFN unit.

BFN IE CDF and LERF

Unit	Revision 6		Revision 7	
	LERF	CDF	LERF	CDF
1	1.11E-06	6.05E-06	1.26E-06	6.93E-06
2	1.06E-06	5.25E-06	1.21E-06	6.29E-06
3	1.05E-06	5.88E-06	1.45E-06	7.72E-06

Per TVA procedure NEDP-26, "Probabilistic Risk Assessment (PRA)," plant modifications are evaluated for their effect on the IE PRA. Significant changes to the model of +/- 10% CDF or LERF require IE PRA update. Revised PRA IE models are only issued after the subject modifications that warrant a change have been installed and placed in service in the plant. There have been no modifications to BFN since the issuance of the Revision 6 model that the LAR was based on or the Revision 7 current approved model that met the revision threshold. Therefore, the risk insights provided in the LAR represents risk insights from the as-operated, as-designed plant. However, any quantification results presented in subsequent RAIs (i.e., APLA RAI 14) will be performed with Revision 7 of the IE PRA.

Fire PRA

The BFN Fire PRA submitted in support of the BFN Transition to National Fire Protection Association Standard (NFPA) 805 by TVA letter dated March 27, 2013 (ML13092A393), included several modifications that have not been installed in the plant to date. The following table identifies the NFPA 805-related modifications that are credited in the Fire PRA and the installation status of the modifications.

NFPA -805 Modifications Credited in the Fire PRA

Modification Title	**Modification Description**	**Status**
Change the normal position of the 4Kv Shutdown Board Manual/AUTO transfer switches from AUTO to MANUAL.	4KV shutdown boards and busses have auto/manual mode switches (43-x) that automatically trip breaker controls to manual. Electrical power is required to trip the switch to manual. In some fire areas, the trip to manual power supply is lost and the breakers become 'stuck' in automatic, making associated boards unavailable. This modification is partially analyzed in FRANX and partially analyzed with CAFTA logic changes.	Partially completed
Install an emergency high pressure makeup pump for each unit.	This new pump would be credited in the PRA for core cooling and for Decay Heat Removal (DHR) when used in conjunction with Hardened Wetwell Vent (HWWV).	Not completed
Install switches in the Main Control Room (MCR) to bypass the shutdown cooling logic interlock for the Residual Heat Removal (RHR) injection valves.	RHR and SBCS are unavailable for vessel injection in many fire scenarios because fire damage to field cables going to FCV 74-47 and 48 limit switches could energize relay 10AK63A or B, placing a CLOSE signal on the Low Pressure Coolant Injection (LPCI) injection valves.	Completed
Change power supply arrangement for Drywell wide range Pressure instruments 1,2,3 PI-64-160A and B and Suppression Pool wide range level LI-64-159A and B to reduce exposure to fires in the reactor buildings.	The current power supply arrangement for these loops is shared with secondary containment damper controls, which have cables routed throughout the plant. This instrumentation is needed to support containment venting through the HWWV.	Partially completed
Move the Common Accident Signal (CAS) inhibit switches from Auxiliary Instrument Room to the MCR	Spurious CAS signal causes unnecessary diesel generator (DG) breaker trips	Partially completed
Provide Drywell wide range Pressure instruments 1,2,3 P-64-160A or B and Suppression Pool wide range level L-64-159A or B on the Backup Control Panel for use in Control Room Abandonment	For fires in FA 16 requiring MCR abandonment, this instrumentation is needed outside of the MCR to support containment venting through the HWWV.	Not completed
Install separate emergency control power fuses in 480V Shutdown Board NORM and ALT feeder breakers.	These breakers are equipped with "43" switches to isolate the control circuit from damage in the control bay, but they do not have emergency fuses as required to be useable after initial fire damage.	Completed
Modify protective relay logic for all eight DGs to eliminate the lockout feature from the 51V relay and to only trip the DG output breaker. Provide a pull-to-lock function for RHR pumps 1B, 2C, and 3A using the MCR hand switch.	Spurious start of RHR pumps has the potential to overload and damage the DGs and cause output breaker lockout. Pull to lock function would be used to prevent spurious starts due to fire damage in the Auxiliary instrument rooms and spreading rooms and the relays will not lockout the breaker.	Partially completed
Fire wrap normal control power cable 3B188-B3 for 4Kv Shutdown Board 3EC in FA 21	Fire damage to RHR Service Water (RHRSW) pump B1 cable 3ES4080-II on SHUTDOWN BOARD EC in FA 21 causes a fault protection issue when control power cable 3B188-B3 is also damaged and ability to trip breakers is lost. This Board is NOT credited in FA 21	Partially completed

Modification Title	Modification Description	Status
Reroute normal control power cable 3B193-B2 for 4Kv Shutdown Board 3ED in FA 21	Fire damage to RHRSW pump D1 cable 3ES4090-II on Shutdown Board ED in FA 21 causes a fault protection issue when control power cable 3B193-B2 is also damaged and ability to trip breakers is lost. This Board is NOT credited in FA 21	Partially completed
Upgrade the Unit 3 condensate booster pumps to larger capacity.	Hydraulic performance assumed for flooding the RPV using the condensate system is based on new condensate booster pumps being installed for EPU. Unit 1 and 2 are completed.	Completed
Install manual valves in the turbine building that allow the control air headers in the individual reactor buildings to be depressurized.	In fire scenarios, containment isolation is needed to reduce LERF in the PRA and Scram Discharge Instrument Volume (SDIV) Vent and drain isolation is needed to prevent core damage from Inter-System Loss of Coolant Accident (ISLOCA). In some fire scenarios outside of the turbine building (TB) fire areas, these functions are damaged by fire.	Removed from scope, not installed.
Install backup diesel generator capable of powering the emergency high pressure makeup pumps for each unit.	Install adequate on-site DG capacity to simultaneously run the three emergency high pressure makeup pumps (one per unit).	Not completed
Provide breaker coordination for 250V DC Turbine Building Distribution Boards.	This modification is needed in order to credit OSP for fire scenarios in the TB, which is needed to lower baseline CDF. The 250V TB Distribution Boards are required to support OSP. Review shows that the feeder breakers and load breakers are not coordinated.	Not completed
Change breaker sizes and settings to protect Balance of Plant (BOP) cables from auto ignition.	The population of circuits identified in Attachment 7 of Calculation EDQ0999870077, Revision 32, "BFN Analysis of the Auxiliary & Control Power System to Identify Associated Circuits - 10CFR50 Appendix R," as not having adequate protection from auto ignition will be further evaluated and those that could propagate a fire from the initial scenario to another location will be modified to protect cables from auto ignition.	Not completed
Reduce the overcurrent (TOC) settings for 4Kv Shutdown Bus feeder breakers to protect SD Bus from overload • Breaker 1126 • Breaker 1226 • Breaker 1132 • Breaker 1232	Spurious start of large loads on the 4Kv Shutdown Boards and overload and damage the 4Kv Shutdown Bus breakers. Setting the TOC relays on the Unit Board feeders to a lower setting will trip before permanent damage and allow use of OSP after the spurious loads are secured.	Not completed
Incipient detection in aux instrument rooms	Incipient Detection to cover all the electrical panels in each units Auxiliary Instrument Room	Completed
Install backup pneumatic actuation capability for the HWWV system which is purely mechanical and not subject to fire damage in the reactor building.	This capability is part of Fukushima modifications. It allows for actuation of FCV 64-221 and 222 from outside of the reactor building.	Not completed
Increase the capacity of the HWWV system to service 3 units at the same time.	This capability is part of Fukushima modifications. Current capability of the HWWV system can only service one Unit at a time, this modification increases the capacity of the HWWV system to service three units at the same time.	Not completed
Modify condensate system flow controls 1,2,3 FIC 2-29 to function automatically and 1,2,3 LCV 2-3 to prevent flow diversion for fires in the Auxiliary Instrument Rooms from the MCR.	Fires in the Unit 1 / 2 / 3 Aux Instrument Rooms can cause a loss of Condensate. The Condensate failure is being caused by a loss of power to FIC-2-29 and LIC-2-3. Condensate system flow controls 1, 2,3 FIC 2-29 and 1, 2,3 LIC 2-3 will be modified to function automatically for fires in the Auxiliary Instrument Rooms	Not completed
Abandon Battery charger 1-CHGD-039-0001	This battery charger is no longer used but is an ignition source for numerous significant target cables (fire scenario 1-3.007-BCHG)	Not completed

As shown in the table above, there are a significant number of modifications that have not been installed to date that are credited in the Fire PRA. However, as stated in the TVA letter dated March 27, 2013, TVA determined that a number of fire areas represented a higher contribution of risk to the BFN core damage frequency (CDF) as a result of a fire than other fire areas and should be subject to interim compensatory measures. These interim compensatory measures were implemented in addition to the currently existing BFN fire protection requirements and compensatory measures and apply until the modifications described in Attachment S, Table S-2, "Plant Modifications Committed," of the NFPA 805 LAR are installed. These interim compensatory measures provide added assurance that fire risk is mitigated until NFPA 805 LAR Table S-2 credited modifications are implemented at BFN. Removal of the modifications from the Fire PRA increases both CDF and LERF, given that the interim compensatory actions are not and cannot be easily credited in the Fire PRA. Note that the interim compensatory measures described in the TVA letter dated March 27, 2013, were revised by TVA letters dated May 16, 2013 (ML13141A291), November 22, 2013 (ML13333A169), August 14, 2014, (ML14231A961), August 26, 2014 (ML14239A325), January 5, 2016 (ML16006A124), and May 10, 2016.

TVA letter dated December 17, 2014 (ML14363A056), evaluated the risk effects of removing the uninstalled modifications from the Fire PRA in response to PRA RAI 19.a.01 associated with the NFPA-805 submittal. The TVA response to PRA RAI 19.a.01 included a table (also provided below) showing the fire risk decrease from the committed modifications listed above. Given the large decrease in CDF and LERF from the credited modifications, any actual risk insights related to the Emergency Core Cooling System (ECCS) Preferred Pump Logic (PPL) LAR will likely be masked by the large increase in the calculated CDF and LERF resulting from removing the non-completed committed modifications from the Fire PRA. Any Fire PRA quantification results in subsequent RAIs (i.e., APLA RAI 14 to be submitted in a subsequent TVA letter) without the non-completed modifications would be considered as information only and not a valid risk insight.

Fire Risk Decrease from Modifications

Unit	Decrease in CDF	Decrease in LERF
Unit 1	1.63E-04	2.48E-05
Unit 2	1.43E-04	2.13E-05
Unit 3	1.53E-04	2.15E-05

External Events

PRA insights for external events are based on the BFN Individual Plant Examination of External Events (IPEEE). Validity of using the IPEEE is discussed in the TVA response to APLA RAI 13.b and APLA RAI 13.c of this letter.

APLA-RAI-13

Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Section 2.3.3, states the following:

> *As a minimum, evaluations of CDF [core damage frequency] and LERF [large early release frequency] should be performed to support any risk-informed changes to TS. The scope of the analysis should include all hazard groups (i.e., internal events, internal flood, internal fires, seismic events, high winds, transportation events, and other external hazards) unless it can be shown that the contribution from specific hazard groups does not affect the decision.*

Section 4.4.3.5 of the LAR addresses risk from external events.

APLA RAI 13.b

The LAR states that the BFN Individual Plant Examination of External Events (IPEEE) did not calculate a Core Damage Frequency (CDF) or Large Early Release Fraction (LERF) due to high winds/tornadoes. The IPEEE analysis concluded that the CDF from high winds was judged to be less than 1 E-6 per year. Since the IPEEE studies are outdated, address the risk from high winds and tornadoes, considering current plant configuration and operation and updated hazard and risk insights. Discuss the changes made and include the updated results as part of APLA RAI 14.

TVA Response

The BFN IPEEE analysis of high winds, floods, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. A screening approach as described in Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," Supplement 4 was used for evaluation of high winds, external floods, and nearby facility/transportation events. No other external events (e.g., volcanic activity) are applicable to BFN. The screening approach used in the analysis of external floods, and nearby facilities/transportation accidents demonstrated that they meet NRC Standard Review Plan (SRP) 1975 criteria and have adequate defense against these threats. Because BFN did not meet the SRP 1975 criteria for high winds, a bounding analysis was performed. This analysis showed the contribution to core damage frequency due to high winds to be less than the IPEEE screening criteria of 1E-06.

In response to APLA RAI 13.b, TVA provides the following additional information for high winds and tornadoes. Evaluations for high winds/tornadoes are included to address any new insights may affect the ECCSPPL evaluations. It should be noted that this information is not intended nor considered to be an update to the IPEEE.

<u>Tornadoes</u>

The NUREG/CR-5042 approach suggests that if the tornado frequency or the local tornado strike frequency is determined to be on the same order of magnitude as the first figure-of-merit for core damage (i.e., CDF ≤1 E-5 per year), then the analysis could be stopped.

BFN Updated Final Safety Analysis Report (UFSAR) Section 2.3.6.2 states that the annual tornado occurrence in the 30 nautical miles radius circle is 1.81 per year and the return probability is 6.98 E-4 per year. Therefore, these scenarios can not be screened out based solely on the likelihood of the event. A bounding estimate is provided below because the local tornado strike frequency is greater than the first figure-of-merit for core damage.

A bounding estimate for ECCS PPL out of service (OOS) due to a tornado is provided below, based on the following scenario assumptions.

- The initiating event is a tornado strike resulting in loss of offsite power and loss of condenser.
- Only fast acting scenarios where either a Stuck Open Relief Valve (SORV) or Loss of Coolant Accident (LOCA) occurs after the tornado causes the plant to trip are considered.
- Both divisions of ECCS PPL are OOS for seven days per year
- A spurious or actual accident signal occurs in the other unit as a result of an SORV or low level signal.
- No credit is given to any operator action to realign buses or reload diesels (i.e., fast acting scenario). However, it should be noted that the time available to prevent core damage if low pressure injection (LPI) is restored would be between 12.9 and 34 minutes (see Note).
- LPI is unavailable if PPL is OOS, SORV or LOCA occurs shortly after the tornado, and an actual or spurious signal occurs in the other unit.
- Only CDF is estimated because there would be adequate time to realign buses prior to a large release.
- The spurious accident signal is due to common cause miscalibration of low level transmitters (5.3E-4) or a relay failing to change state (9.92E-5)
- The High Pressure Coolant Injection (HPCI) failure due to all causes is 0.1.
- The probability of all LOCA in the first four hours after a tornado is approximately 10% relative to the probability of SORV and thus is not included in the calculation.

Note: MAAP case ATWS12 (ATWS with two SORVs, recirculation pump trip, and Standby Liquid Control (SLC) success, and all high pressure injection (HPI) and LPI systems failed) yields 12.9 minutes to the onset of core damage. MAAP case CASE05 (transient with a single SORV and all HPI and LPI systems failed) yields 34 minutes to the onset of core damage. MAAP case CASE01 (transient with all HPI and LPI systems failed) yields 37 minutes to the onset of core damage.

Based on the inputs and assumptions listed above core damage is estimated as follows:

CDF(tor) = f(sl) * (PPL(oos)+PPL(fd)) *(SORV1*HPCI+SORV2) * (SORVy+Ps)

where,

f(sl) = Tornado strike likelihood from UFSAR = 6.98 E-4
PPL(oos) = ECCS PPL unavailability = 7/365 = 1.92 E-2
PPL(fd) = ECCS PPL logic failure on demand = 9.92 E-5
SORV1 = One SORV fails to reclose = 7.69 E-4
SORV2 = Two or more SORVs fail to reclose = 2.55 E-5
SORVy = One or more SORVs in accident unit = 7.69 E-4 + 2.55 E-5 = 7.95 E-4
Ps = Spurious signal in other unit = CCF miscalibration or relay failure =
5.3 E-4 + 9.92 E-5 = 6.29 E-4
HPCI = 0.1

CDF(tor) = 6.98 E-4*(1.92 E-2+9.92 E-5)*(7.69 E-4 *0.1+2.55 E-5)*(7.95 E-4 + (6.29 E-4))
CDF(tor) = 6.98 E-4 * 1.93 E-2 * 1.02 E-4 * 1.42 E-3
CDF(tor) = 1.96 E-12 per year

Based on the conservative estimate shown above, the contribution to CDF from tornadoes when ECCS PPL is OOS or failed is negligible. The contribution to LERF would be significantly smaller because there would be significant time to realign alternate power, and load the DGs. Therefore, no changes to the models or ECCS PPL evaluation have been performed for tornadoes.

<u>Hurricanes and High Winds:</u>

Improved understanding and enhanced models have indicated that for some sites, hurricane winds, which are often lower speed than design basis tornado winds, may produce more intense missiles than tornado winds. However, the impact of missiles from hurricanes is mitigated as noted below.

- BFN's location does not make it susceptible to the more energetic missiles potentially created by a hurricane in coastal sites. BFN is approximately 290 miles inland from the Gulf of Mexico
- A significant amount of time would be available to prepare for any high winds, storms and other effects associated with a hurricane or tropical storm

TVA determined that high straight winds and hurricanes do not affect scenarios involving the ECCS PPL. The primary effect of these events would be potential loss of condenser or loss of offsite power. Any potential effects due to loss of the condenser or loss of offsite power are already addressed by the appropriate initiators included in the model. As stated in the TVA responses to APLA RAI 2 and 3 provided in TVA letter dated April 15, 2016 (ML16106A323), failures or unavailability of ECCS PPL is only significant for fast acting scenarios combined with either a real or spurious accident signal in the other unit because there would be little time to realign power to boards or load the diesel in these fast acting scenarios. For hurricanes, there is ample time available and procedural direction is available to prepare for associated severe weather effects. Similarly, for the high straight wind events, adequate time and guidance is available to prepare for the effects of storms, based on severe storm forecast evaluations by plant personnel. Therefore, no changes to the models or ECCS PPL evaluation have been performed for high straight winds or hurricanes.

<u>APLA RAI 13.c</u>

Address the risk from other external hazards, such as external flooding, transportation, and nearby facility accidents. Since the IPEEE studies are outdated, address these other external hazards considering current plant configuration, operation, and updated hazard and risk insights. Discuss the changes made and include the updated results as part of APLA RAI 14.

<u>TVA Response</u>

TVA reviewed the BFN IPEEE submitted on July 24, 1995, to determine the contribution from external risk associated with external flooding and transportation and nearby facilities accidents. Section 2.3.2 of the IPEEE states:

> "A screening approach is used for evaluation of risk from high winds, external floods and nearby facility/transportation events. The flowchart shown in the GL Supplement (Figure 1 of Reference 2.3 [Generic Letter 88-20, Supplement 4, dated June 28, 1991]) is used as basic foundation of the screening approach. Basically, the method consists of reviewing the analyses previously completed in supporting of licensing, reviewing changes to plant environs since Operating License (OL) issuance and verifying that the plant design conforms with the 1975 SRP criteria."

In response to APLA RAI 13.c, TVA provides the following additional information for external floods. The latest evaluation for external floods was reviewed to determine whether any new insights may affect the ECCS PPL evaluations. It should be noted that this information is not intended or considered to be an update to the IPEEE.

External Flooding

By letter dated March 12, 2015 (ML15072A130), TVA submitted the Flooding Hazard Reevaluation Report (HRR) for BFN, Units 1, 2, and 3, in response to the NRC Letter, "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated March 12, 2012 (ML12053A340). The Flooding HRR describes the approach, methods and results from the reevaluation of flood hazards at BFN, Units 1, 2, and 3. The report describes the eight flood-causing mechanisms and a combined-effect flood identified in Attachment 1 to Enclosure 2 of the NRC letter dated March 12, 2012, along with the potential effects on BFN, Units 1, 2, and 3.

TVA has determined that the external flooding events discussed in the HRR do not affect scenarios involving the ECCS PPL. As stated in the TVA responses to APLA RAI 2 and 3 provided in TVA letter dated April 15, 2016 (ML16106A323), failure or unavailability of ECCS PPL is only significant for fast acting scenarios combined with either a real or spurious accident signal in the other unit because there would be little time to realign power to boards, or load the diesel in these fast acting scenarios. For the probable maximum flood (PMF) events, ample time is available and procedural direction is available to prepare. Similarly, for the Local Intense Precipitation (LIP) events, adequate time and guidance is available to prepare for the effects of storms, based on severe storm forecast evaluations by plant personnel. The risk of the analyzed external flooding events combined with ECCS PPL unavailability during a LOCA and a spurious accident signal is deemed insignificant.

The above conclusion is based on the results of the identified potential plant effects (Section 12 of the HRR) due to external floods, minimal effect to structures or equipment used to prevent or mitigate core damage and early release, and the low likelihood of a LOCA combined with a spurious signal (fast acting scenarios). These external events do not affect the likelihood of fast acting scenarios involving the ECCS PPL. Any potential effects due to loss of the condenser or loss of offsite power are already addressed by the appropriate initiators included in the model. Therefore, no changes to the models or ECCS PPL evaluation have been performed for external floods.

The information provided in the HRR related to the potential flooding effects at BFN is briefly summarized as follows.

The Flooding HRR shows that some flood levels exceed the Current Licensing Basis (CLB) levels. The increased levels are the results of newer methodologies and guidance which typically exceed the methodologies and guidance that were used to establish the CLB for existing plants.

An Integrated Assessment (IA) is required within two years of submitting the Flooding HRR if flood levels determined during the flood hazard reevaluation are not bounded by the CLB flood levels. TVA committed to complete an IA and submit the report no later than March 12, 2017.

The HRR identified three flood causing mechanisms that are not bounded by the CLB for BFN. These three mechanisms and a brief summary of the potential plant effects are as follows.

1. Local Intense Precipitation (LIP) - The re-evaluation of the LIP event determined that the flood water would potentially exceed the plant critical elevation for over an hour by as much as 1.6 feet. Because the reevaluated LIP flood hazard is not bounded by the CLB, an IA will be performed as previously discussed where the effect of the exceeded flood hazard on the plant's safety related structures, systems, and components will be examined in detail. To evaluate the potential plant effects due to the potential LIP flood elevation, access doors, as well as other openings at or below the LIP flood height that would allow water into the Reactor Buildings, Diesel Generator Buildings, Intake Pumping Station and Radwaste Building, were reviewed using site drawings. In addition, these potential flood water ingress paths were observed during a site walkdown performed in February 2015. Flood walls and penetrations were reviewed and determined to be acceptable for the PMF height of 578.0 feet and are acceptable for the maximum 566.6 feet LIP flood elevation.

    - Reactor Building - The LIP flood exceeds the finished floor elevation by only 0.2 feet (2.4 inches). An equipment/personnel air lock and a personnel access door provides access on the south side of the building. Given the airtight design of the doors, the very limited flood height above the door, and the short duration of flood exposure, the amount of water leakage through an inadequate door seal would be very small and no water would be expected in the Reactor Building. A small side door provides personnel access. This door is a normally closed watertight door and has a 3-1/2 inch threshold, which would not be exceeded by the LIP flood.

      The Reactor Building doors on the north side would potentially be exposed to flood water when the outside LIP flood levels exceed the Turbine Building floor elevation of 565.0 feet from either the Lower East or Lower West areas. The personnel access doors are normally closed, watertight doors designed for the PMF water height of 572.5 feet. LIP flood water would not enter the Reactor Building through these doors. A small amount of water in-leakage would be similar to the internal flooding analysis for the 565.0 general area.

    - Diesel Generator Buildings - The Diesel Generator Buildings have a finished floor elevation of 565.5 feet. The LIP flood would exceed the finished floor elevation by 0.7 feet and 1.1 feet for the Unit 1/2 Diesel Generator Building and the Unit 3 Diesel Generator Building, respectively. Both buildings have five similar exterior doors, each accessing one of the four diesel generator bays or the CO_2 room. These doors have removable seal plates extending 10-3/8 inches above the 565.0 feet grade elevation. The external doors are normally closed, watertight doors designed for the PMF water height of 578.0 feet. Little or no LIP flood water would enter the Diesel Generator Buildings through the doors. Given the watertight design of the door, the limited flood height, and the short duration of exposure, the amount of leakage through an inadequate door seal would be small and would not jeopardize DG operation.

      An emergency drain path exists in the corridor outside the diesel generator bays to drain water in the event of an 18-nch Emergency Equipment Cooling Water (EECW) system header break. The emergency drain lines have normally open shutoff valves and are routed to valve pits just outside the buildings. The valve pits are located at site grade. LIP flood water may backflow through the emergency drain lines into the corridor and backflow through the floor drain piping into the diesel generator bay compartments. Backflow of LIP flood water into the diesel building would result in a Diesel Generator Floor Drain Sump Level High alarm in the control room. The alarm response procedure requires Operations personnel to be dispatched to investigate the cause of the alarm. For the LIP, interim actions, such as dispatch of plant operators to close the emergency drain isolation valves based on severe storm forecast would be evaluated. It is noted that for the PMF event, ample time is available and procedural direction is given to close the interior flooding drain isolation valves prior to flooding the diesel bays.

During the PMF, steps are taken to prevent flood water inflow to the 7-day diesel fuel tanks due to potential PMF damage to diesel fuel transfer piping located in the yard. The LIP associated effects, such as debris loads, hydrodynamic and hydrostatic loads, are expected to be negligible due to the low flow velocities and shallow water depths. The truck load connection is rarely used, but operating procedures require the cap to be installed after any use. The truck fill boxes do not represent a potential for fuel oil contamination by flood water.

- Intake Pumping Station - The LIP flood would exceed the finished floor elevation by 1.6 feet. The Intake Pumping Station has four similar exterior doors, each accessing one of the four RHRSW pump compartments. The external doors are normally closed, watertight doors designed for the PMF water height of 578.0 feet. Given the watertight design of the door, the limited flood height, and the short duration of flood exposure, the amount of leakage through an inadequate door seal would be small and within the sump pump capacity margin. The LIP flood would not jeopardize RHRSW pump operation.

- Radwaste Building - The Radwaste Building has a finished floor elevation of 565.0 feet. The LIP flood would exceed the finished floor elevation by 1.2 feet. There are two exterior doors to the Radwaste Building and one exterior door to the Radwaste Evaporator Building, which communicates directly to the Radwaste Building through unprotected openings. The external doors are watertight doors designed for the PMF water height of 578.0 feet. There are two doors to the Radwaste Building from the Service Building. Both doors are watertight and designed for the PMF water height of 572.5 feet. Additional access to the Radwaste Building is via Turbine Building doors at floor elevation 565.0 feet and at floor elevation 554.5 feet.. These doors are normally closed, watertight doors designed for the PMF water height of 572.5 feet.

  Given the watertight design of the doors, the limited flood height, and the short duration of exposure, little or no LIP flood water would enter the Radwaste Building through any of the closed doors. The open doors are in areas that are heavily travelled. Prior analysis indicates that there is sufficient time to close both doors to the Service Building and that flood water entering the Radwaste Building would be handled by the floor drains without degrading flood safety.

- Turbine Building - In the turbine building, the water would generally spread throughout the floor area, spilling into the lower elevations at 557.0 feet and 551.0 feet. The Turbine Building does not contain safety related equipment and is allowed to flood during the PMF. Doors to the Turbine Building open outward and, although not water-tight, could be closed to severely restrict flow of water into the building.

- BFN East Switchyard channel - The LIP re-evaluation of the BFN East Switchyard channel resulted in a flood elevation of up to 578.17 feet, which exceeds the CLB LIP flood elevation of 578.0 feet. Overflow from the East Switchyard channel is fully contained in the Cooling Tower (CT) hot water discharge channel and in the switchyard area. There are no plant safety effects due to this increased flow in the CT discharge channel. Overflow from the BFN East switchyard channel would also enter the switchyard area north of the plant main site. However, the elevation of the site north of the BFN turbine building is at least 578.6 feet, which prevents the overflow from the East Switchyard channel from reaching the lower plant areas.

2. Upstream Dam Breaches or Failures - The re-evaluation of upstream dam breaches or failure results in a maximum BFN site flood elevation of 558.8 feet as a result of the single failure of the Watts Bar dam during a 500-year flood event. Because this elevation is below the plant grade, this condition does not affect the safe operation of BFN. No interim actions are required. Single seismic failures were not considered in the CLB, therefore an IA will be performed as previously discussed.

3. Combined Effects Floods caused by Seismic Dam Failures - The re-evaluation of seismically induced flooding results in a maximum BFN site flood elevation of 560.9 feet. The combined effects of seismically induced dam failure, 500-year storm rainfall and maximum wind wave results in a flood elevation below plant grade. The safe operation of BFN is not affected. No interim actions are required. Combined effects floods caused by seismic dam failures were not considered in the CLB, therefore an IA will be performed as previously discussed.

The latter two mechanisms were not considered to be applicable in the CLB and thus do not have comparable CLB flood elevations.

As stated in the HRR, each of the above mechanisms were evaluated and interim actions defined if needed. In addition to committing to perform an IA, TVA letter dated March 12, 2015, also committed to two interim actions for the BFN site. For the first interim action, TVA committed to revise existing procedures or develop new procedures to ensure that doors or other openings susceptible to a LIP flood event are maintained closed or controlled otherwise by procedure, compensatory measure, and/or work activity to protect ingress pathways in the event of a LIP. These actions have been implemented at BFN.

For the second interim action, TVA committed to determine a resolution to the potential backflow through the Diesel Generator interior flooding drain lines into the diesel generator buildings during the LIP event by September 28, 2015. TVA revised the commitment by letter dated September 28, 2015 (ML15271A278). The revised commitment states, "TVA will perform a two-dimensional hydrologic analysis of the local intense precipitation (LIP) event, and, if required to protect safety-related functions, revise existing procedures or create new procedures to isolate the diesel generator interior flooding drain lines for the LIP event based on the results of the two-dimensional hydrologic model and the NRC-endorsed NEI 15-05 “Warning Time for Local Intense Precipitation Events,” by August 31, 2016."

<u>Transportation, and Nearby Facility Accidents</u>

Section 5.3 of the IPEEE discusses the screening performed for transportation and nearby facility accidents. The IPEEE concludes:

> "The evaluation of Nearby Industrial, Transportation and Military Facilities has not resulted in the identification of any vulnerabilities. BFNP conforms to the 1975 SRP Criteria and, therefore the original design basis analysis of potential hazards in the site vicinity is

> considered adequate and acceptable. Using the progressive screening approach outlined in NUREG 1407 and Supplement 4 to GL 88-20, Nearby Industrial, Transportation and Military Facilities can be screened out for the BFNP IPEEE, and no further analyses of these potential hazards are necessary."

The IPEEE also states:

> "Prior to restart of Unit 2 in May of 1991, the NRC issued a Safety Evaluation Report (SER) including a supplemental SER where the NRC concluded that even if all toxic gases transported by barges past Browns Ferry are considered, the probability that a toxic release would result in a severe accident condition exceeding 10 CFR 100 guidelines is sufficiently small and meets the staff's established regulatory position."

In response to APLA RAI 13.b, TVA provides the following additional information for transportation and nearby facility accidents. The latest evaluation for transportation and nearby facility accidents was reviewed to determine whether any new insights may affect the ECCS PPL evaluations. It should be noted that this information is not intended or considered to be an update to the IPEEE.

TVA calculation NDQ0031890038, "Main Control Room Habitability During A Hazardous Chemical Release," evaluates the habitability of the BFN MCRs for potential hazards resulting from hazardous chemicals stored on or near the site, or chemicals that are transported near the site by barge, rail, road, or pipeline using the guidelines outlined in Regulatory Guide 1.78, Revision 0, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." Revision 7 of NDQ0031890038, issued on July 16, 2014, documented a periodic assessment of offsite hazardous chemical sources.

The calculation reviewed chemicals stored on site, stored in nearby Industrial facilities, and transported within five miles of BFN. Nearby industrial facilities were contacted to provide an update on quantities of hazardous chemicals stored at their respective sites. The companies contacted were Hexel, OCI Chemical, and Nucor Steel. A substantial change in chemical quantities stored was not reported. TVA River Operations provided an update on hazardous chemicals barged past BFN since 2008. There was no substantial increase in the hazardous chemical quantities transported; therefore, the calculation results remain unaffected.

All rail lines lie outside a five·mile radius of the plant and do not have to be considered. Additionally, there are no military activities within the five mile radius of the plant that could affect MCR habitability. Hazardous chemical releases involving the quantities and types of chemicals stored at nearby industrial facilities are bounded by the evaluations done for chemical releases from barge accidents and do not affect MCR habitability. Xylene and Natural Gas pipelines passing within five miles of the BFN site have been evaluated to not affect MCR habitability. Hazardous chemicals shipped by road could be within five miles of the site (i.e., 4.5 miles), but are assumed to be of the same types as shipped by barge. Therefore, chemical shipments by road are assumed to not affect the MCR because the much larger and closer barge shipments are assumed to be bounding and have been evaluated by the calculation to not affect MCR habitability. The calculation also found that hazardous chemicals stored on site could not affect the MCR upon rupture of the largest chemical container in a storage area closest to the MCR ventilation intake.

The calculation considered two types of accidents. The first type is maximum concentration accidents and the second type is maximum concentration duration accidents. Results of the calculation indicate that neither the onsite chemicals in quantities greater than 100 pounds, chemicals shipped past the site by rail, pipeline, or road, nor chemicals stored in industrial facilities within five miles of BFN affect MCR habitability. Of the chemicals barged past the site, only Chlorine has the potential to affect MCR habitability. However, the shipment frequency is low enough (approximately 32 per year) that the calculated frequency of a Chlorine barge accident potentially affecting MCR habitability is below the threshold of 1.0 E-6 per year.

Based on the results of the "Main Control Room Habitability During A Hazardous Chemical Release" calculation, the contribution to risk from transportation, and nearby facility accidents when ECCS PPL is OOS or failed is appropriately addressed by the IPEEE results. Therefore, no changes to the models or PPL evaluation have been performed for transportation and nearby facility accidents.

APLA-RAI-15

Regulatory Guide 1.177 for Tier 2 states that the licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service, consistent with the proposed TS change. Provide the results of your Tier 2 analyses that provide restrictions on dominant risk-significant configurations and discuss the basis for the restrictions.

TVA Response

PRA evaluations determined that taking the ECCS PPL out of service is not risk significant; in fact the risk was evaluated as negligible. Two sensitivity evaluations were performed to determine the risk increase. The first sensitivity assigned a probability of 0.1 to both the reliability and unavailability events included in the model. The results of this sensitivity showed that the risk increase to CDF was less than 1E-7, and LERF increase was negligible. The second sensitivity assumed the reliability of ECCS PPL was 0.1 and the unavailability was set to 1.0 (True). The results of this sensitivity indicated that the increase in CDF was on the order of 1E-6, and the increase in LERF was less than 1E-7.

The risk of taking the ECCS PPL out of service, combined with other equipment was not initially evaluated because it was deemed to be dominated by the risk of other components that are taken OOS at the same time (i.e., RHR or Core Spray). Any potential risk-significant configurations would be identified by the work control process evaluations performed several weeks prior to taking the ECCS PPL components out of service.

In addition, when a plant risk program (i.e., EOOS) model update is performed, the resulting insights (i.e., assigned risk color) for selected systems or trains taken out of service between the prior and current models is included in the EOOS calculations. This is done to help identify and communicate changes in risk associated with taking a system or train out of service. This process helps identify any potential changes in risk insights prior to releasing an EOOS model update.

As noted in the response to APLA RAI 16 provided in TVA letter dated April 29, 2016 (ML16123A071), TVA uses EOOS to perform contemporaneous assessments of the overall effect on the safety of proposed plant configurations before performing and during performance of maintenance activities that remove equipment from service to avoid risk-significant plant configurations. For on-line maintenance, a risk assessment is performed prior to implementation and emergent work is evaluated against the assessed scope. TVA assesses and manages plant configurations prior to entering the maintenance configuration.

ENCLOSURE 2

Tennessee Valley Authority Browns Ferry Nuclear Plant, Units 1, 2, and 3

Summary of BFN Request for Additional Information Response Dates

Request for Additional Information (RAI) Question Number	Due Date	Actual Date of Response
Electrical Engineering Branch (EEEB)		
EEEB RAI 1	May 11, 2016	CNL-16-078, May 11, 2016
EEEB RAI 2	May 11, 2016	CNL-16-078, May 11, 2016
EEEB RAI 3	May 11, 2016	CNL-16-078, May 11, 2016
EEEB RAI 4	May 11, 2016	CNL-16-078, May 11, 2016
Instrumentation and Controls Branch (EICB)		
EICB RAI 1	May 11, 2016	CNL-16-078, May 11, 2016
EICB RAI 2	May 11, 2016	CNL-16-078, May 11, 2016
EICB RAI 3	June 16, 2016	
Probabilistic Risk Assessment Branch (PRA) Licensing Branch (APLA)		
APLA RAI 1	April 15, 2016	CNL-16-066, April 15, 2016
APLA RAI 2	April 15, 2016	CNL-16-066, April 15, 2016
APLA RAI 3	April 15, 2016	CNL-16-066, April 15, 2016
APLA RAI 4	May 11, 2016	CNL-16-078, May 11, 2016
APLA RAI 5	June 16, 2016	
APLA RAI 6a	May 11, 2016	CNL-16-078, May 11, 2016
APLA RAI 6b	May 11, 2016	CNL-16-078, May 11, 2016
APLA RAI 6c	May 11, 2016	CNL-16-078, May 11, 2016
APLA RAI 6d	June 16, 2016	
APLA RAI 7	May 11, 2016	CNL-16-078, May 11, 2016
APLA RAI 8	April 15, 2016	CNL-16-066, April 15, 2016
APLA RAI 9	April 15, 2016	CNL-16-066, April 15, 2016

Request for Additional Information (RAI) Question Number	Due Date	Actual Date of Response
APLA RAI 10	April 29, 2016	CNL-16-076, April 29, 2016
APLA RAI 11	April 29, 2016	CNL-16-076, April 29, 2016
APLA RAI 12	May 25, 2016	CNL-16-082, May 25, 2015
APLA RAI 13a	June 16, 2016	
APLA RAI 13b	May 25, 2016	CNL-16-082, May 25, 2015
APLA RAI 13c	May 25, 2016	CNL-16-082, May 25, 2015
APLA RAI 14	June 16, 2016	
APLA RAI 15	May 25, 2016	CNL-16-082, May 25, 2015
APLA RAI 16	April 29, 2016	CNL-16-076, April 29, 2016

Summary

April 15, 2016: APLA RAI 1, 2, 3, 8, 9
April 29, 2016: APLA RAI 10, 11, 16
May 11, 2016: APLA RAI 4, 6a, 6b, 6c, 7; EEEB RAI 1, 2, 3, 4; EICB RAI 1, 2
May 25, 2016: APLA RAI 12, 13b, 13c, 15
June 16, 2016: APLA RAI 5, 6d,13a, 14; EICB RAI 3