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

CNL-16-077

April 27, 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: **Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU) - Supplement 14, Responses to Requests for Additional Information**

References:

1. Letter from TVA to NRC, CNL-15-169, "Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU)," dated September 21, 2015 (ML15282A152)

2. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information Related to License Amendment Request Regarding Extended Power Uprate (CAC Nos. MF6741, MF6742, and MF6743)," dated April 14, 2016 (ML16085A079)

By the Reference 1 letter, Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) for the Extended Power Uprate (EPU) of Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3. The proposed LAR modifies the renewed operating licenses to increase the maximum authorized core thermal power level from the current licensed thermal power of 3458 megawatts to 3952 megawatts. During their technical review of the LAR, the Nuclear Regulatory Commission (NRC) identified the need for additional information. The Reference 2 letter provided NRC Requests for Additional Information (RAI) related to probabilistic risk assessment. The due date for the responses to the NRC RAIs provided by the Reference 2 letter is April 27, 2016. The enclosure to this letter provides the responses to the RAIs included in the Reference 2 letter.

TVA has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in the Reference 1 letter. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the supplemental information in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed license amendment. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Alabama State Department of Public Health.

There are no new regulatory commitments associated with this submittal. If there are any questions or if additional information is needed, please contact Mr. Edward D. Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 27th day of April 2016.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure: Responses to NRC Requests for Additional Information PRA-RAI 1, PRA-RAI 2, PRA-RAI 3, PRA-RAI 4, PRA-RAI 5, PRA-RAI 6, PRA-RAI 7, PRA-RAI 8 and PRA-RAI 9

cc:

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

ENCLOSURE

Responses to NRC Requests for Additional Information
PRA-RAI 1, PRA-RAI 2, PRA-RAI 3, PRA-RAI 4, PRA-RAI 5, PRA-RAI 6, PRA-RAI 7, PRA-RAI 8 and PRA-RAI 9

ENCLOSURE

APLA-RAI 01

In accordance with Appendix D of Standard Review Plan (SRP), Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," dated June 2007, the NRC staff reviews the risk evaluations of the BFN, Units 1, 2, and 3, extended power uprate (EPU) to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

In response to probablisitic risk assessment (PRA) RAI 24 associated with the TVA's LAR to transition to National Fire Protection Association 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" ADAMS Accession No. ML14363A057), the licensee provided the results of a composite analysis that shows the integrated impact on the fire risk (i.e., core damage frequency (CDF), large early release frequency (LERF), change in CDF (ΔCDF), and change in LERF (ΔLERF)) after replacing identified methods and weaknesses with alternative methods that are acceptable to the NRC. It is not clear from Sections 1.2.3, "FPRA Quality," and A.2, "FPRA Technical Adequacy," in Attachment 44, "Probabilistic Risk Assessment," of the EPU LAR whether the fire PRA (FPRA) model used for the EPU risk evaluation (hereafter referred to as EPU FPRA) incorporated the modeling changes reflected in the composite analysis in the response to PRA RAI 24. Provide the following additional information:

(a) *Clarify whether the EPU FPRA model incorporated the modeling changes identified in the composite analysis in the response to PRA RAI 24. If the EPU FPRA model did not include these modeling changes, then (1) provide updated EPU risk results (including sensitivity analyses) using the EPU FPRA model updated for these changes, or (2) explain how these modeling changes would not significantly impact the EPU risk results.*

(b) *Indicate whether a peer review(s) has been performed for those modeling changes identified in the composite analysis. As applicable, provide a list of the facts and observations (F&Os) from this peer review(s), and explain how these F&Os were dispositioned for this application. If a peer review was not performed for these modeling changes, then provide sufficient information for NRC staff to compare the technical adequacy of the analysis to Regulatory Guide (RG) 1.200,"An Approach for Determining the Technical Adequacy of Probablistic Risk Assessment Results for Risk-Informed Activities," or justify why a peer review was not necessary (e.g., modeling change is not considered a PRA "upgrade").*

TVA Response:

(a) The Extended Power Uprate (EPU) fire probabilistic risk assessment (FPRA) model incorporates the changes discussed in the December 17, 2014, response to Browns Ferry Nuclear Plant (BFN) National Fire Protection Association Standard 805 (NFPA 805) PRA RAI 24 (ML14363A057).

(b) As noted in Section A.2 of Attachment 44, “Probabilistic Risk Assessment,” of the EPU License Amendment Request (LAR), a focused scope peer review of the BFN FPRA was conducted from May 12 to 15, 2015. This focused scope peer review evaluated specific changes made to the FPRA and assessed specific F&Os from the previous two FPRA peer reviews, including certain changes described in the response to PRA RAI 24. The final conclusion of the peer reviews was that the BFN FPRA meets Capability Category II following final resolution and closure of all of the F&Os. A complete list of open finding level F&Os, their dispositions, and their impacts to the EPU are provided in Table A-2 of Attachment 44, “Probabilistic Risk Assessment,” of the EPU LAR.

APLA-RAI 02

In accordance with Appendix D of SRP for the review of Safety Analysis Reports for Nuclear Power Plants, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

Section 4.1.3, "Accident Sequences," in Attachment 44 of the EPU LAR states:

> *"However, BFN uses a relatively simplified approach for LOOP recovery based on early recovery or late recovery of offsite power (30 minutes or 4 hours). It was concluded that the change in power would not impact the LOOP recovery file."*

It is not clear to the staff why the change in reactor power does not impact the loss of offsite power (LOOP) recovery file. Describe the simplified approach for LOOP recovery, and justify why the change in power due to the EPU does not impact LOOP recovery

TVA Response:

There are two times of interest with respect to recovery of offsite power: approximately 0.5 hour and 4 hours after the Loss-of-Offsite Power (LOOP). Approximately one half-hour is the time to core uncovery during a LOOP with failure of all injection sources. Approximately four hours is the expected battery life with no load reduction. Because battery life is unaffected by the EPU, this factor remains the same. Therefore, the time of interest for considering the effects of the EPU is the time to core uncovery.

The core uncovery time is an input to the calculation of LOOP recovery factors. The time from the start of the event until core uncovery is calculated to determine the minimum amount of time available to recover offsite power. For the purposes of this calculation, it is assumed that there is no power available from the diesel generators for the duration of the LOOP, which gives a conservative time to core damage. The time to core uncovery during a LOOP with failure of all injection sources is calculated to be 32 minutes under Current Licensed Thermal Power (CLTP) conditions and 30 minutes under EPU conditions. The same core uncovery times are used in calculating HFEs that involve loss of initial injection sources (for example, see HFA_0LPIINIT30, Failure to Establish Low-Pressure Injection Given Loss of High Pressure Injection, in Table B-1 of Attachment 44 of the EPU LAR).

LOOP initiating events are divided into four categories: plant centered, switchyard centered, grid related, and weather related. Table 2-1 provides estimated probabilities that the LOOP will not be recovered within 30 minutes and within 32 minutes for EPU and CLTP, respectively. The probabilities in Table 2-1 are calculated from the lognormal parameters given in Table 4 of the LOOP 2010 Summary Update to NUREG/CR-6890 (Analysis of Loss of Offsite Power Events, 2010 Update - http://nrcoe.inel.gov/resultsdb/publicdocs/LOSP/loop-summary-update-2010.pdf).

The CLTP and EPU PRA models both use a 30 minute core uncovery time. This is more conservative for CLTP than it is for EPU. The impact of the two-minute difference in uncovery time is not included in the CLTP model because the associated change in probability of failure to recover offsite power has an insignificant impact on risk. The change in risk can be estimated

using the Fussell-Vesely (FV) importances. For example, note in Table 2-1 that for Unit 1 Core Damage Frequency (CDF), the internal events FV importances are approximately 6E-3 at most, which implies that a reduction of a few percent in LOOP recovery failure probability to model CLTP will not cause a significant reduction in CDF. For example, the FV for plant centered LOOP for Unit 1 is 5.98E-3 and the fractional reduction in failure probability is 3.5%, which is the largest recovery probability change among the four LOOP categories (Table 2-1). The product (5.98E-3)(3.5%) = 0.021% gives the approximate percentage reduction in CDF from EPU to CLTP for the plant centered LOOP recovery. That is, ΔCDF = (0.021%)(7.14E-06/yr) = 1.5E-9/yr, where 7.14E-06/yr is the Unit 1 internal events EPU CDF from Table 5-1 of Attachment 44 of the EPU LAR. To continue the example for Unit 1 Large Early Release Fraction (LERF), ΔLERF = (4.80E-3)(3.5%)(1.61E-6/yr) = 2.7E-10/yr, where 1.61E-6/yr is the Unit 1 internal events EPU LERF from Table 5-1 of Attachment 44 of the EPU LAR. These ΔCDF and ΔLERF contributions from changes in LOOP recovery probabilities (ΔCDF = 1.5E-9/yr and ΔLERF = 2.7E-10/yr) are not significant contributions to overall internal events delta risks for implementation of the EPU FPRA and may be omitted from the delta risk calculation. The LOOP recovery failure probabilities for other plant units have similar importances and imply similar insignificant delta risks.

Table 2-1. LOOP Recovery Failure Probabilities

	Time to Uncovery (min)	Plant Centered	Switchyard Centered	Grid Related	Weather Related
EPU	30	5.22E-01	6.03E-01	8.63E-01	7.95E-01
CLTP	32	5.04E-01	5.84E-01	8.48E-01	7.86E-01
% Reduction	--	3.5%	3.1%	1.7%	1.1%
U1 CDF BE	--	HFR_1PC30MIN	HFR_1SW30MIN	HFR_1GR30MIN	HFR_1WE30MIN
U1 EPU CDF FV	--	5.98E-03	5.67E-03	7.40E-04	4.41E-03
U1 EPU LERF FV	--	4.80E-03	5.72E-03	7.20E-04	4.19E-03
U1 CLTP CDF FV	--	5.58E-03	5.09E-03	6.70E-04	4.03E-03
U1 CLTP LERF FV	--	4.17E-03	4.89E-03	5.90E-04	3.57E-03

APLA-RAI 03

In accordance with Appendix D of SRP, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

(a) Changes in human error probabilities (HEPs) due to the EPU is a significant contributor to the change in risk. Section 4.1.6, "Human Error Probabilities," in Attachment 44 of the EPU LAR states:

> *"All operator actions in the model were screened to determine the impact from EPU. However, the analysis focused additional scrutiny on several operator actions that were considered significant to the results. The operator actions identified for explicit review were selected based on the following criteria: "Time critical evolutions (i.e., less than 45 minutes available) action."*

The description of the approach for screening operator actions is not clear in Section 4.1.6 of LAR Attachment 44. For example, it is not clear how "all operator actions in the model were screened," or what is meant by "45 minutes available." Clarify the approach for screening all operator actions. In this discussion, provide the basis for using a time critical evolution of 45 minutes, and explain what this time represents (e.g., system time window (T_{sw}) or time available for action (T_{avail})). Also, justify why risk significant operator actions (e.g., operator actions having a high Fussell-Vesely or a high risk achievement worth importance to CDF or LERF) were not considered for detailed review.

(b) Section 4.2.4, "Post Fire Human Error Probabilities," in Attachment 44 of the EPU LAR states:

> *"All human failure events (HFEs) in the FPRA model were reviewed and considered for modification for the FPRA EPU impact study. HFEs that satisfy either of the following conditions were selected for detailed review:*
>
> - *Fussell-Vesely (FV) importance greater than 5E-03, or*
> - *T_{sw} less than 40 min."*

The description of the approach for screening operator actions in Section 4.2.4 of LAR Attachment 44 is not fully understood. Provide the basis for using a system time window of 40 minutes in identifying operator actions for detailed review. Also, justify why risk significant operator actions based on a high risk achievement worth (RAW) (e.g., RAW importance to CDF or LERF greater than 2) were not considered for detailed review.

TVA Response:

(a) The sentence "All operator actions in the model were screened to determine the impact from EPU" used in Section 4.1.6 of LAR Attachment 44 was intended to express that all operator actions in the model were reviewed. It was not meant to imply that human failure events (HFEs) were eliminated from further review based on a screening criterion. All HFEs of the internal events PRA were evaluated for potential impact from EPU.

This evaluation involved a review of the performance shaping factors (PSFs) associated with the operator actions. The review showed that, among these PSFs, the timing element was the only one to have a significant impact on the human error probability (HEP) associated with the HFEs. Other PSFs, such as potential re-scaling of indicators and setpoint changes, which represents the human-machine interface PSF, were found to have no measurable impact on human performance (Section 4.1.6 of LAR Attachment 44).

The HFEs having a high risk importance measure (Fussell-Vesely or risk achievement worth) are noteworthy, because a change in their HEP has the potential to lead to important changes in the risk metrics of the plant, i.e., CDF or LERF. As such, these HFEs were candidates for a detailed review. However, upon further investigation it was realized that they did not warrant extra scrutiny because the single underlying parameter that both 1) governed the HEPs, and 2) was associated with potential impacts from EPU was the timing element PSF. As such, this parameter was sufficient in itself for the identification of HFEs requiring a detailed review. Therefore, these risk metrics were not used to identify HFEs requiring detailed review.

The time critical evolution of 45 minutes is not an actual time window (T_{sw}) or time available for action (T_{avail}), but rather an approximate, encompassing estimate of the time before which initial injection into the reactor pressure vessel needs to occur after reactor scram to avoid core damage. The 45-minute estimate was selected based on a review of the two sets of Modular Accident Analysis Program (MAAP) cases discussed in Section 4.2.4.1 of LAR Attachment 44 and performed to investigate the impacts of EPU on the timing PSF. That section of the LAR nominally addresses EPU impacts on the fire PRA (FPRA), but is also applicable to the internal events PRA. The value of 45 minutes was selected for the internal events PRA and the value of 40 minutes was selected for the FPRA. This difference is due to the fact that the internal events PRA analysis was performed by a different practitioner than the FPRA analysis, but it has no impact on the results, as further discussed in response to part (b) of this Request for Additional Information (RAI), which also addresses the technical basis for these time windows.

(b) All post-initiator HFEs of the FPRA were evaluated for potential impact from EPU. Section 4.2.4 of LAR Attachment 44 cites Fussell-Vesely as a risk importance metric used as a criterion for detailed review. While it is true that Fussell-Vesely was initially considered as an explicit criterion for detailed review, it was realized upon further investigation that the single underlying parameter that both 1) governed the HEPs, and 2) was associated with potential impacts from EPU was the timing element PSF. As such, this parameter was sufficient in itself for the identification of HFEs requiring a detailed review. Therefore, Fussell-Vesely was nominally used as a criterion but in practice did not provide additional insights. The same conclusion was reached with the risk achievement worth metric, although it was not mentioned in Section 4.2.4 of LAR Attachment 44 as an explicit criterion.

EPU conditions lead to shorter system time windows (T_{sw}), due to shorter time to boil-off resulting from increased power levels. Of particular importance is the time before which initial injection into the reactor pressure vessel needs to occur after reactor scram to avoid core damage. The MAAP cases discussed in Section 4.2.4.1 of LAR Attachment 44 indicate that all internal events and FPRA HFEs related to establishing initial injection have an assigned time window (T_{sw}) of less than 40 minutes (thereby indicating that operator action is required to take place in less than 40 minutes after reactor scram in order to avoid core damage). Therefore, for the fire HFEs, the encompassing 40-minute value was used as a discriminator for the time window of HFEs most susceptible to being impacted by EPU changes (i.e., those HFEs with a time window of 40 minutes or less). For the internal events HFEs, a value of 45 minutes was selected. The only difference between the 40 and 45 minute values is that the internal events PRA analysis was performed by a different practitioner than the FPRA analysis. Either 40 or 45 minutes could have been used for both analysis and the results would have been the same.

The HFEs with a time window greater than 40 minutes were also reviewed for potential EPU timing change impacts, but no required changes to the HEPs were identified.

APLA-RAI 04

In accordance with Appendix D of SRP, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

Section 4.6, "EPU Impact to Shutdown Risk," in Attachment 44 of the EPU LAR states that Reference 27 in Attachment 44 evaluates the impact on shutdown risk due to changes in the offsite alternating current (AC) recovery failure probabilities, and assessed a bounding ΔCDF of approximately 1 percent due to the EPU. Describe this bounding analysis. Also include in this discussion: how the offsite AC recovery failure probabilities were changed as a result of the EPU and how this analysis is bounding.

TVA Response:

Bounding Analysis:
The methodology used to determine the normalized increase in CDF due to EPU is summarized in Attachment F of Reference 27 in Attachment 44. The Attachment F analysis used decay heat calculations utilizing both CLTP and EPU thermal power to determine the time to boil-off from the top of active fuel to 2/3 core height for CLTP and EPU conditions, and then extrapolated the time to gap release based on decay heat level ratios by assuming that gap release occurs 0.5 hours after 2/3 core height is reached one day after shutdown. Gap release is the release of fission products in the fuel pin gap, which occurs immediately after failure of the fuel cladding and is the first radiological indication of core damage. The time to reach 2/3 core height through boil-off plus the time to gap release is the time to core damage. This approach is based on calculations performed by Sandia Laboratories and summarized in SECY-93-190. The differences in the time to core damage from normal Reactor Pressure Vessel (RPV) level, RPV flange level, and cavity flooded level were calculated. These estimates were then used to calculate the relative change in the offsite AC power recovery between CLTP and EPU conditions. Structural heat capacities such as vessel walls and through wall heat losses were conservatively not credited in this calculation. Crediting structural heat capacities and through wall losses would have the effect of lengthening the time to core damage, increasing the time available and likelihood of recovering offsite power and reducing the associated core damage risk. Therefore, this analysis is bounding.

Offsite Power Recovery Probability Analysis:
The offsite power recovery failure probabilities were determined using the data presented in Table 4-1 of NUREG/CR-6890. Table 4-1 shows offsite power recovery failure probability as a function of Loss of Offsite Power (LOOP) duration. The data for the composite (i.e., integrated data for plant, switchyard, grid, and weather related LOOP events) LOOP non-recovery curve for LOOP events experienced during shutdown conditions were used.

For example, the time to core damage on Day 1 of the outage was estimated to be 6.0 hours for CLTP and 5.3 hours for EPU based on Attachment F of Reference 27 in Attachment 44. The offsite AC power recovery failure probability is 8.64E-2 at 6.0 hours and approximately 9.62E-2 at 5.3 hours using a curve fit on the data presented in Table 4-1 of NUREG/CR-6890. The ratio of these recovery failure probabilities is approximately 1.11. For Day 2, when water is at the

RPV flange, this ratio was estimated at 1.13. For Days 3 through 29 with the refueling cavity flooded, the increase was judged to be negligible because of the long time period until core damage occurs (on the order of 89 hours for EPU conditions). For Day 30 with the water level back at normal levels, the ratio was estimated to be 1.13. These ratios are used to support the outage risk analysis below.

Outage Risk Analysis:
Because BFN does not have a shutdown (SD) PRA, the effect of EPU on BFN's outage risk was determined by a review of similar Boiling Water Reactors (BWRs) that do have a SD PRA. The percentage contribution to shutdown CDF of each distinct phase of an outage was determined by review of these shutdown PRAs. Four outage phases are modeled. The assumption in the BFN evaluation is that the weighted percentage of risk (or normalized risk) of each phase of an outage relative to other outage phases is equivalent to the relative risk of different outage phases for other BWRs (see column 3 of Table 5 from Reference 27 in EPU LAR Attachment 44 below). The LOOP/ Station Blackout (SBO) percentage contribution to CDF for each phase of the outage (column 4 of Table 5) was multiplied by the ratio of the EPU versus CLTP offsite AC power recovery probability (column 7 of Table 5) for each phase. The product represents the EPU risk increase of LOOP/SBO contribution. This is added to the non-LOOP/SBO contribution for the outage phase to derive the total contribution of the outage phase to the total outage risk for EPU (column 8 of Table 5). The non-LOOP/SBO risk is largely due to unintended inventory diversion and is unchanged by EPU. The results in column 8 were then summed in order to estimate a normalized increase in the overall CDF of approximately 1%. This is the same approach that was taken for the Peach Bottom and Monticello EPU submittals which were both approved by the NRC.

Table 5: Estimated Impact on Shutdown Risk Due to Offsite AC Recovery Failure Probability Increases Due to EPU

Outage Phase	Initial Water Level	Outage Phase Contribution to Overall SD CDF (CLTP)	LOOP/SBO Contribution to Outage Phase CDF	Time to Core Damage (hours)(1)		Factor Increase in Offsite AC Recovery Failure Probability (3)	Outage Phase Contribution to Overall SD CDF (EPU) (2)
				CLTP	EPU		
Day 1	Normal	0.1	0.3	6.0	5.3	1.11	0.103
Day 2	RPV Flange	0.15	0.3	13.3	11.7	1.13	0.156
Days 3-29 (use 16 days)	Flooded	0.7	0.1	101.3	88.8	negligible	0.700
Day 30	Normal	0.05	0.2	27.2	24.0	1.13	0.051
Normalized CDF (CLTP):		1.00	Normalized CDF (EPU):				1.01

(1) Calculated using Equation 22 of Attachment F of Reference 27 in Attachment 44
(2) Calculated as:
[(1 - 4th column) * 3rd column] + [4th column * 3rd column * 7th column]
The first contribution is the non-LOOP portion of the phase CDF (the portion unaffected by the change in offsite AC recovery failure probabilities). The second contribution is the LOOP portion of the phase CDF (the portion impacted by changes in offsite AC recovery failure probabilities). Example for Day 1: [(1- 0.3) * 0.1] + [0.3 * 0.1 * 1.11] = 0.07 + 0.033 = 0.103
(3) The factor increase in offsite AC recovery failure probability was calculated using the integrated AC recovery failure probability data for shutdown conditions tabulated in Table 4-1 of NUREG/CR-6890. A curve fit to the tabulated data was used to estimate the failure probabilities at intermediate values. For example, at t=6 hours the AC recovery failure probability is 8.64E-2 and at t=5.3 hours the AC recovery failure probability is 9.62E-2 (a factor of 1.11 higher).

APLA-RAI 05

In accordance with Appendix D of SRP, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

Section 5.4, "Seismic Risk," in Attachment 44 of the EPU LAR states that a conservative bounding analysis was applied to estimate an upper value for seismic CDFs. Describe this bounding analysis. Also include in this discussion:

a. Specific values applied to estimate the seismic CDFs;
b. Whether modifications to the containment/drywell/suppression pool structures as a result of the EPU are considered; and
c. How this analysis is bounding.

TVA Response:

The values used for the seismic CDFs were obtained from Section W.2.3 ("Total CDF and LERF") of the BFN NFPA 805 submittal. The BFN NFPA 805 submittal Section W.2.3 used estimates shown in Appendix D (ML100270756, "Seismic Core Damage Frequencies") of ML100270598 ("Safety/Risk Assessment Results for Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants") to determine the seismic core damage frequency estimates for BFN. The upper bound estimates reported in Attachment 44 of the EPU LAR are the estimates for BFN provided in the "weakest link model" column of Table D-1 of ML100270756. These weakest link model values are the highest CDF reported among those seismic CDF models listed for BFN in Table D-1. Therefore the seismic CDF is bounding. The estimates of CDF are shown below.

BFN Unit 1: 3.7E-06/yr
BFN Unit 2: 5.4E-06/yr
BFN Unit 3: 5.4E-06/yr

There are two EPU-related modifications to the containment/drywell/suppression pool structures. Because there is no seismic PRA for BFN, these modifications were evaluated on a qualitative basis and are judged not to affect the seismic CDF values quoted above. One modification involved adding a reinforcing pad on the Emergency Core Cooling System (ECCS) ring header on Unit 2 and Unit 3 to improve thermal stress protection. Unit 1 did not require this modification. This change has an insignificant impact on the deadweight and seismic analysis of the ECCS ring header. The total weight of the new pad (less than 20 pounds) is insignificant when compared to the total distributed weight of the 30" ring header (approximately 444 lbs/ft). The modification enables the pipe stresses at this location to meet American Society of Mechanical Engineers (ASME) code stress limits. The second modification involved modifying an existing pipe support on the Unit 2 Main Steam piping to accommodate increased dynamic loads on the piping system during turbine stop valve closure under EPU conditions. Unit 1 and Unit 3 did not require this modification. This modification meets the qualifications to be considered a Seismic Category I structure and does not affect the seismic CDF of Unit 2.

ENCLOSURE

APLA-RAI 06

In accordance with Appendix D of SRP, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

Based on the discussion in Sections 4.1.6, "Human Error Probabilities," and 5.7.4, "Uncertainties," in Attachment 44 of the EPU LAR, the minimum joint HEP (JHEP) of 1.0E-7 was retained in the internal events PRA for comparison of results between the current licensed thermal power (CLTP) and EPU cases and because the large number of independent operator errors and associated combinations skew the results in a conservative direction. The results of sensitivity analyses on JHEPs were presented in Section 5.7.1.7, "Sensitivity for Minimum JHEP ("Floor Values")," of EPU LAR Attachment 44 using minimum JHEP values of 0 and 1E-06. These sensitivity analyses showed that the risk results are sensitive to the number of JHEPs and the minimum JHEP assumed. The following observations were made in regards to the licensee's treatment of minimum JHEP in the internal events PRA:

- *RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," states: "the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative assumptions (i.e., change generally remains in the appropriate region)." The ΔCDFs and ΔLERFs from the sensitivity analyses in Tables 5-26 and 5-27 in Attachment 44 of the EPU LAR are relative to the EPU base case and not relative to the CLTP base case. As such, the staff could not confirm that the results from these sensitivity analyses still meet the risk acceptance guidelines of RG 1.174, Revision 2.*

- *Guidance in NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," recommends JHEP values should not be below 1E-05, because "it is typically hard to defend that other dependent failure modes that are not usually treated ... cannot occur." Table 4-4 of Electric Power Research Institute (EPRI) 1021081, "Establishing Minimum Acceptable Values for Probabilities of Human Failure Events," provides a lower limiting value of 1E-06 for sequences with a "very low" level of dependence. This issue was also raised in F&O 4-21 from the internal events PRA peer review listed in Table A-1 in Attachment 44 of the EPU LAR. The NRC staff notes that underestimation of minimum JHEP could result in non-conservative risk results (i.e., CDF, LERF, ΔCDF, and ΔLERF).*

- *Furthermore, the staff has considered the licensee's response to NFPA 805 PRA RAIs 01.v and 24 (ADAMS Accession No. ML14363A057 (dated December 17, 2014)). In this response, the licensee addressed the minimum JHEP in the FPRA by updating the FPRA to apply a floor value of 1E-05 to all HEP combinations that do not include long-term decay heat removal (DHR) HFEs or those HFEs cued and guided by Severe Accident Mitigation Guideline (SAMG) procedures. For the remaining combinations, a floor value of 1.0E-06 was applied given that a low dependency exists between long-term DHR and SAMG actions and other earlier actions. The NRC staff concluded in its safety evaluation dated October 28, 2015 (ADAMS Accession No. ML15212A796) that the fire PRA included an acceptable minimum JHEP value; however, these changes were not incorporated in the internal events PRA.*

Given the issues identified above regarding the licensee's treatment of minimum JHEP in the internal events PRA, provide one of the following:

(a) For JHEP values below 1.0E-05 used in the PRA for the sensitivity analyses in Table 5-27 of LAR Attachment 44 (which uses the 1E-06 floor), provide a detailed justification(s) for why use of the NUREG-1792 lower value guideline (i.e., minimum JHEP value of 1E-05) is inapplicable for this application. In this discussion, also include: an estimate of the number of these JHEPs below 1.0E-05, and provide at least two different examples where your justification is applied. For those JHEP values where inapplicability of the NUREG-1792 lower value guideline cannot be justified, explain how underestimating those JHEPs (e.g., using a floor of 1E-06 rather than a floor of 1E-05) impacts the risk results (i.e., CDF, LERF, ΔCDF, and ΔLERF) of this application. In addition, provide the CDF, LERF, ∆CDF and ∆LERF between the EPU and CLTP cases using the 1E-06 floor, and confirm that these results, combined with the risks from other hazards (i.e., fires, seismic, shutdown, and other external events) still meet the risk acceptance guidelines of RG 1.174, Revision 2. If RG 1.174 risk acceptance guidelines are exceeded, then please provide a detailed justification to support the conclusion that no "special circumstances" are created by the proposed EPU, include a discussion of which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

(b) Alternatively, provide the internal events PRA results between the EPU and CLTP cases (i.e., CDF, LERF, ΔCDF, and ΔLERF) where both use a minimum JHEP value of 1E-05 (or use a minimum JHEP consistent with the approach used for the licensee's FPRA) and confirm that these results, combined with the risks from other hazards (i.e., fires, seismic, shutdown, and other external events) still meet the risk acceptance guidelines of RG 1.174, Revision 2. If RG 1.174 risk acceptance guidelines are exceeded, then please provide a detailed justification to support the conclusion that no "special circumstances" are created by the proposed EPU. Include a discussion of which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

TVA Response:

In accordance with option (b) in this RAI, a combined sensitivity study for the internal events PRA was performed for the response to APLA-RAI 08 that is consistent with the joint human error probability approach used for the Fire PRA. The results of the combined sensitivity study and the comparison against the RG 1.174 Revision 2, risk acceptance guidelines were provided as part of the response to APLA-RAI 08.

APLA-RAI 07

In accordance with Appendix D of SRP, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

RG 1.174, Revision 2, states: "the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative assumptions." Section 5.7.1.4, "Sensitivity for Impact to Transient Initiators," and Section 5.7.1.6, "Sensitivity for Impact to LOCA [Loss-of-Coolant Accident] Frequencies," in Attachment 44 of the EPU LAR provides the results of various sensitivity analyses. However, the ΔCDF and ΔLERF results from these sensitivity analyses are relative to the "base case EPU [internal events] PRA" rather than the base case CLTP PRA, which is needed to confirm that the risk acceptance guidelines of RG 1.174 are still met. Based on a comparison of the results between the sensitivity cases in Sections 5.7.1.4 and 5.7.1.6 and the associated CLTP base cases in Table 5-15 in Attachment 44, the total ΔCDF and ΔLERF (for internal events and fires) may exceed the risk acceptance guidelines of RG 1.174 for some sensitivity cases (e.g., total ΔLERF for loss of feedwater in Units 1 and 2; total ΔLERF for loss of condenser vacuum, total and partial loss of condensate in Units 1, 2 and 3; total ΔLERF for excessive feedwater flow in Units 1 and 2; total ΔLERF for inadvertent Main Steam Isolation Valve closure in Units 1 and 2; total ΔLERF for LOCA in Unit 1).

For the sensitivity analyses in Sections 5.7.1.4 and 5.7.1.6 in Attachment 44 of the EPU LAR, provide the ΔCDF and ΔLERF relative to the CLTP base case, and confirm that the results of these sensitivity analyses combined with the risks from other hazards (i.e., fires, seismic, shutdown, and other external events) still meet the risk acceptance guidelines of RG 1.174, Revision 2. If RG 1.174 risk acceptance guidelines are exceeded, then please provide a detailed justification to support the conclusion that no "special circumstances" are created by the proposed EPU, include a discussion of which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

TVA Response:

The intent of the sensitivity studies with increased initiating-event frequencies is to investigate the sensitivities to conservatively postulated increases in initiating-event frequency. Following precedent from an approved EPU LAR for a similar BWR, each sensitivity study, with the exception of the Loss-of-Coolant Accident (LOCA) study, postulates an increase in initiating frequency by one event per year in the first year and by 0.5 events per year in the second year, and no increase for the remaining years of a long-term data period of ten years. Each sensitivity study uses the average over ten years of the initiating event frequencies thus postulated. Again following precedent from the similar BWR's EPU LAR, the LOCA sensitivity study conservatively doubles the LOCA initiating event frequencies for the small, medium, and large LOCA categories and for feedwater high energy line breaks. As discussed further below in this response, the resulting increased initiating event frequencies are much greater than would actually be experienced due to the EPU. In the case of the similar BWR's EPU, the NRC found in its Safety Evaluation (SE) for the similar BWR's EPU (ML14133A046) that "there are no issues with the evaluation of internal initiating event frequencies associated with the PBAPS internal events PRA that would rebut the presumption of adequate protection or warrant denial of this license amendment."

Table 7-1 provides the change in Core Damage Frequency (ΔCDF) and the change in Large Early Release Frequency (ΔLERF) relative to the CLTP base case for the sensitivity studies in Sections 5.7.1.4 and 5.7.1.6 in Attachment 44 of the EPU LAR. As can be seen in the "Total Sens CDF" and "Total ΔCDF" columns of Table 7-1, the total sensitivity CDF, which includes contributions from internal events, fire, seismic, and other external events, is always less than 1E-4 and total ΔCDF is always less than 1E-5. Therefore, despite the known conservatism of the sensitivity studies, the RG 1.174 risk acceptance guidelines are met for CDF for all of the sensitivity studies listed. However, in the "Total ΔLERF" column of Table 7-1, ΔLERF exceeds 1E-6 in some cases. That is, the RG 1.174 risk acceptance guideline for ΔLERF is exceeded for the following sensitivity studies for one or more plant units (Table 7-1).

- Loss of Feedwater
- Loss of Condenser/Condensate
- Excessive Feedwater Flow
- Inadvertent Main Steam Isolation Valve (MSIV) Closure
- Impact to LOCA Frequencies

The percentage increases in initiating event frequencies for the sensitivity studies listed above are provided in Table 7-2. Note that the postulated increases range from about 100% to 2000%. These increases are demonstrated by industry experience to be far in excess of any increases that may be experienced as a result of the EPU. To investigate the effect of EPUs on initiating event frequencies, members of the BWR Owners' Group that have implemented EPUs were surveyed. The survey covered the following topics: (1) occurrences of specific initiating events addressed in the sensitivity studies, as well as initiating events in general; and (2) whether the plant had experienced changes in initiating event frequencies before and after the EPU. The responses from survey participants indicate that the postulated increases in initiating-event frequencies for the sensitivity studies are very conservative and do not reflect realistic expectations of increased frequency of plant trips due to the EPU. In addition, the latest annual update on initiating events at US nuclear power plants, Idaho National Laboratory (INL) report INL/EXT-14-31428 Revision 1, "Initiating Event Rates at U.S. Nuclear Power Plants, 1988-2013," indicates that frequencies of initiating events at US nuclear power plants do not show statistically significant trends over the past ten years, with the exception of loss of feedwater, which shows a highly significant decrease in frequency. Any increases in initiating event frequencies that might be due to implementation of EPUs in the past ten years are not significant enough to be detected in the trends reported.

Because the increases in initiating event frequencies postulated for the sensitivity studies are much greater than realistically expected increases, the results of the sensitivity studies (presented in Table 7-1) should not be interpreted as indicating that "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements.

Further information related to the subject of this RAI is provided in response to APLA-RAI 08, which describes a combined sensitivity study that incorporates realistic increases in initiating-event frequencies.

Table 7-1. Delta Risk Results for Internal Events Sensitivity Cases

Base Case and Fire Risk Data	Unit	CDF	LERF	ΔCDF	ΔLERF	Total ΔCDF	Total ΔLERF	Total Sens CDF	Total Sens LERF
Total CLTP Risk	1	5.91E-05	7.96E-06	--	--	--	--	--	--
Total CLTP Risk	2	5.96E-05	7.99E-06	--	--	--	--	--	--
Total CLTP Risk	3	6.47E-05	7.18E-06	--	--	--	--	--	--
Fire ΔRisk (EPU-CLTP)	1	--	--	1.2E-06	6.0E-07	--	--	--	--
Fire ΔRisk (EPU-CLTP)	2	--	--	1.2E-06	5.0E-07	--	--	--	--
Fire ΔRisk (EPU-CLTP)	3	--	--	1.2E-06	3.9E-07	--	--	--	--
Internal Events CLTP Base	1	6.63E-06	1.44E-06	--	--	--	--	--	--
Internal Events CLTP Base	2	5.96E-06	1.39E-06	--	--	--	--	--	--
Internal Events CLTP Base	3	6.57E-06	1.38E-06	--	--	--	--	--	--
Internal Events Sensitivity Case									
Turbine Trip	1	7.17E-06	1.65E-06	5.4E-07	2.1E-07	1.7E-06	8.1E-07	6.1E-05	8.8E-06
Turbine Trip	2	6.50E-06	1.59E-06	5.4E-07	2.0E-07	1.8E-06	7.0E-07	6.1E-05	8.7E-06
Turbine Trip	3	7.07E-06	1.54E-06	5.0E-07	1.6E-07	1.7E-06	5.5E-07	6.6E-05	7.7E-06
Loss of Feedwater	1	8.58E-06	1.98E-06	2.0E-06	5.4E-07	3.1E-06	1.1E-06	6.2E-05	9.1E-06
Loss of Feedwater	2	7.91E-06	2.29E-06	2.0E-06	9.0E-07	3.2E-06	1.4E-06	6.3E-05	9.4E-06
Loss of Feedwater	3	8.29E-06	1.85E-06	1.7E-06	4.7E-07	2.9E-06	8.6E-07	6.8E-05	8.0E-06
Loss of Condenser/Condensate	1	1.10E-05	2.48E-06	4.4E-06	1.0E-06	5.6E-06	1.6E-06	6.5E-05	9.6E-06
Loss of Condenser/Condensate	2	1.03E-05	2.46E-06	4.3E-06	1.1E-06	5.6E-06	1.6E-06	6.5E-05	9.6E-06
Loss of Condenser/Condensate	3	1.04E-05	2.33E-06	3.8E-06	9.5E-07	5.0E-06	1.3E-06	7.0E-05	8.5E-06
RCS High Pressure Trip	1	7.16E-06	1.65E-06	5.3E-07	2.1E-07	1.7E-06	8.1E-07	6.1E-05	8.8E-06
RCS High Pressure Trip	2	6.49E-06	1.59E-06	5.3E-07	2.0E-07	1.8E-06	7.0E-07	6.1E-05	8.7E-06
RCS High Pressure Trip	3	7.06E-06	1.54E-06	4.9E-07	1.6E-07	1.7E-06	5.5E-07	6.6E-05	7.7E-06
Excessive Feedwater Flow	1	8.20E-06	1.96E-06	1.6E-06	5.2E-07	2.8E-06	1.1E-06	6.2E-05	9.1E-06
Excessive Feedwater Flow	2	7.53E-06	1.90E-06	1.6E-06	5.1E-07	2.8E-06	1.0E-06	6.2E-05	9.0E-06
Excessive Feedwater Flow	3	7.99E-06	1.84E-06	1.4E-06	4.6E-07	2.6E-06	8.5E-07	6.7E-05	8.0E-06

Table 7-1. Delta Risk Results for Internal Events Sensitivity Cases

Base Case and Fire Risk Data	Unit	CDF	LERF	ΔCDF	ΔLERF	Total ΔCDF	Total ΔLERF	Total Sens CDF	Total Sens LERF
Inadvertent MSIV Closure	1	8.55E-06	1.97E-06	1.9E-06	5.3E-07	3.1E-06	1.1E-06	6.2E-05	9.1E-06
Inadvertent MSIV Closure	2	7.88E-06	1.92E-06	1.9E-06	5.3E-07	3.2E-06	1.0E-06	6.3E-05	9.0E-06
Inadvertent MSIV Closure	3	8.26E-06	1.84E-06	1.7E-06	4.6E-07	2.9E-06	8.5E-07	6.8E-05	8.0E-06
Impact to LOCA Frequencies	1	7.40E-06	1.86E-06	7.7E-07	4.2E-07	2.0E-06	1.0E-06	6.1E-05	9.0E-06
Impact to LOCA Frequencies	2	6.84E-06	1.80E-06	8.8E-07	4.1E-07	2.1E-06	9.1E-07	6.2E-05	8.9E-06
Impact to LOCA Frequencies	3	7.27E-06	1.75E-06	7.0E-07	3.7E-07	1.9E-06	7.6E-07	6.7E-05	7.9E-06

Table 7-2. Frequency Increases Used in the Initiating Event Sensitivity Studies

Sensitivity Study	Initiating Event ID	CLTP Frequency	Sensitivity Frequency	Frequency Increase
Loss of Feedwater	%TLFW	9.38E-02	2.44E-01	160%
	%PLFW	9.21E-02	2.42E-01	163%
Loss of Condenser/Condensate	%LCV	1.03E-01	2.53E-01	146%
	%TLCF	7.48E-03	1.58E-01	2012%
	%PLCF	1.74E-02	1.67E-01	860%
Excessive Feedwater Flow	%EXFW	4.74E-02	1.97E-01	316%
Inadvertent MSIV Closure.	%IMSIV	7.21E-02	2.22E-01	208%
Impact to LOCA Frequencies	Various	Various	Various	100%

APLA-RAI 08

In accordance with Appendix D of SRP, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

Section 2.5.5, "Comparisons with Acceptance Guidelines," of RG 1.174, Revision 2, states that "the appropriate numerical measures to use in the initial comparison of the PRA results to the acceptance guidelines are mean values." However, the licensee compared point estimate values against the risk acceptance guidelines of RG 1.174. The licensee did perform a parametric uncertainty assessment in Section A.3, "Parametric, Model, and Completeness Uncertainties," of LAR Attachment 44 to estimate the mean risk values for this application and stated:

> *The results of these analyses show that the mean results and point estimates are approximately equal … Therefore, the point estimates are suitable for calculating delta risks. The conclusions drawn when comparing the point estimates and the propagated means with the RG 1.174 acceptance guidelines would not be different.*

Based on the staff's observations, point estimates may not always be suitable for calculating the total risk and the change in risk for this application. For example, the Unit 1 mean total ΔLERF (i.e., 2.34E-07 + 8.15E-07 = 1.05E-06, from Tables A-3 and A-4 of LAR Attachment 44) exceeds the risk acceptance guidelines of RG 1.174. While the Unit 1 point estimate total ΔLERF (i.e., 1.70E-07 + 6.04E-07 = 7.74E-07, from Tables A-3 and A-4 of LAR Attachment 44) meets these acceptance guidelines. Another example, the estimated total LERF and ΔLERF values for the EPU baseline case and the associated sensitivity analyses, are close to or exceeds the RG 1.174 acceptance guidelines as pointed out in the previous RAIs; therefore, the use of mean risk values may be more appropriate in these cases.

Additionally, Section 1.2.10, "Interpretation of Results Technical Elements," of RG 1.200, Revision 2, states:

> *The sensitivity of the model results to model boundary conditions and other assumptions is evaluated, using sensitivity analyses to look at assumptions both individually and in logical combinations. The combinations analyzed are chosen to account for interactions among the variables. NUREG-1855 provides guidance on the treatment of uncertainties associated with PRA.*

However, the sensitivity analyses in Section 5.7.1, "Internal Events PRA Sensitivity Analyses," in Attachment 44 of the EPU LAR did not consider combined effects from the impact of EPU conditions.

The NRC staff requests that the licensee perform a sensitivity analysis(es) using the internal events PRA that considers the combined effects from the impact of EPU conditions (e.g., combined impacts from sensitivity studies in Sections 5.7.1.4 and 5.7.1.6 of LAR Attachment 44). This sensitivity analysis(es) should also take into consideration: (1) the use of minimum JHEP values in response to PRA RAI 06, and (2) the assessment of mean risk values where the state-of-knowledge correlation can be important (e.g., in the assessment of LERF, ∆LERF).

Describe this sensitivity analysis(es); provide the associated risk results (i.e., CDF, LERF, ΔCDF and ΔLERF); and confirm that these results combined with the risks from other hazards (i.e., fires, seismic, shutdown, and other external events) still meet the risk acceptance guidelines of RG 1.174, Revision 2. (Note: the ΔCDF and ΔLERF should be relative to the CLTP base case, and the approach used to address minimum JHEP is expected to be similar for the EPU and CLTP cases, since this is a modeling attribute that impacts both cases.) If RG 1.174 risk acceptance guidelines are exceeded, then please provide a detailed justification to support the conclusion that no "special circumstances" are created by the proposed EPU, include a discussion of which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms

TVA Response:

This response describes a sensitivity analysis using the internal events PRA that considers the combined effects from the impact of EPU conditions (i.e., combined impacts from sensitivity studies in Sections 5.7.1.4 and 5.7.1.6 of LAR Attachment 44). This sensitivity analysis also includes: (1) the use of minimum joint human error probability (JHEP) values in response to APLA-RAI 06, and (2) the assessment of mean risk values for both the internal events risk and the fire risk (note that for seismic and other risks, no mean values exist, therefore the available point estimates are used as proxy for the mean risk). As discussed below, these results combined with the risks from other hazards still meet the risk acceptance guidelines of RG 1.174, Revision 2. The change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF) are computed relative to the CLTP base case, and the approach used to address minimum JHEP is similar for the EPU and CLTP cases.

As noted in the response to APLA-RAI 07, further information regarding sensitivity to postulated increases in initiating event frequencies is provided in this response. With reference to the survey of operational experience described in the response to APLA-RAI 07, it is noted that the operational experience reported by survey respondents shows that the original sensitivity studies reported in Attachment 44 of the EPU LAR are very conservative. A more realistic set of increases in initiating event frequencies is postulated in the present sensitivity study, consistent with the survey of operational experience discussed in the response to APLA-RAI 07. For each affected initiating event (that is, those included in the sensitivity studies reported in Sections 5.7.1.4 and 5.7.1.6 of LAR Attachment 44), the current sensitivity study postulates a 100% increase in initiating frequency in the first year, a 50% increase in the second year, and no increase for the remaining years of a long-term data period of 10 years. For the sensitivity study performed for the response to this RAI, each initiating event frequency is averaged over a long-term data period of 10 years. That is, in effect, each affected initiating event is increased by 15% for the sensitivity study.

As noted in the response to APLA-RAI 06, the combined sensitivity study for the internal events PRA presented in this response (to APLA-RAI 08) uses minimum JHEPs consistent with the approach used for the BFN FPRA. In addition to adopting the BFN FPRA approach to the JHEP floors, the present sensitivity study (that is, for APLA-RAI 08) removes excessive conservatisms of the internal events PRA by incorporating refinements in the dependency levels of some human failure event (HFE) combinations. In particular, the dependency level between HFEs that involve (1) failing to depressurize the reactor pressure vessel after core damage to recover the core in-vessel and (2) flooding primary containment was changed from complete to medium. These two HFEs are intended to reduce the LERF risk after core damage. In the initial evaluation of JHEPs, the dependency level between these two actions was conservatively

set to complete based on inadequate staffing resources. However, by the time these actions would take place, the Technical Support Center would be in place and sufficient personnel would be at the plant to carry out required actions. As a result, it was determined that this additional support would be sufficient to decrease the level of dependency between these two actions, which was accordingly reduced to medium. As noted in the RAI, the BFN Unit 1 mean total ΔLERF as calculated from Tables A-3 and A-4 of LAR Attachment 44 exceeds the risk acceptance guidelines of RG 1.174 (i.e., 2.34E-07/yr + 8.15E-07/yr = 1.05E-06/yr). In the combined sensitivity study reported in this response (see Table 8-1), the BFN Unit 1 ΔLERF value was calculated to be 9.00E-07/yr. The difference is due to the refinement noted above in which excessive conservatisms of the internal events PRA were removed in the dependency levels of some HFE combinations.

For the combined sensitivity study, the risk acceptance guidelines of RG 1.174 associated with Region II of that guide are met for total risk and delta risk (Table 8-1). For the EPU case, the total LERF of BFN Unit 1 reaches but does not exceed the acceptance guideline threshold of 1E-05/yr, and the total LERF of BFN Unit 2 is close to, but less than that threshold. These numerical estimates embed some significant conservatisms of the BFN internal events PRA models. Specifically, these models conservatively do not credit the Emergency High Pressure Makeup (EHPM) pump, which is to be installed as an NFPA 805 modification to provide an independent and diverse injection system to supplement the existing injection systems. If the EHPM pump were credited in the internal events model, the CDF and LERF would decrease significantly, revealing greater margins with respect to the acceptance guidelines of Region II.

Table 8-1. Summary Risk Results for the Combined Sensitivity Study

BFN Plant Unit	Risk Measure	Power Case	Mean Internal Events Risk (/ yr)	Mean Fire Risk (/ yr)	Seismic Risk (/ yr)	Other Risk (/ yr)	Mean Estimate for Total Risk (/ yr)
1	CDF	CLTP	1.91E-05	4.78E-05	3.7E-06	1.0E-06	7.16E-05
1	CDF	EPU	1.97E-05	4.86E-05	3.7E-06	1.0E-06	7.30E-05
1	CDF	∆CDF	6.04E-07	8.30E-07	0.0E+00	0.0E+00	1.43E-06
1	LERF	CLTP	2.73E-06	5.90E-06	3.7E-07	1.0E-07	9.10E-06
1	LERF	EPU	2.82E-06	6.71E-06	3.7E-07	1.0E-07	1.00E-05
1	LERF	∆LERF	8.48E-08	8.15E-07	0.0E+00	0.0E+00	9.00E-07
2	CDF	CLTP	1.78E-05	4.71E-05	5.4E-06	1.0E-06	7.13E-05
2	CDF	EPU	1.85E-05	4.83E-05	5.4E-06	1.0E-06	7.32E-05
2	CDF	∆CDF	6.24E-07	1.12E-06	0.0E+00	0.0E+00	1.74E-06
2	LERF	CLTP	2.64E-06	5.89E-06	5.4E-07	1.0E-07	9.17E-06
2	LERF	EPU	2.72E-06	6.51E-06	5.4E-07	1.0E-07	9.87E-06
2	LERF	∆LERF	7.34E-08	6.25E-07	0.0E+00	0.0E+00	6.98E-07
3	CDF	CLTP	1.59E-05	5.11E-05	5.4E-06	1.0E-06	7.34E-05
3	CDF	EPU	1.66E-05	5.35E-05	5.4E-06	1.0E-06	7.65E-05
3	CDF	∆CDF	7.29E-07	2.35E-06	0.0E+00	0.0E+00	3.08E-06
3	LERF	CLTP	2.37E-06	5.06E-06	5.4E-07	1.0E-07	8.07E-06
3	LERF	EPU	2.46E-06	5.55E-06	5.4E-07	1.0E-07	8.65E-06
3	LERF	∆LERF	8.30E-08	4.86E-07	0.0E+00	0.0E+00	5.69E-07

APLA-RAI 09

In accordance with Appendix D of SRP, Chapter 19.2, the NRC staff reviews the BFN EPU risk evaluations to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection from compliance with the regulations and other license requirements. Therefore, the licensee's risk evaluations need to be sufficient for the staff to conclude that the risk impact from internal events, external events, and shutdown operations is acceptable and does not create special circumstances.

As discussed in Attachment 44 of the EPU LAR, the change in risk results for this application is driven by the change in HEPs as a result of the EPU. This demonstrates the importance of calculating realistic changes in HEPs. In order to confirm the reasonableness of these calculations and to understand why the increase in some HEPs are much larger than others, the staff requests that the licensee explain how the following HEPs were quantified for both the EPU and CLTP cases, provide sufficient detail and numerical values to understand the basis for these HEPs:

- *HFA_0HCIINIT30, "Operator Fails to Initiate HPI (30 Min)," in the internal events PRA and listed in Table 4-4 of LAR Attachment 44.*
- *HFA_0002RPV_LVL, "Operator Fails to Maintain RPV Level," in the internal events PRA and listed in Table 4-4 of LAR Attachment 44.*
- *JHEP COMBINATION_1195, "HEP dependency factor for HFA_0HCIINIT30, HFA_0002RPV_LVL," in the internal events PRA and listed in Table 4-5 of LAR Attachment 44.*
- *HFFA0ASD_RCIC, "Operator Fails to Start RCIC," in the fire PRA and listed in Table 4-10 of LAR Attachment 44.*
- *HFFA0RHRCS_LPP, "Oper Fails to Bypass ECCS Low Pressure Permissive," in the FPRA and listed in Table 4-10 of LAR Attachment 44*

TVA Response:

In this response, for each Human Failure Event (HFE) discussed individually, dependency refers to inter-action dependencies within that HFE. Dependency between HFEs is reserved for the discussion of Joint Human Error Probability (JHEP) COMBINATION_1195. The Modular Accident Analysis Program (MAAP) analysis used below is discussed in Section 4.2.4.1 of LAR Attachment 44.

The only differences between the EPU and CLTP cases for HFEs HFA_0002RPV_LVL, HFA_0HCIINIT30, HFFA0RHRCS_LPP, and HFFA0ASD_RCIC are the System Time Windows (Tsw). The System Time Window (Tsw) as defined by the Electric Power Research Institute (EPRI) is the time from the start of the event until the action is no longer beneficial. Due to lower initial power levels for the CLTP case, there are approximately seven more minutes to accomplish the actions in the CLTP sequences than in the EPU sequences. Based on the MAAP analysis, injection can be delayed up to 42 minutes for the CLTP case, whereas it must be initiated within 35 minutes for the EPU case. Each HFE referenced in this RAI was analyzed

in the same manner, except for the system time window Tsw changes which resulted in a change in the dependency level as discussed below.

HFFA0RHRCS_LPP is used in the Fire Probabilistic Risk Assessment (FPRA). It models the override of the Residual Heat Removal (RHR) low pressure permissive in the event fire damage prevents that permissive. The MAAP analysis indicates that injection must occur within the Tsw of 35 minutes for EPU, but because the action to depressurize is modeled under an HFE distinct from HFFA0RHRCS_LPP, it was assumed (for both CLTP and EPU) that less than 35 minutes would actually be available to recognize the need for an RHR low pressure permissive override and perform it. Thus, HFFA0RHRCS_LPP was estimated to be performed within a 15-minute system time window for EPU. This time was shorter than the 35 minutes Tsw to allow additional actions necessary to establish injection. For the CLTP analysis, a Tsw of 22 min was used. The EPRI Human Reliability Analysis (HRA) calculator was used for the analysis. For both cases, the net effect of the system time window difference was to increase the dependency by one level for the EPU case. The CLTP case used a dependency level of medium and the EPU case used a dependency level of high. The EPRI HRA Calculator uses these dependency levels to adjust the values of the cognitive performance shaping factors and some execution step failure probabilities that are calculated by the HRA Calculator. The HRA Calculator computes a conditional probability based on the selected dependency level and multiplies this by the applicable performance shaping factor or execution step failure probability. This conditional probability is typically around 3.5 times greater for a high dependency than it is for a medium dependency. More detail of this process is found in NUREG/CR-1278, “Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications,” dated 1983. The Human Error Probability (HEP) for this action increased from 5.80E-03 for the CLTP case to 1.94E-02 for the EPU case.

HFE HFFA0ASD_RCIC is used in the FPRA; HFA_0002RPV_LVL and HFA_0HCIINIT30 are used in the Internal Events PRA. All three HFEs model establishing an initial injection source into the reactor vessel. Those sources are the Reactor Core Isolation Cooling (RCIC) System, High Pressure Coolant Injection (HPCI) System, and Condensate System. It should be noted that the non-abandonment FPRA model and the abandonment FPRA model utilize different HFE names for the initiation of RCIC. HFFA0ASD_RCIC is the RCIC initiation name used in the main control room abandonment FPRA model. For the EPU analysis of these HFEs, a Tsw of 35 minutes was used because MAAP indicates for these sequences injection must occur within 35 minutes. For the CLTP analysis of these HFEs, a Tsw of 42 minutes was used because MAAP indicates for these sequences injection must occur within 42 minutes. The EPRI HRA calculator was also used for the analysis. For both cases, the net effect of the seven minute system time window difference was to increase the dependency by one level for the EPU case. For HFFA0ASD_RCIC, the CLTP case used a dependency of medium and the EPU case used a dependency of high. For HFA_0002RPV_LVL and HFA_0HCIINIT30, the CLTP case used a dependency of low and the EPU case used a dependency of medium. The dependency levels are higher for the abandonment action due to the longer execution time and stress levels.

The HEP for HFFA0ASD_RCIC increased slightly from 2.99E-02 for the CLTP case to 3.45E-02 for the EPU case.

The HEP for HFA_0002RPV_LVL increased from 4.78E-04 for the CLTP case to 1.32E-03 for the EPU case.

The HEP for HFA_0HCIINIT30 increased from 1.50E-03 for the CLTP case to 3.06E-03 for the EPU case.

The JHEP COMBINATION_1195 represents the joint combination of the HFEs HFA_0HCIINIT30 and HFA_0002RPV_LVL. The combination involves the failure to initiate HPCI followed by a failure to establish condensate injection; both failures happen close in time to each other. For both the CLTP and EPU cases, a complete dependency is conservatively assumed between these actions. The initiation action HFA_0HCIINIT30 would occur first and the combination JHEP is, due to the complete dependency, the same as the HEP probability for HFA_0HCIINIT30. This HEP changes between the CLTP case and the EPU case as discussed above. That is, the JHEP for JHEP COMBINATION_1195 increased from 1.5E-03 for the CLTP case to 3.06E-03 for the EPU case.