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

March 21, 2016

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST FOR ADDING NEW SPECIFICATIONS TO TECHNICAL SPECIFICATION 3.3.8.3 (CAC NOS. MF6738, MF6739, AND MF6740)

Dear Mr. Shea:

By letter dated September 16, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15260B125), Tennessee Valley Authority, (TVA, the licensee) requested an amendment to Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68, for the Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3, respectively. The proposed changes revise BFN, Units 1 and 2, Technical Specifications (TSs) 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. The changes proposed for BFN, Unit 3 (i.e., relocating the requirements for common accident signal logic and unit priority re-trip log, are made for consistency with the changes to BFN, Units 1 and 2, TSs).

On February 1, 2016, a public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) and representatives of TVA at NRC headquarters. The purpose of the meeting was for TVA to provide a description of logic functions; a summary of changes in the license amendment request; and for the NRC staff to discuss its draft request for additional information (RAI) and preliminary concerns associated with pump logics, loss of power, and use of the BFN probabilistic risk assessment (PRA) program with the TVA representatives. The meeting summary is located under ADAMS Accession No. ML16043A277.

The NRC staff reviewed the information provided in the license amendment request and the presentation slides that were provided during the public meeting held on February 1, 2016 (ADAMS Accession No. ML16028A096), and determined that additional information is needed. On February 11, 2016, the NRC staff forwarded to TVA, by e-mail, a revised RAI from staff of the Instrumentation and Controls Branch, and a draft RAI from staff of the Electrical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation. On February 19, 2016, the NRC staff forwarded another e-mail to TVA that contained a draft RAI from the staff of the Probabilistic Risk Assessment Licensing Branch. On February 26, 2016, staff from the NRC and TVA staff held a conference call to provide the licensee an opportunity to clarify any portion of the staff RAI and discuss the timeframe for which TVA may provide the requested information. The finalized NRC staff's RAI (Enclosure 1) was discussed with Mr. Edward Schrull of your staff. Mr. Schrull provided proposed dates for response to the staff RAI (Enclosure 2). The NRC staff agreed with the proposed dates.

If you have any questions, please contact me at 301-415-1447 or Farideh.Saba@nrc.gov.

Sincerely,

Farideh E Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

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

cc w/enclosures: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST RELATED TO

TECHNICAL SPECIFICATION CHANGES TO REACTOR CORE SAFETY LIMITS

TENNESSEE VALLEY AUTHORITY

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

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

By letter dated September 16, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15260B125), Tennessee Valley Authority, (TVA, the licensee) requested an amendment to Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68, for the Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3, respectively. The proposed changes revise BFN, Units 1 and 2, Technical Specifications (TSs) by adding a new specification governing the safety functions for the emergency core cooling system (ECCS) preferred pump logic (PPL), common accident signal (CAS) logic, and the unit priority re-trip logic (UPRTL). The changes proposed for BFN, Unit 3 (i.e., relocating the requirements for CAS logic and UPRTL, are made for consistency with the changes to BFN, Units 1 and 2, TSs).

On February 1, 2016, a public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) and representatives of TVA at NRC headquarters. The meeting summary is located under ADAMS Accession No. ML16043A277.

The U.S. Nuclear Regulatory Commission (NRC) staff from the Electrical Engineering Branch (EEEB) and Instrumentation and Controls Branch (EICB), Division of Electrical Engineering, and the staff from the Probalistic Risk Assessment (PRA) Licensing Branch (APLA), Division of Risk Assessment, Office of Nuclear Reactor Regulation, have reviewed the information provided in the license amendment request (LAR) and the information provided during the public meeting held on February 1, 2016 (ADAMS Accession No. ML16028A096).

EEEB-Request for Additional Information (RAI)-1

In the presentation slides provided by TVA during the public meeting, slide 9 states that the CAS blocks the 4 kV shutdown board auto transfer logic, and blocks the 4 kV degraded voltage trips.

a. Explain how the CAS blocks the 4 kV shutdown board auto transfer logic and blocks the 4 kV degraded voltage trips. Also, explain the purpose of these CAS blocks and clarify whether the CAS affects the degraded voltage signal that initiates the emergency diesel generator.

b. The following is stated in the Updated Final Safety Analysis Report (UFSAR), Section 8.4.5.2:

> Each board and the startup bus has its source breakers interlocked to prevent paralleling power sources, and each is provided with manual and automatic bus transfer schemes. Automatic transfers are initiated by generator and transformer protective relays, degraded under voltage on 4160-V shutdown boards and loss of voltage at the normal supply (except for loss of voltage on 4160-V unit board 1A, 1B, 2A, and 2B). Transfer is blocked through manually-reset lockout relays in case of faulted bus. Each bus section is provided with a manual-automatic transfer selector switch.

The above paragraph does not state that automatic transfer at the 4 kV shutdown board is blocked by CAS. Clarify whether CAS blocks the automatic transfer at the 4 kV shutdown board.

EEEB-RAI-2

Presentation slide 9 also states that CAS trips the RHRSW (Residual Heat Removal Service Water) Pumps A2 and C2. Explain the purpose of CAS tripping the RHRSW Pumps A2 and C2. Also explain why the CAS tripping does not apply to the other set of RHRSW Pumps B2 and D2.

EEEB-RAI-3

Assuming loss-of-offsite power event, provide the latest summary of loadings of all eight emergency diesel generators (EDGs) with the PPL, for first 24 hours duration, assuming:

a. Loss-of-coolant accident (LOCA) in Unit 1, followed by spurious accident in Unit 2;

b. Spurious accident in Unit 1, followed by LOCA in Unit 2;

c. LOCA in Unit 1 and loss of an EDG; and,

d. LOCA in Unit 2 and loss of an EDG.

Only worst case loading combinations from a or b and from c or d may be provided.

EEEB-RAI-4

The BFN UFSAR, Section 8.4.4, "Safety Design Basis," states:

> The normal and alternate offsite power circuits for each unit shall each be sufficient to supply the power to shut down the unit and maintain it in a safe condition under normal or accident situations. One of these circuits shall be available within a few seconds following a loss of coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. The other circuit shall be available in sufficient time to assure that

plant safety design limits are not exceeded. Only one unit is assumed to be in an accident condition.

a. Explain how the above design-basis criteria continues to be satisfied with the PPL. Also, explain whether single failure in the offsite power system circuits has been considered while analyzing accident conditions with the PPL (when the accident loads are fed from offsite power system circuits).

b. TS Bases 3.8.1 stated:

> The Class 1E AC distribution system is divided into redundant divisions, so loss of any one division does not prevent the minimum safety functions from being performed. Each of four 4.16kV shutdown boards has two offsite power supplies available and a single DG [diesel generator]. <u>Only offsite power delivered through the normal feeder breakers can be credited since common accident signal (CAS) logic (CAS A/CAS B) will trip the alternate breaker.</u> This prevents an overload condition if all shutdown boards had been aligned to the same shutdown bus, and thus to the same transformer winding.

Based on the above, it appears that even though each 4.16 kV shutdown board is provided with two offsite circuits, only one normally closed offsite circuit can be credited in case of an accident. Considering this, explain the purpose of the following statement in the UFSAR, Section 8.4.4, "Safety Design Basis": "The other circuit shall be available in sufficient time to assure the plant safety design limits are not exceeded."

<u>EICB-RAI-1</u>

The LAR describes that the core spray system is initiated by sensors and relays based on low reactor water level (Level 1 setpoint) or high drywell pressure, coincident with low reactor pressure. These same sensors and relays are used to initiate the CAS and the LAR references the UFSAR, Section 7.4. One of the functions of the CAS is to send a signal to start all eight Unit 1, 2, and 3 DGs. The UFSAR (BTN 25; page 7.4-18) also refers formally to a signal labeled "Pre-Accident Signal" (note all first letters are capitalized giving the impression this is a specific signal) that is generated by low reactor water level (Level 1 setpoint) or high drywell pressures and sends a signal to start all eight Unit 1, 2, and 3 DGs. The use of the "Pre-Accident Signal" term was not discussed in the LAR. Is this "Pre-Accident Signal" a separate duplicate signal and part of the logic system?

<u>EICB-RAI-2</u>

In the LAR, Section 4.1, "System Description," under "Unit Priority Re-Trip Logic," the first sentence states, "Following an initiation of a CAS on either Unit 2 or 3 (which trips all eight DG output breakers), subsequent accident signal trips of the DG output breakers are blocked." Considering the twin-like relationship between BFN, Unit 1 and Unit 2, should this actually state something to the effect, "Following an initiation of a CAS on either Unit 2 or Unit 3, or either Unit 1 or Unit 3 (either combination, which will trip all eight DG output breakers), subsequent accident signal trips of the DG output breakers are blocked?"

EICB-RAI-3

The BACKGROUND in the BASES for the new TS 3.3.8.3 for BFN, Unit 1 {page B 3.3-275; page 117 of the LAR} states, "In the event of an accident signal in either Unit 1 or Unit 2, all of the ECCS equipment associated with the accident unit will start." It further states:

> The diesel generators and Standby AC Power System are designed to accommodate spurious accident signals from any unit and in any order, real followed by a spurious signal, real coincident with a spurious signal, and spurious followed by a real accident signal. If the ECCS loads for both Units 1 and 2 were allowed to start during combinations of real and spurious accident signals, the combined Unit 1/2 ECCS pumps would overload the 4KV shutdown boards and their associated diesel generators." In CONDITIONS B and C, when one of the two logic division is inoperable (there are 2 CAS logic division and 2 unit priority re-trip logic divisions) COMPLETION TIME to repair is 7 days, justified by the fact the other division is available. In CONDITION A, both divisions of ECCS PPL can be INOPERABLE with up to 7 days to restore, in which time there is no automatic logic and, from BFN own BASES, the plant is at risk of, "overload(ing) the 4KV shutdown boards and their associated diesel generators."

Provide an explanation why CONDITION A should not be the same as CONDITIONS B AND C in allowing only one INOPERABLE ECCS PPL.

APLA-RAI-1

The LAR does not appear to clearly indicate the proposed risk-informed TS changes. Summarize which TS Limiting Conditions of Operation (LCOs) and Conditions or Surveillance Requirements are being proposed as risk-informed TS changes and have been evaluated using PRA for this LAR.

APLA-RAI-2

Explain the equipment impact of the PPL unavailabilities allowed by the proposed TS change and how it is modeled in the internal events PRA. Include an explanation of same unit and opposite unit equipment impacts. Indicate whether the PRA evaluates single and multiple unit risk for the PPL unavailabilities and is included in the PRA results.

APLA-RAI-3

Explain whether PPL unavailabilities could result in no low pressure injection (as well as no high pressure injection) following a LOCA (to include feedwater and steamline break). If so, explain the plant's defense-in-depth for the LOCA scenario.

APLA-RAI-4

The CAS system, the Pre-Accident Signal (PAS) system, and the UPRTL system respond to LOCA scenarios, including feedwater and steamline break. Explain how one or more of PPL divisions unavailable affects the functions of the CAS, PAS, and the UPRTL systems.

APLA-RAI-5

The CAS and the PAS are initiated by the core spray (CS) initiation logic, the UPRTL is initiated by the residual heat removal (RHR) logic, and the PPL is initiated by the CS and RHR initiation logic, according to the public meeting slides dated February 1, 2016 (ADAMS Accession No. ML16028A096). Since PPL division(s) inoperable concurrently with failures of CS and RHR initiation logic impacts the plant response, the PRA should model (1) maintenance and testing of the PPL, CS, and RHR initiation logic if allowed when PPL division(s) are inoperable; and (2) the unreliability of the PPL, CAS, PAS, and UPRTL, conservatively or in detail. Update the internal events PRA as necessary. Discuss the updates and explain the impact on the PRA results reported in the LAR as part of APLA-RAI-14.

APLA-RAI-6

The LAR in Section 3.0, "Background," discusses (1) a potential overloading of diesel generators if the ECCS pumps were started out of their required sequence; and (2) overloading of a diesel if an RHR pump was allowed to start on a diesel that was already loaded with any large 4 kV load, as well as the potential to overload affected shutdown boards with normal power available if an RHR pump were to start on a board already loaded with a CS pump and an emergency equipment cooling water pump.

The LAR also states:

> Following the shutdown of all three BFN units in 1986, a Condition Adverse to Quality Report was initiated to document that the AC power supply system and ECCS initiation logic could not accommodate various combinations of spurious and real accident signals as described in the UFSAR Section 8.5.2. As part of the Base Line Commitment process, TVA identified that modification of the BFN Accident Signal Logic and Unit 1/2 Preferred Pump Logic would be required to support continued multi-unit operations.

The PPL modification appears to be part of the resolution of the power reliability issues. It is not clear that the reliability of normal and emergency power has been analyzed with respect to the CAS, PAS, or UPRTL functions, or for additional equipment out-of-service for maintenance (other than PPL-related components) that would not load onto the bus. The reliability of normal and emergency power, given PPL division(s) inoperable as allowed by the proposed TS LCO, should be appropriately accounted for in the PRA.

Address the following related to the reliability of the power sources:

a. It appears that the Unit 1/2 PPL modification is related to resolving the power reliability issues described in the LAR. Discuss whether the proposed TS changes allowing PPL division(s) to be inoperable at-power or during shutdown modes can affect the reliability of normal or emergency power for the accident unit(s). If so, explain the scenarios.

b. Discuss the normal or emergency power reliability impacts, if any, given PPL division(s) inoperable and a CAS, PAS, or UPRTL signal is received. If there are any impacts, explain the scenarios.

c. If PPL division(s) loads and other large bus load(s) are out-of-service for maintenance, discuss whether the reliability of sequencing on loads following an accident signal can be impacted by not having enough loads on the bus. If so, explain the scenarios.

d. If power reliability can be affected as in parts i, ii, or iii, describe how your PRA models the impact on normal and emergency power reliability, if any. If there is a reliability impact that is not modeled in the PRA, update your internal events PRA. Discuss the updates, if any, and explain the impact on the PRA results reported in the LAR as part of APLA-RAI-14.

APLA-RAI-7

The LAR states, for the proposed TS 3.3.8.3:

> The division(s) of ECCS Preferred Pump Logic required to be operable during operation in Modes 4 and 5 dependent on the configuration of the RHR or Core Spray pumps required to be operable, or in operation.

The LAR does not discuss shutdown risk. Explain why shutdown risk is not included in the LAR; otherwise, discuss your shutdown risk assessment. The LAR states that, "The opposite unit RHR pumps that are tripped by the ECCS Preferred Pump Logic are locked out from manually re-starting for 60 seconds." Include this impact on RHR cooling in your discussion, as well as your assessment of the reliability of recovering RHR manually prior to boiling. Discuss shutdown risk results as part of APLA-RAI-14.

APLA-RAI-8

Prior to the proposed TS change, there was no explicit LCO for UPRTL. The LAR states that the LCO for UPRTL is implicitly required by the TS 3.8.1 Condition D, which applied to the CAS. In the proposed TS change for Unit 1 and Unit 2, LCO 3.3.8.3, Condition B, applies to CAS, and LCO 3.3.8.3, Condition C, applies to UPRTL. For Unit 3, LCO 3.3.8.3, Condition A, applies to CAS, and LCO 3.3.8.3, Condition B, applies to UPRTL. These TS LCO Conditions for the UPRTL have a 7-day completion time and are new, proposed changes. If these UPRTL TS LCOs are proposed as risk-informed changes, then include the associated risk for Units 1, 2, and 3 in the results reported in the LAR as part of APLA-RAI-14.

APLA-RAI-9

Note 1 to the proposed SR 3.3.8.3.1 states that, "When a division is placed in an inoperable status solely for performance of a surveillance, entry into associated Conditions and Required Actions may be delayed for up to six hours provided the associated redundant division is OPERABLE." The proposed TS Bases state that the, "PRA demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the ECCS Preferred Pump, CAS, and Unit Priority Re-trip Logics will initiate when necessary." If the LAR is justifying the

6 hour testing allowance time as a risk-informed TS change, then provide the PRA justification for the 6 hours testing allowance for the ECCS PPL, CAS, and UPRTL. Provide the LAR risk results impact as part of APLA-RAI-14.

APLA-RAI-10

The LAR states that the 2009 peer review of the internal events PRA found that 63 supporting requirements were not met or met at Capability Category I. The NRC staff notes that the findings and observations (F&Os) from the 2009 peer review and their disposition were provided to the NRC as a part of the BFN application to transition to National Fire Protection Association (NFPA) Standard 805. The NFPA 805 LAR states that the peer reviews in May and September of 2009 resulted in 125 F&Os for the internal events and the internal flooding PRA.

Clarify the discrepancy in number of F&Os reported in the NFPA 805 LAR and in the current LAR.

a. Resolution to F&O 2-31 states:

   The common cause failure probability of two Motor Operated Valves (MOV's) to close is less than 1E-5. The Residual Heat Removal (RHR) pump start failure probability is approximately 1.4E-3. The failure of two MOV's to close is less than 2 orders of magnitude lower than another failure that would fail the system in a similar manner. Therefore, failure to close (or modulate) either the Low Pressure Coolant Injection (LPCI) or Suppression Pool Cooling (SPC) injection path can be neglected." NUREG/CR-6928 indicates a failure rate of ~1E-3/demand for an MOV failing to close. NUREG/CR-5497 indicates a Common Cause Failure (CCF) probability for a Boiling Water Reactor (BWR) RHR MOV to close of ~ 3.2% (CCCG=2), suggesting a "generic" CCF probability of ~3.2E-5/demand, or slightly above the two-order-of-magnitude threshold for neglecting the failure rate. Confirm that plant-specific Bayesian updating of the generic MOV failure rate could reduce this to the claimed <1E-5/demand. In addition, if these MOVs represent a potential interfacing system loss-of-coolant accident (ISLOCA) concern, discuss whether this pathway falls into the ISLOCA frequency used in the PRA and if it meets the screening supporting requirements IE-A2 and IE-C6.

b. F&Os 4-21: The peer review questioned the joint human error probability (HEP) for several combined operator actions as possibly being too low. In the resolution, the licensee noted that the human reliability analysis (HRA) calculator provides the capability to explicitly calculate the joint HEP for dependent and independent human failure events, thereby reducing potential conservatism that could result from the use of a threshold lower bound a priori, such as 1E-5 as suggested by NUREG-1792. Discuss the impact on the application (using a PRA which accounts for any updates for this application) of using the HRA calculator results as lower bounds instead of the suggested threshold lower bound value of 1E-5.

c. F&O 6-5 found that high pressure coolant injection (HPCI) steamline breaks are excluded as an initiator from the PRA. The pipe break frequencies used old Electric

Power Research Institute (EPRI) and WASH-1400 data, apparently a factor of 100 lower than more recent pipe rupture data. Also, a lower CCF value from NUREG/CR-6928 was used. The licensee's resolution states that until the newer EPRI pipe rupture frequency data are publicly released, results remain based on the older data, and that although the correct CCF value is now used, the results do not change the conclusion to exclude the HPCI steamline break as an initiator. The HPCI steamline break frequency of 1.97E-09/year does not appear consistent with NUREG/CR-6928. NUREG/CR-6928 estimates a frequency for BWR medium LOCA of 1E-4/year (which includes diameters for steam). Justify the exclusion of HPCI steamline breaks or update the internal events PRA model to include this initiator. If the PRA is updated, discuss the update and provide the risk impact on the application as part of APLA- RAI-14.

APLA-RAI-11

Provide an overview of the changes in the internal events PRA that occurred after the 2009 peer review, and determine whether any of these changes qualify as a PRA upgrade that would require a focused scope peer review.

APLA-RAI-12

The 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 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.

APLA-RAI-13

Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Techhnical 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.

a. The LAR states that the fire model considers spurious operation of the ECCS PPL and that adding unavailability of the ECCS PPL to the fire model would decrease CDF/LERF. The LAR does not provide sufficient information for the NRC staff to find this conclusion reasonable. If the fire PRA does not include the PPL unavailability, update the fire PRA to include the PPL unavailability associated with the proposed TS changes, as well as its unreliability, consistent with the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." In addition, include changes to the internal events PRA, as appropriate, for the fire PRA. Discuss the changes made and include the results for the proposed TS changes as part of APLA-RAI-14.

b. The LAR states that the BFN Individual Plant Examination of External Events (IPEEE) did not calculate a CDF or LERF due to high winds/tornadoes. The IPEEE analysis concluded that the CDF from high winds was judged to be less than 1E-6/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.

c. 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.

APLA-RAI-14

If changes were made to the PRAs or other risk assessments related to APLA-RAIs 5, 6.iv, 7, 8, 9, 10.d, 12, 13.a, 13.b, or 13.c, provide updated CDF, ΔCDF, LERF, and ΔLERF results for the LAR.

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.

APLA-RAI-16

RG 1.177, Sections 2.3.7.1 and 2.3.7.2 discuss the configuration risk management program (CRMP) and the key principles of the CRMP. Evaluate the BFN CRMP against the RG 1.177 CRMP key principles and confirm that the BFN CRMP satisfies these key principles.

TENNESSEE VALLEY AUTHORITY PROPOSED RESPONSE DATES

Tennessee Valley Authority (TVA) proposed to respond to the U.S. Nuclear Regulatory Commission's (NRC's) request for additional information (RAI) in three separate letters with due dates of April 15, 2016; April 29, 2016; and May 25, 2016, as shown below. TVA stated that the proposed schedule considers:

1. Browns Ferry Nuclear Plant (BFN) is currently in a refueling outage, resulting in resource constraints for the technical staff to develop and review responses to the RAI.

2. The current probabilistic risk assessment (PRA) model considers the logic systems to be part of the associated ECCS systems and does not consider individual components in sufficient detail to respond to many of the PRA RAI.

3. The current PRA model incorporates all of the National Fire Protection Association (NFPA) 805 related modifications, some of which have not yet been installed. To respond to some of the RAIs, the not-yet-installed major NFPA 805 modifications must be removed from the PRA model.

4. Consequently, the PRA model must be revised and then be re-quantified in order to respond to the remainder of the PRA RAIs.

RAI	**Proposed Response Date**
APLA-RAI-1	4/15/16
APLA-RAI-2	4/15/16
APLA-RAI-3	4/15/16
APLA-RAI-4	5/25/16
APLA-RAI-5	5/25/16
APLA-RAI-6i	4/15/16
APLA-RAI-6ii	4/15/16
APLA-RAI-6iii	4/15/16
APLA-RAI-6iv	5/25/16
APLA-RAI-7	4/15/16
APLA-RAI-8	4/15/16
APLA-RAI-9	4/15/16
APLA-RAI-10a	4/29/16
APLA-RAI-10b	4/29/16
APLA-RAI-10c	4/29/16
APLA-RAI-10d	4/29/16
APLA-RAI-11	4/29/16
APLA-RAI-12	5/25/16
APLA-RAI-13a	5/25/16
APLA-RAI-13b	5/25/16
APLA-RAI-13c	5/25/16
APLA-RAI-14	5/25/16
APLA-RAI-15	5/25/16
APLA-RAI-16	4/29/16

EEEB-RAI-1a	4/15/16
EEEB-RAI-1b	4/15/16
EEEB-RAI-2	4/15/16
EEEB-RAI-3	4/15/16
EEEB-RAI-4	4/15/16
EICB-RAI-1	4/15/16
EICB-RAI-2	4/15/16
EICB-RAI-3	4/15/16

If you have any questions, please contact me at 301-415-1447 or Farideh.Saba@nrc.gov.

Sincerely,

/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

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

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ADAMS Accession No.: ML16074A126 *** By email**

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