~~Proprietary Information~~
~~Withhold from Public Disclosure Under 10 CFR 2.390~~
This letter is decontrolled when separated from Enclosure 2

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

CNL-16-048

March 24, 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 7, 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 February 25, 2016 (ML16041A022)

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 (RAIs) related to fire protection. The due date for the responses to the NRC RAIs provided by the Reference 2 letter is March 25, 2016. Enclosure 1 to this letter provides the responses to the RAIs included in the Reference 2 letter.

Enclosure 2 to this letter provides a supplement to the Power Uprate Safety Analysis Report (PUSAR) (NEDC-33860P, Revision 0) and the Fuel Uprate Safety Analysis Report (FUSAR) (ANP-3403P, Revision 2). AREVA Inc. (AREVA) and GE-Hitachi Nuclear Energy Americas LLC (GEH) consider portions of the information provided in Enclosure 2 of this letter to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390, Public inspections, exemptions, requests for withholding. An affidavit for withholding information, executed by AREVA, is provided in Enclosure 4. An affidavit for withholding information, executed by GEH, is provided in Enclosure 5. Enclosure 3 to this letter provides the non-proprietary versions of the PUSAR (NEDO-33860, Revision 0) and FUSAR (ANP-3403NP, Revision 2). Therefore, on behalf of AREVA and GEH, TVA requests that Enclosure 2 be withheld from public disclosure in accordance with the AREVA and GEH affidavits and the provisions of 10 CFR 2.390.

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, without the proprietary information, 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 Edward D. Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 24th day of March 2016.

Respectfully,

J. W. Shea

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

Enclosures

cc: See Page 3

Enclosures:

1. Response to NRC Requests for Additional Information AFPB-RAI 1, AFPB-RAI 2, AFPB-RAI 3, AFPB-RAI 4, AFPB-RAI 5, AFPB-RAI 6 and AFPB-RAI 7
2. Supplement to PUSAR (NEDC-33860P, Revision 0) and FUSAR (ANP-3403P, Revision 2) - (Proprietary version)
3. Supplement to PUSAR (NEDO-33860, Revision 0) and FUSAR (ANP-3403NP, Revision 2) - (Non-proprietary version)
4. AREVA Affidavit for ANP-3403P, Revision 2
5. GE Hitachi Nuclear Energy Affidavit for NEDC-33860P, Revision 0

cc:

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
State Health Officer, Alabama Department of Public Health (w/o Enclosure 2)

ENCLOSURE 1

Response to NRC Requests for Additional Information
AFPB-RAI 1, AFPB-RAI 2, AFPB-RAI 3, AFPB-RAI 4, AFPB-RAI 5, AFPB-RAI 6 and AFPB-RAI 7

ENCLOSURE 1

AFPB-RAI 1

The NRC staff notes that LAR Attachment 6 to the Safety Analysis Report (SAR) for BFN, Units 1, 2, and 3, Extended Power Uprate (EPU), NEDC-33860P,[1] Revision 0, September 2015, Section 2.5.1.4, "Fire Protection," states that, "The transition to NFPA 805 is currently under NRC staff review, and TVA anticipates its approval prior to implementation [of] EPU operation. Accordingly, the fire protection analysis described in this section is based on the NFPA 805 implementation…"

On October 10, 2015, the NRC issued a license amendment for BFN, Units 1, 2, and 3, to incorporate National Fire Protection Association (NFPA) 805 fire protection licensing basis in accordance with Title 10 of Code of Federal Regulations (10 CFR), Section 50.48(c). The amendment approved the transition of the licensee's fire protection program to a risk-informed, performance-based program based on the 2001 Edition of NFPA 805. The staff requests that the licensee provide a supplement to LAR Attachment 6 to the SAR for BFN Units 1, 2, and 3, EPU, NEDC-33860P, Revision 0, September 2015, Section 2.5.1.4, "Fire Protection," consistent with the approved NFPA 805 licensing basis.

TVA Response:

Enclosure 2 of this letter provides the requested supplement to license amendment request (LAR) Attachment 6 which is the Power Uprate Safety Analysis Report (PUSAR) and it also provides the conforming changes to LAR Attachment 8 which is the Fuel Uprate Safety Analysis Report (FUSAR). Enclosure 3 of this letter provides the conforming changes to the non-proprietary version of the PUSAR (LAR Attachment 7) and to the non-proprietary version of the FUSAR (LAR Attachment 9). The changes incorporate the NRC-issued license amendments (ADAMS Accession Number ML15212A796) approving the transition of Browns Ferry's licensing basis to the National Fire Protection Association (NFPA) 805 standard in accordance with 10 CFR 50.48(c). Enclosure 2 is proprietary as it provides a supplement to proprietary documents (LAR Attachments 6 and 8).

[1] Note: This document is marked "Proprietary" (non-publicly available information) in TVA letter dated September 21, 2015. All contents of this document should be handled in accordance with the NRC guidance of handling Sensitive Unclassified Non-Safeguards Information (SUNSI).

ENCLOSURE 1

AFPB-RAI 2

Attachment 1 to Matrix 5 "Supplemental Fire Protection Review Criteria, Plant Systems", of NRR RS-001, Revision 0, "Review Standard for Extended Power Uprates" (ADAMS Accession No. ML033640024) states:

> *Power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire. However, the licensee's application should confirm that these elements are not impacted by the extended power uprate.*

The NRC staff notes that LAR Attachment 6 to SAR for BFN, Units 1, 2, and 3, EPU, NEDC-33860P, Revision 0, September 2015, Section 2.5.1.4.1, "Fire Protection Program," specifically addresses items (1) through (4) above. The staff requests that the licensee provide statements to address item (5) and a statement confirming no increase in the potential for a radiological release resulting from a fire.

TVA Response:

I. Impact of Extended Power Uprate (EPU) on …(5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown (CSD).

TVA confirms that the proposed EPU has no impact on the BFN procedures and resources necessary for the repair of systems required to maintain the fuel in a safe and stable condition.

The NRC approved the EPU licensing basis for fire for Browns Ferry Nuclear Plant (BFN) in their Safety Evaluation (SE) for the BFN transition to the NFPA 805 (Reference 1). In the NRC SE, the following statements are made in Section 3.2.1.2:

"3.2.1.2 Attribute Alignment -- Aligns with Intent

For certain of the NEI [Nuclear Energy Institute] 00-01, Chapter 3 attributes, the licensee determined that the SSA [Safe Shutdown Analysis] aligns with the intent of the attribute and provided additional clarification when describing its means of alignment. The attributes identified in LAR [License Amendment Request] Attachment B, Table B-2 as having this condition are as follows:

- 3.1 Safe Shutdown Systems and Path Development
- 3.1.1.9 72-Hour Coping
- 3.1.2.4 Decay Heat Removal
- 3.4.1.5 Repairs

The licensee indicated that for the above-listed attributes, the NSCA [Nuclear Safety Capability Assessment] does not ensure CSD [cold shutdown] and does not specifically align with the guidance. The licensee further stated that NFPA 805 does not require a plant to transition to

CSD following a fire and that the NSCA and NFPA 805 only require maintaining the fuel in a safe and stable condition (i.e., there is no requirement to achieve and maintain CSD) and contains no 72-hour coping time. The NRC staff concludes that the methods as described by the licensee are acceptable because they meet the requirement of NFPA 805, which is to maintain the fuel in a safe and stable condition."

Element (5) in Attachment 1 to Matrix 5 of Nuclear Reactor Regulation (NRR) RS-001 concerning the cold shutdown requirement is not applicable for BFN at EPU conditions because BFN at EPU conditions will continue to comply with the approved NFPA 805 licensing basis for fire protection.

II. Impact of EPU on the potential for a radiological release resulting from a fire.

The increase in decay heat has no effect on engineered controls and administrative controls, pre-fire plans, or fire brigade response procedures and training procedures. Therefore, TVA confirms there is no increase, at EPU conditions, in the potential for a radiological release resulting from a fire.

From the NRC SE for the BFN transition to NFPA-805 (Reference 1), Radioactive Release Performance Criteria:

"3.6.8 Conclusion for Section 3.6

The NRC staff's evaluation is based on:

(1) Information and analyses provided in the LAR and RAI [Request for Additional Information] responses;
(2) Use of engineered controls and administrative controls to contain potential releases;
(3) Use of pre-fire plans, and
(4) Use of revised fire brigade response procedures and training procedures.

Based on the above, the NRC staff concludes that the licensee's RI/PB [Risk - Informed / Performance - Based] fire protection program provides reasonable assurance that radiation releases to any unrestricted area resulting from the direct effects of fire suppression activities are as low as reasonably achievable and are not likely to exceed the radiological release performance criteria of NFPA 805 and the radiological dose limits in 10 CFR 20."

Reference

1. U.S. Nuclear Regulatory Commission letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program In Accordance With 10 CFR 50.48(c) (CAC Nos. MF1185, MF1186, and MF1187)," dated October 28, 2015 (ML15212A796).

AFPB-RAI 3

The NRC staff notes that LAR Attachment 6 to the SAR for BFN, Units 1, 2, and 3, EPU, NEDC-33860P, Revision 0, September 2015, Section 2.5.1.4.1, "Fire Protection Program," states:

> *The higher decay heat associated with EPU results in higher heat input into the suppression pool which, without mitigation, will result in higher suppression pool temperatures. The higher decay heat may also result in lower vessel water levels or higher Peak Cladding Temperatures (PCTs), depending on the plant-specific analysis basis. As a result of these effects, fire suppression and detection systems, operator response time, peak clad temperature (PCT), and suppression pool temperature need to be addressed.*

The staff requests the licensee verify that additional heat in the plant environment from the EPU will not:

a. Impact any required operator manual actions (referred to as recovery actions per NFPA 805 licensing basis) being performed at their designated time (including, e.g., verifying that under EPU conditions, recovery actions and repairs required to demonstrate the availability of a success path to achieve the nuclear safety performance criteria are feasible and have been evaluated for the additional risk due to their use), or

b. Require any new recovery actions due to additional heat in the plant environment to maintain the plant in safe and stable conditions.

TVA Response:

The increases in maximum suppression pool temperature and in maximum drywell temperature from Current Licensed Thermal Power to EPU conditions, after an NFPA 805 fire event, are 2.3°F and <0.5°F respectively. These small increases in temperature would have minimal effect on the local area temperatures where the operator recovery actions are required to take place. Therefore, the additional heat load in the plant environment from the EPU will not (1) impact any required operator manual actions being performed at their designated time, or (2) require any new recovery actions, due to additional heat in the plant environment, to maintain the plant in safe and stable conditions.

AFPB-RAI 4

The staff notes that LAR Attachment 6 to the SAR for BFN, Units 1, 2, and 3, EPU, NEDC-33860P, Revision 0, September 2015, Section 2.5.1.4.1, "Fire Protection Program," states that, "…Other EPU modifications will be assessed and assured not to adversely affect the ability to achieve and maintain the fuel in a safe and stable condition in the event of a fire…" Further, Section 2.11.1.2.2, "Fire Safe Shutdown (FSS) Events," states:

> *Attachment 47 of the EPU LAR provides a listing and discussion of the modifications planned for EPU. The effect of these modifications on the Browns Ferry Fire Protection Program will be evaluated, in accordance with TVA's configuration change process, prior to EPU implementation. Per the process, these modifications will be evaluated to assure the changes do not affect the approved Fire Protection Program and will not adversely affect the ability to achieve and maintain safe shutdown in accordance with the current Browns Ferry license conditions and procedures…"*

The NRC staff notes that modifications associated with the EPU have not yet been completed to address the impact on the fire protection program. The staff requests that the licensee discuss how the results of plant modifications would impact the fire protection program and the plant's compliance with the fire protection program licensing basis (10 CFR 50.48(c)).

TVA Response:

All of the modifications listed in Attachment 47 of the EPU LAR have been reviewed for impact on the Fire Protection Program and the plant's compliance with the fire protection program licensing basis, 10 CFR 50.48(c).

Two modifications, which have not been completed, will result in a positive impact, i.e., plant risk improvement, on the 10 CFR 50.48(c) compliant fire protection program when they are implemented. These modifications are as follows: the Emergency High Pressure Make-Up Pump modifications and the Hardened Wetwell Vent modifications.

All other modifications listed in Attachment 47 of the EPU LAR do not have an adverse impact on the Fire Protection Program, including the plant's compliance with the fire protection program licensing basis, 10 CFR 50.48(c).

AFPB-RAI 5

The NRC staff notes that LAR Attachment 6 to the SAR for BFN, Units 1, 2, and 3, EPU, NEDC-33860P, Revision 0, September 2015, Section 2.5.1.4.1, "Fire Protection Program," states:

> *Original NFP 805 analyses were performed at EPU conditions and therefore operator action times cannot be compared to CLTP [current licensed thermal power] conditions. To ensure that PCT remains less than the acceptance criterion in the most limiting scenario, one LPCI pump must be manually aligned for injection within 20 minutes…*

The staff requests the licensee provide a technical justification for the 20-minute time for the operator to perform the actions to align LPCI pump injection. Include discussion of the original time margin without CLTP conditions (i.e., what was assumed for the NFPA 805 analyses) and why the 20-minute assumption is deemed adequate.

TVA Response:

As stated in Section 2.5.1.4.1 of Attachment 6 (PUSAR) to the EPU LAR, the 20-minute operator action time to align Low Pressure Coolant Injection (LPCI) for injection is unchanged from the 20-minute operator action time assumed in the Browns Ferry Nuclear Plant NFPA 805 LAR (Reference 1). TVA Calculation MDQ0009992012000094, "NFPA-805 NPSH, Containment Parameters, and Areva Fuel PCT Analysis," Revision 1, dated March 14, 2013 (provided as Reference 6 of Attachment X in the NFPA 805 LAR) contains the analysis for fuel Peak Clad Temperature (PCT) that includes the 20-minute operator action time for aligning LPCI. The fuel PCT analysis contained in TVA calculation MDQ0009992012000094 demonstrates that the fuel is maintained in a safe and stable condition when spurious openings of multiple Main Steam Relief Valves are assumed.

The technical justification for the 20-minute operator action time for aligning LPCI is that the required alignment can be performed by the operators for all fire scenarios in less time than assumed in the design analysis. TVA Calculation MDQ0009992012000108, "NFPA 805 Operator Action Feasibility Analysis," (Reference 6.39 of the NFPA 805 LAR) documents the results of TVA demonstrations to confirm that required NFPA 805 operator actions can be performed within the required time constraints assumed in the design analyses. For the 20-minute operator action time for aligning LPCI, the maximum time to perform LPCI alignment was 14 minutes, which is less than the 20-minute required action time. Therefore, the 20-minute operator action time is justified.

TVA has subsequently re-validated the 20-minute operator action time for LPCI alignment as part of the NFPA 805 implementation activities. This revalidation was performed using the Fire Safe Shutdown (FSS) procedures developed as part of the NFPA 805 implementation. The maximum time for operators to complete LPCI alignment using the FSS procedures was 15 minutes for all three BFN Units.

Reference

1. TVA letter to U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant, Units 1, 2, and 3, License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specification Change TS-480)," dated March 27, 2013 (ML13092A393).

AFPB-RAI 6

The NRC staff notes that LAR Attachment 6 to the SAR for BFN, Units 1, 2, and 3, EPU, NEDC-33860P, Revision 0, September 2015, Section 2.5.1.4.2, "Fire Event," states:

> *The results of Case 4, and the evaluations in Section 2.6.5.2, FUSAR [Fuel Uprate Safety Analysis Report] Section 2.5.1.4, and LAR Attachment 39, demonstrate that the peak fuel cladding temperature, vessel water level, and suppression pool temperature meet the acceptance criteria and the time available for the operators to perform the necessary actions is sufficient. Therefore, EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of a fire event and satisfies the requirement of achieving and maintaining the fuel in a safe and stable condition in the event of a fire.*

The staff requests the licensee confirm that the above analysis cases at EPU conditions meet the NFPA 805 licensing basis to achieve and maintain the fuel in a safe and stable condition in the event of a fire.

TVA Response:

TVA confirms that the above analyses at EPU conditions meet the NFPA 805 licensing basis to achieve and maintain the fuel in a safe and stable condition in the event of a fire.

AFPB-RAI 7

Some plants credit aspects of their fire protection system for other than fire protection activities (e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems). If BFN, Units 1, 2, and 3, credits its fire protection system in this way, identify the specific situations and discuss to what extent, if any, the EPU affects these "non-fire-protection" aspects of the plant fire protection system.

TVA Response:

The Browns Ferry Nuclear Plant (BFN) does not credit the Fire Protection System (FPS) for non-fire protection activities in any Design Basis scenario. However, the BFN symptom-based Emergency Operating Instructions (EOIs) and Severe Accident Management Guidelines (SAMGs) provide operator direction for potential use of the FPS for the following non-fire suppression functions during beyond design basis events:

1. Reactor Pressure Vessel injection when preferred water sources are not available;
2. Makeup to the Spent Fuel Storage Pool(s) when preferred water sources are not available;
3. Injection to Primary Containment for pressure suppression and/or radiological release scrubbing when preferred water sources are not available; and
4. External makeup to the condensate system hotwell in order to use the condensate/feedwater system for Reactor Pressure Vessel makeup.

These potential uses are part of the current BFN configuration, are not affected by EPU, and are consistent with industry-accepted guidelines.

For BFN, the high pressure fire protection (HPFP) system is a combined system with the Raw Service Water (RSW) system. Attachment A to Reference 1 under topic "NFPA 805 Ch. 3 Reference 3.5.16 [Water Supply Dedicated Limits]" documents NRC prior approval of this arrangement.

Under normal operating conditions, the fire pumps are in standby. The RSW pumps and the two RSW head tanks maintain the HPFP headers pressurized and provide water service to components supplied by the RSW system. Under abnormal conditions where either the RSW pumps are manually secured, the RSW head tanks are isolated or the RSW pumps trip, a fire pump is started to maintain the HPFP system headers pressurized. The started fire pump will also provide water to the loads normally supplied by RSW which include charging water to the Emergency Equipment Cooling Water System (south header), charging water to the Residual Heat Removal Service Water System, Circulating Water (CCW) pump motor cooling and bearing lubrication water, seal water to waste transfer pumps and CCW vacuum pumps, Cooling Tower lift pump bearing lubrication water and motor cooling water, and cooling water to Off-Gas glycol units. The below table provides the abnormal conditions that would result in loss of RSW and the starting of fire protection pumps. As stated in Section 2.5.3.2.2 of EPU LAR Attachment 6, there are no power dependent loads on the RSW system and therefore there are no heat load increases due to EPU. EPU does not affect the current TVA use of the HPFP system under abnormal conditions where there is a loss of the RSW pumps or isolation of the RSW head tanks.

Abnormal Condition	Action Taken	Discussion
Degraded Raw Cooling Water (RCW) Capability	A fire pump is started and RSW pumps are manually secured.	The RSW pumps take suction from the RCW suction header downstream of the RCW suction strainers. The RSW pumps are secured to reduce the load on the RCW suction strainers, which may be the cause of the RCW capability degradation.
Loss of Offsite Power	Diesel fire pump is started.	RSW pumps trip due to loss of offsite power.
Loss of Plant Preferred (power)	RSW head tanks isolate due to loss of Plant Preferred. A fire pump is started.	Per design, starting of a fire pump results in tripping of running RSW pumps.
Loss of Control Air	RSW head tanks isolate due to loss of control air. A fire pump is started.	Per design, starting of a fire pump results in tripping of running RSW pumps.

In addition, TVA currently uses a fire pump to supplement the RSW system when the RSW system cannot meet the total demand. Challenges to RSW demand have occurred when the BFN cooling towers are in service, which have resulted in inadequate flow and pressure to the bearing lubricated water supply for the cooling tower lift pumps. In Reference 2 (TVA response to FPE-RAI 04) and Enclosure 3 to Reference 3 for the BFN transition to a 10 CFR 50.48(c) compliant fire protection program (NFPA 805), TVA committed to a new modification (Modification 106 as listed in Enclosure 3 to Reference 3) to "install additional equipment to provide water to the cooling tower lift pump bearing lubrication water system in order to provide this system a water supply independent from the RSW and HPFP systems to ensure that pressure is maintained in the fire protection system during normal operation without using a fire pump." In Reference 4, the NRC staff concluded the TVA's response to the Reference 2 FPE-RAI 04 is acceptable because TVA provided the requested information and added a plant modification to ensure the NFPA 805, Chapter 3 requirements are met. The NRC staff concluded that this action is acceptable because it will incorporate the provisions of NFPA 805 in the fire protection program and would be required by the proposed license condition. The Reference 4 license condition for implementing the proposed modification 106 is no later than the end of the second refueling outage (for each BFN unit) following issuance of the NFPA 805 license amendment. EPU does not affect either the current TVA use of the HPFP system to supplement the RSW system or the modification for NFPA 805 described above.

There is no other approved non-fire suppression use of fire protection water. Thus, the fire protection system design demands will not be impacted except 1) in the case of a beyond design basis event where system use could be directed consistent with industry accepted guidelines, 2) in the case of abnormal conditions where either the RSW pumps are manually secured, the RSW head tanks are isolated or the RSW pumps trip, which results in manual starting of a fire pump consistent with the Reference 4 licensing basis or 3) the current use of a fire pump to supplement RSW to provide water to the cooling tower lift pump bearing lubrication water system (a use that will be eliminated by modifications to comply with the Reference 4 licensing basis). A beyond design basis event would also include a design basis threat that includes a concurrent fire. In that case, the fire protection system would be used for fire-fighting purposes and may be used for non-fire suppression purposes.

The use of the fire protection system for non-fire suppression purposes during such beyond design basis events involving a concurrent fire would involve decisions by shift operations and onsite and offsite Emergency Response Organization (ERO) personnel in accordance with existing EOI and SAMG procedures, the BFN Emergency Plan, and BFN Emergency Management Guidelines. This would include the use of existing BFN equipment available for mitigation of extreme damage events. These decisions would be based on protection of the health and safety of the public and site personnel, and on the capability of available plant equipment and site personnel. In such an event, offsite fire protection equipment would be brought onsite if necessary to augment the onsite FPS capability.

References

1. TVA letter to U.S. Nuclear Regulatory Commission, "Browns Ferry Nuclear Plant, Units 1, 2, and 3, License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specification Change TS-480)," dated March 27, 2013 (ML13092A393).

2. TVA letter to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding the LAR to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants for the Browns Ferry Nuclear Plant, Units 1, 2, and 3," December 17, 2014 (ML14363A057).

3. TVA letter to U.S. Nuclear Regulatory Commission, "Update to License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (TAC Nos. MF1185, MF1186, and MF1187) - Revised Safe Shutdown Analysis Request for Additional Information 15," dated September 8, 2015 (ML15251A598).

4. U.S. Nuclear Regulatory Commission letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program In Accordance With 10 CFR 50.48(c) (CAC Nos. MF1185, MF1186, and MF1187)," dated October 28, 2015 (ML15212A796).

Withhold from Public Disclosure Under 10 CFR 2.390

ENCLOSURE 2

Supplement to PUSAR (NEDC-33860P, Revision 0) and FUSAR (ANP-3403P, Revision 2)

(Proprietary versions)

ENCLOSURE 3

Supplement to PUSAR (NEDO-33860, Revision 0) and FUSAR (ANP-3403NP, Revision 2)

(Non-proprietary versions)

HITACHI

GE Hitachi Nuclear Energy

NEDO-33860
Revision 0
September 2015

Non-Proprietary Information – Class I (Public)

SAFETY ANALYSIS REPORT
FOR
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3
EXTENDED POWER UPRATE

UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria set out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the Browns Ferry UFSAR where there is subject matter relating to the intent of that particular criteria.

While Browns Ferry is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the Browns Ferry comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in Browns Ferry UFSAR Appendix A: draft GDC-4. Final GDC-3 is applicable to Browns Ferry as described in the Browns Ferry Fire Protection Report, Volume 1, Revision 20.

Fire Protection is described in Browns Ferry UFSAR Section 10.11, "Fire Protection Systems" and the Fire Protection Report.

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the Browns Ferry License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 11). The fire protection systems are documented in NUREG-1843, Section 2.3.3.6. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, and structural steel are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them. Management of aging effects on the fire protection systems is documented in NUREG-1843, Section 3.3.

Insert A

~~By letter dated March 27, 2013 (Reference 65), TVA submitted a LAR to transition Browns Ferry's licensing basis to the National Fire Protection Association (NFPA) 805 standard in accordance with 10 CFR 50.48(c). The current licensing basis per 10 CFR 50.48(b) will be superseded. The transition to NFPA 805 is currently under NRC staff review, and TVA anticipates its approval prior to implementing EPU operation. Accordingly, the fire protection analysis described in this section is based on NFPA 805 implementation. Although TVA fully expects approval of the transition to NFPA for Browns Ferry, Appendix A to this PUSAR provides an EPU evaluation under the current fire protection program in accordance with 10 CFR 50.48(b) and 10 CFR 50, Appendix R requirements.~~

Technical Evaluation

2.5.1.4.1 Fire Protection Program

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the

Insert A:

By letter dated October 28, 2015 (Reference 65), the NRC issued license amendments approving the transition of Browns Ferry’s licensing basis to the National Fire Protection Association (NFPA) 805 standard in accordance with 10 CFR 50.48(c).

59. Letter, Margaret H. Chernoff (NRC) to Karl W. Singer (TVA), "Browns Ferry Nuclear Plant, Units 1, 2 and 3 — Issuance of Amendments Regarding The Instrument Setpoint Program (TAC NOS. MC9518, MC9519, AND MC9520) (TS-453)," September 14, 2006. (ADAMS Accession Number ML061680008).

60. RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977.

61. GE Hitachi Nuclear Energy Services Information Letter, "Main Steam Line High Flow Trip Setting," SIL No. 438, Revision 2, May 13, 2013.

62. GE Hitachi Nuclear Energy 10 CFR Part 21 Communication, "Error in Main Steam Line High Flow Calculational Methodology," SC 12-18 Revision 2, February 8, 2013.

63. GE Nuclear Energy Safety Communication, SC04-14, "Narrow Range Water Level Instrument Level 3 Trip Final Report," October 11, 2004.

64. Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," Revision 1, July 1977.

Insert B

~~65. TVA Letter to NRC, License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specification Change TS-480), March 27, 2013 (ADAMS Accession Number ML13092A392).~~

66. National Fire Protection Association, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," NFPA 805, 2001 Edition.

67. Nuclear Energy Institute, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," NEI 04-02, Revision 2, April 2008.

68. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

69. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Revision 2, March 1978.

70. R. E. Adams and R.D. Ackley, "Removal of Elemental Radioiodine from Flowing Humid Air by Iodized Charcoals," ORNL-TM-2040, November 2, 1967.

71. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (ML003739601)," Revision 2, June 1974.

72. Letter, Eva A. Brown (NRC) to Karl W. Singer (TVA), "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of

Insert B:

65. NRC Letter to TVA, Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in accordance with 10 CFR 50.48(c) (CAC NOS. MF 1185, MF 1186, and MF 1187), October 28, 2015 (ADAMS Accession Number ML15212A796).

APPENDIX A

EPU fire event evaluation based on the current fire protection program in accordance with 10 CFR 50.48(b) and 10 CFR 50, Appendix R requirements.

2.5.1.4 Fire Protection

Technical Evaluation

2.5.1.4.1 Fire Protection Program

NEDC-33004P-A, Revision 4, "Constant Pressure Power Uprate," Class III, July 2003 (also referred to as CLTR) was approved by the NRC as an acceptable method for evaluating the effects of CPPUs. Section 6.7 of the CLTR addresses the effect of CPPU on the FPP. The results of this evaluation are described below.

As explicitly stated in Section 6.7 of the CLTR, [[

]] Therefore, the reactor and containment responses and operator actions were evaluated on a plant-specific basis for EPU.

This section addresses the effect of EPU on the FPP, fire suppression and detection systems, and reactor and containment system responses to postulated 10 CFR 50 Appendix R fire events. Browns Ferry meets all CLTR dispositions.(1) The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Fire Suppression and Detection Systems	Plant Specific	Meets CLTR Disposition
Operator Response Time	Plant Specific	Meets CLTR Disposition(1)
Peak Cladding Temperature	Plant Specific	Meets CLTR Disposition(1)
Vessel Water Level	Plant Specific	Meets CLTR Disposition(1)
Suppression Pool Temperature	Plant Specific	Meets CLTR Disposition(1)

(1) The Browns Ferry Fire Protection Program has been analyzed at EPU conditions. These analyses show that Browns Ferry meets all CLTR dispositions. However, Browns Ferry currently is not in compliance with 10 CFR 50 Appendix R and is presently employing compensatory measures, allowed by discretionary enforcement, while the plant transitions to NFPA 805 fire protection requirements (Reference 65).

The higher decay heat associated with EPU may reduce the time available for the operator to perform the actions necessary to achieve and maintain cold shutdown conditions. The higher decay heat also may result in higher suppression pool temperatures, in lower vessel water levels or higher peak cladding temperatures (PCTs), depending on the plant-specific analysis basis. A plant-specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming EPU conditions. The evaluation determined the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event.

[[

]] The EPU requires no new operator actions for fire safe shutdown of the plant and there are no operator manual actions required inside the primary containment.

The effect of EPU on pump NPSH during a fire event is discussed in Section 2.6.5.2.

Browns Ferry does not take credit in any safety analysis for the fire protection system other than for fire protection activities. Procedural guidance is provided under Emergency Operator Instructions (EOIs), Severe Accident Management Guidelines (SAMGs), and Safe Shutdown Instructions (SSIs), for utilizing fire protection system pumps to supply water to the reactor, spent fuel storage pools, the drywell, or the suppression chamber, if necessary. However, this use of the non-safety related fire protection system is not credited in any safety analysis, and EPU operation will not require any changes to these procedures regarding the utilization of the fire protection system.

The reactor and containment responses to the postulated 10 CFR 50 Appendix R fire events at EPU conditions are provided in Section 2.5.1.4.2. The results show that the peak fuel cladding temperature, reactor water level, and suppression pool temperature are within the acceptance limits. There is an analytical reduction of five minutes from CLTP conditions for the operators to perform the necessary actions to achieve and maintain safe shutdown conditions; however the actual time, procedurally stipulated for this action, remains unchanged. Cold shutdown is achieved well within the 72 hours required by Appendix R.

Therefore, with consideration of the information contained in footnote (1) from the table under Section 2.5.1.4.1, the Fire Protection Program at Browns Ferry meets all CLTR dispositions.

2.5.1.4.2 Fire Event

The limiting Appendix R fire events were analyzed under EPU conditions. The fuel heatup analysis was performed using the NRC approved AREVA LOCA Methodology (RELAX/HUXY). The containment analysis was performed using the NRC approved GEH SHEX model (Reference 7). These evaluations were used to determine the effect of EPU on fuel cladding integrity, reactor vessel integrity, and containment integrity as a result of the fire event.

The four shutdown methods defined in the Browns Ferry Fire Protection Report are as follows:

Case 1: No spurious operation of plant equipment occurs and the operator opens three MSRVs at 25 minutes into the event.

Case 2: One MSRV opens immediately due to a spurious opening signal generated as a result of the fire. The MSRV is reclosed at 10 minutes into the event due to the operator action. The operator opens three MSRVs at 20 minutes into the event.

Case 3: One MSRV opens immediately as in Case 2, but remains open throughout the event. The operator opens three MSRVs at 20 minutes into the event.

Case 4: MSRVs are used to maintain a controlled reactor cooldown with HPCI providing reactor makeup inventory. As reactor pressure decreases, HPCI will trip and make-up inventory is provided by RHR in LPCI alignment while continuing in SPC mode. To achieve cold shutdown, RHR is realigned at 12 hours into SDC mode.

The bounding PCT for Browns Ferry is seen in shutdown method Case 1 with one RHR pump in LPCI mode. The peak PCT is 1119°F which is less than the 1500°F acceptance criteria. (See FUSAR Section 2.5.1.4.2).

The highest peak suppression pool temperature (SPT) for Browns Ferry occurs in shutdown method Case 1. This case follows the assumption, stated in the Browns Ferry Fire Protection Report, that offsite power is assumed to be unavailable during a fire event. While performing analyses for Browns Ferry's NFPA 805 transition, it was learned that a slight revision to Case 1, called Case Max SPT, bounds Case 1 with respect to effect on containment parameters. Case Max SPT is outlined below.

Case Max SPT (See Tables 2.5-1, 2.5-2, 2.5-3 and Figure 2.5-1)

As part of the NFPA 805 transition, a variation of Case 1 (identified as Case Max SPT) was generated to examine the scenario involving no loss of offsite power and the continued injection of condensate into the RPV until the hotwell inventory is exhausted. Case Max SPT bounds Case 1 with respect to effect on containment parameters. Identical to Case 1 except instead of having one RHR pump aligned in the LPCI mode at 20 minutes, a condensate pump is allowed to maintain vessel inventory until the hotwell contents are exhausted (approximately 40 minutes). Then one RHR pump is aligned in the LPCI mode.

If offsite power were available, which is possible for fires in some areas, then it would be possible for a condensate pump to inject the hotwell volume to the RPV via the condensate/feedwater system. This inventory, approximately 90,000 gallons, would get heated by the piping and reactor core and eventually relieved to the suppression pool through the MSRVs when the RHR pump injects into the vessel in the LPCI/ASDC mode. The peak SPT for Case Max SPT is 208°F and this meets the containment integrity acceptance criteria of <281°F and the torus attached piping limit of <223°F (See Section 2.2.2.2.2.2).

This is the worst-case scenario for peak SPT and is used as an input in the analysis for available NPSH for the safe shutdown system pumps. Analyses show that containment accident pressure credit is not required to ensure adequate pump NPSH to mitigate a fire event. (See Section 2.6.5.2 and EPU LAR Attachment 39.)

The results of Case Max SPT, the evaluations in Section 2.6.5.2, FUSAR Section 2.5.1.4.2 and EPU LAR Attachment 39, demonstrate that the peak fuel cladding temperature, vessel water level, and suppression pool temperature meet the acceptance criteria and the actual time stipulated for the operators to perform the necessary actions is unchanged from CLTP conditions. With the maximum suppression pool temperature of 208°F using the RHR heat exchanger K-value (307 Btu/sec-°F) in EPU LAR Attachment 39, the time to reach cold shutdown is within the 72 hours required by Appendix R. Therefore, EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of a fire event and satisfies the requirement of achieving and maintaining safe shutdown conditions in the event of a fire.

Table 2.5-1 Appendix R Fire Case Max SPT (EPU) Fire Event Key Inputs

Input Parameters	Values
Reactor Thermal Power	3952 MWt
RPV Dome Pressure	1055 psia
Decay Heat	ANS 5.1-1979 without 2σ uncertainty adder and with GEH SIL 636 recommendations
Initial Suppression Pool Liquid Volume	124,200 ft^3
Initial Suppression Pool and Wetwell Airspace Temperature	95 °F
Initial Wetwell Pressure	14.4 psia
Initial Drywell Pressure	15.5 psia
Initial Drywell Temperature	150 °F
Initial Wetwell Relative Humidity	100%
Initial Drywell Relative Humidity	20%
Drywell and Wetwell and Pool Heat Sinks Modeled	Yes
Drywell Heat Load Modeled	Yes
RHR Service Water Temperature	92 °F
RHR Heat Exchanger "K" Factor per Loop	307 Btu/sec-°F
Number of RHR Loops Available	1
Number of RHR Pump in one RHR Loop	1
ASDC RHR Flow Rate	7,500 gpm
Condensate Available for Injection	90,000 gallons

Table 2.5-2 Appendix R Fire Event Evaluation Results

Parameter	CLTP (105% of OLTP)	EPU Shutdown Method Case 1 (120% of OLTP)	Appendix R Criteria
Peak Fuel Cladding Temperature (°F)	Note 1	1119 Case 1	1500
Maximum Reactor Pressure at Vessel Bottom Head (psig)	Note 1	1224 Case 1	1375
Maximum Drywell Pressure (psig)	Note 1	15 Case 4	56
Maximum Drywell Temperature (°F)	Note 1	<281[2] Case 4	281
Bulk Suppression Pool Temperature (°F)	Note 1	208 Case Max SPT	Note 3

Notes:

1. Formal calculations for the Appendix R Fire Event were not performed at CLTP conditions.
2. Assuming no drywell cooling is available, the drywell air temperature would exceed 281°F at 18.5 hours. In the implementing SSIs for this safe shutdown pathway, RHR shutdown cooling is entered at 12 hours, which will result in a rapid decrease in reactor pressure, water saturation temperature, and vessel temperature. Therefore, the primary containment wall temperature will not be exceeded using this fire safe shutdown pathway. Entry into ASDC mode at 12 hours would have a similar, but more immediate effect in reducing drywell temperature because water at suppression chamber temperature is pumped to the reactor.
3. The bulk suppression pool temperature must be low enough to assure adequate suppression capability during reactor depressurization and to assure adequate net positive suction head for the systems using the suppression pool as a water source.

Table 2.5-3 Appendix R Fire Case Max SPT (EPU) Sequence of Events

Approximate Elapsed Time	Events
0 seconds	• Reactor scram occurs. • Main Steam Isolation Valves (MSIVs) start to close. • Feedwater pump is tripped. • Drywell coolers are tripped. • Condensate system continues to operate.
3.5 seconds	MSIVs are fully closed. After isolation, MSRVs automatically start to open and close to maintain RPV pressure.
25 min	Begin rapid depressurization using 3 MSRVs. RPV makeup is supplied by the Condensate system.
~ 40 min	Condensate inventory available for injection is depleted. Operators secure condensate flow and initiate ASDC using 7,500 gpm of RHR flow in the LPCI mode.
2 hours	RHR heat exchanger is placed into service.
72 hours	Event is terminated.

Table 2.6-3 Browns Ferry Peak Suppression Pool Temperature for Postulated ATWS, Station Blackout, and Appendix R Fire Events

Event	Peak Suppression Pool Temperature
Limiting ATWS (Loss of Offsite Power)	173.3°F
Station Blackout	203.7°F
Appendix R Fire	208.0°F

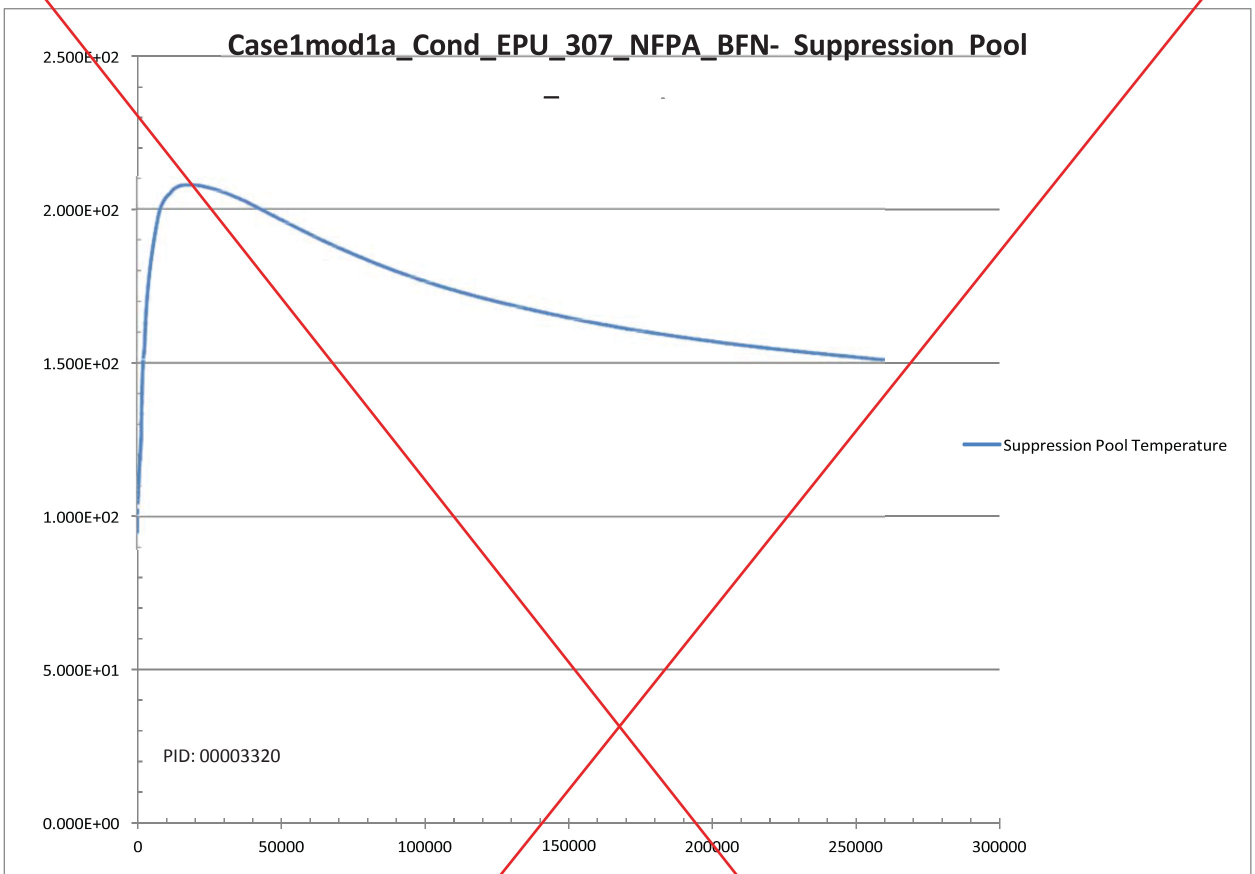

Figure 2.5-1 Appendix R Case Max SPT (EPU) Fire Event Suppression Pool Temperature

Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3

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Revision 2

August 2015

AREVA Inc.

Tables

Figures

proposed General Design Criteria of 1967. The BFN UFSAR, Appendix A, "Conformance to AEC Proposed General Design Criteria," contains this comparative evaluation. This evaluation discusses each of the groups of criteria sent out in the July 1967 AEC release. For each group of criteria, there is a statement of TVA's understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of each of the proposed criteria is a table of references to locations in the BFN UFSAR where there is subject matter relating to the intent of that particular criteria.

While BFN is not generally licensed to the final GDC or the 1967 AEC proposed General Design Criteria, a comparison of the final GDC to the applicable AEC proposed General Design Criteria can usually be made. For the final GDC listed in the Regulatory Evaluation above, the BFN comparative evaluation of the comparable 1967 AEC proposed General Design Criteria (referred to here as "draft GDC") is contained in BFN UFSAR Appendix A: draft GDC-4. Final GDC-3 is applicable to BFN as described the BFN Fire Protection Report, Vol. 1, Rev. 20.

Fire Protection is described in BFN UFSAR Section 10.11, "Fire Protection Systems" and the Fire Protection Report.

Systems and system component materials of construction, operating history, and programs used to manage aging effects were evaluated for plant license renewal and documented in the BFN License Renewal Safety Evaluation Report (SER), NUREG-1843, dated April 2006 (Reference 7). The fire protection systems are documented in NUREG-1843, Section 2.3.3.6. Fire barrier materials are addressed as a commodity group, while walls, floors, doors, structural steel etc., are evaluated within the building that contains them. Components credited with achieving safe shutdown following a fire are evaluated within the system that contains them. Management of aging effects on the fire protection systems is documented in NUREG-1843, Section 3.3.

Insert C

~~**Technical Evaluation – Appendix R**~~

~~The limiting Appendix R fire event was analyzed assuming operation with AREVA fuel at EPU conditions. The Appendix R analyses were performed with LOCA Evaluation Models developed by AREVA and approved for reactor licensing analyses by the U.S. Nuclear Regulatory~~

Insert C:

By letter dated October 28, 2015, the NRC issued license amendments approving the transition of Browns Ferry's licensing basis to the National Fire Protection Association (NFPA) 805 standard in accordance with 10 CFR 50.48(c).

Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model. The EXEM BWR-2000 Evaluation Model and NRC approval are documented in References 8 through 12. The LOCA analysis codes are not explicitly approved for Appendix R analyses; however the physical phenomena, depressurization via the ADS, tracking the water level, and the fuel response (calculation of PCT) are similar to LOCA requirements.

The evaluation determined the effect of EPU on fuel cladding integrity and reactor vessel integrity as a result of the fire event.

The analyses were performed using plant parameters and plant geometry presented specifically for BFN Units 1, 2, and 3 in Reference 13. The plant parameters specified are based on EPU operation and the plant geometry includes any modifications necessary for EPU at BFN.

A complete Appendix R fire event analysis will address the following topics:

- Fire suppression and detection systems
- Operator response time
- Reactor vessel water level
- Suppression pool temperature
- Peak cladding temperature (PCT)

The topics of “fire suppression and detection systems” and “suppression pool temperature” are not fuel related and were not analyzed by AREVA. The behavior of reactor vessel water level in an Appendix R event is also not fuel related however, vessel water level was calculated in the analysis.

Three cases or scenarios of Appendix R events were analyzed. The cases differ primarily in the assumptions for main steam relief valve (MSRV) operation.

- Case 1 (base case) – 3 MSRVs are opened by operator action 25 minutes after accident initiation.
- Case 2 – Spurious signal opens 1 MSRV (Stuck Open Relief Valve (SORV)) for 10 minutes after accident initiation. 3 MSRVs are opened at 20 minutes.
- Case 3 –Spurious signal opens 1 MSRV (SORV) throughout the event. 3 MSRVs are opened at 20 minutes.

The primary parameter calculated in the cases was PCT. The results from these calculations are provided in Table 2.5-1, Figure 2.5-1, and Figure 2.5-2.

The Appendix R event blowdown phase was defined as the period after the start of the fire when mass is lost through the MSRVs. Blowdown was assumed to end when the ADS (MSRVs) decreased vessel pressure to the pressure where rated LPCS flow would have occurred if it were available. The vessel liquid level typically drops below the top of the active fuel during blowdown. One LPCI pump was available to recover the core during the refill period. Reflood was assumed to occur when sufficient entrained liquid mass flow to end the fuel heatup was calculated at the core hot node. PCT occurs at the time of hot node reflood. The system and hot channel calculations were continued past the time of hot node reflood to confirm reflood of the entire core and to calculate the vessel water level behavior after the time of hot node reflood.

The results of the Appendix R analysis for AREVA fuel at EPU conditions demonstrate that fuel cladding integrity and reactor vessel integrity will be maintained using the licensee specified and validated operator response times. Case 1 is the limiting case. The PCT is 1119°F and the peak vessel pressure is 1224 psia, both of which remain below the acceptance limits of 1500°F and 1375 psig.

Technical Evaluation – NFPA 805

The limiting NFPA 805 fire event was analyzed assuming operation with AREVA fuel at EPU conditions. The NFPA 805 analyses were performed with LOCA Evaluation Models developed by AREVA and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by AREVA for LOCA analyses are collectively referred to as the EXEM BWR-2000 Evaluation Model. The EXEM BWR-2000 Evaluation Model and NRC approval are documented in References 8 through 12. The LOCA analysis codes are not explicitly approved for NFPA 805 analyses; however the physical phenomena, depressurization via the ADS, tracking the water level, and the fuel response (calculation of PCT) are similar to LOCA requirements.

The evaluation determined the effect of EPU on fuel cladding integrity and reactor vessel integrity as a result of the fire event.

Table 2.5-1 Browns Ferry Appendix R Fire Event Results

CASE	ECCS	PCT (°F)	Peak Pressure at Bottom of Vessel (psia)
Appendix R Case 1 3 MSRVs opened at 25 minutes	1 LCPI operational	1119	1224
Appendix R Case 2 3 MSRVs opened at 20 minutes with 1 MSRV Open for 10 Minutes	1 LCPI operational	1000	1214
Appendix R Case 3 3 MSRVs opened at 20 minutes with 1 MSRV Open Throughout	1 LCPI operational	828	1214

Table 2.5-2 Browns Ferry NFPA 805 Fire Event Results

CASE	ECCS	PCT (°F)	Peak Pressure at Bottom of Vessel (psia)
MSO of 11 MSRVs	1 LPCI on at 20 Minutes	1330	1097

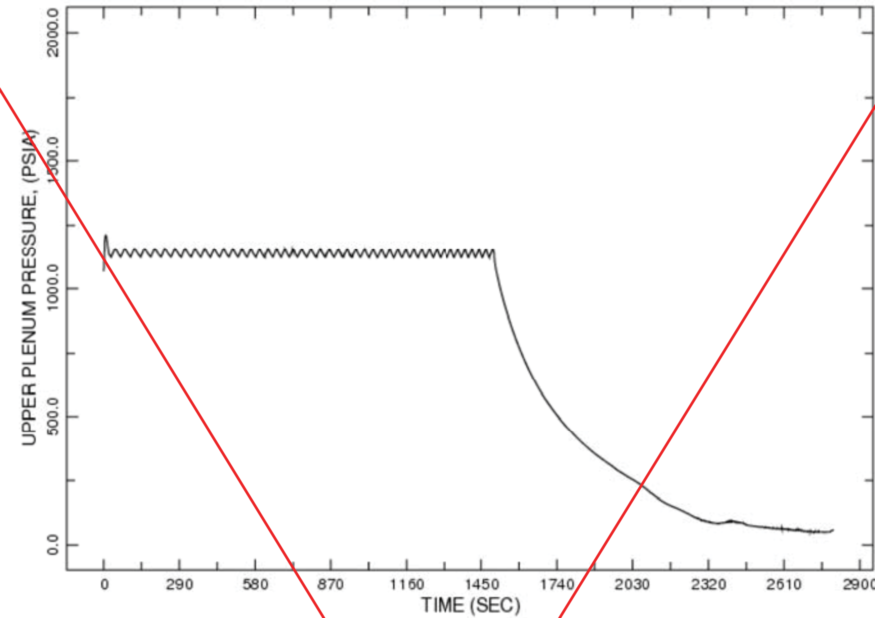

Figure 2.5-1 Appendix R Upper Plenum Pressure Case 1 - 3 MSRVs Opened at 25 Minutes

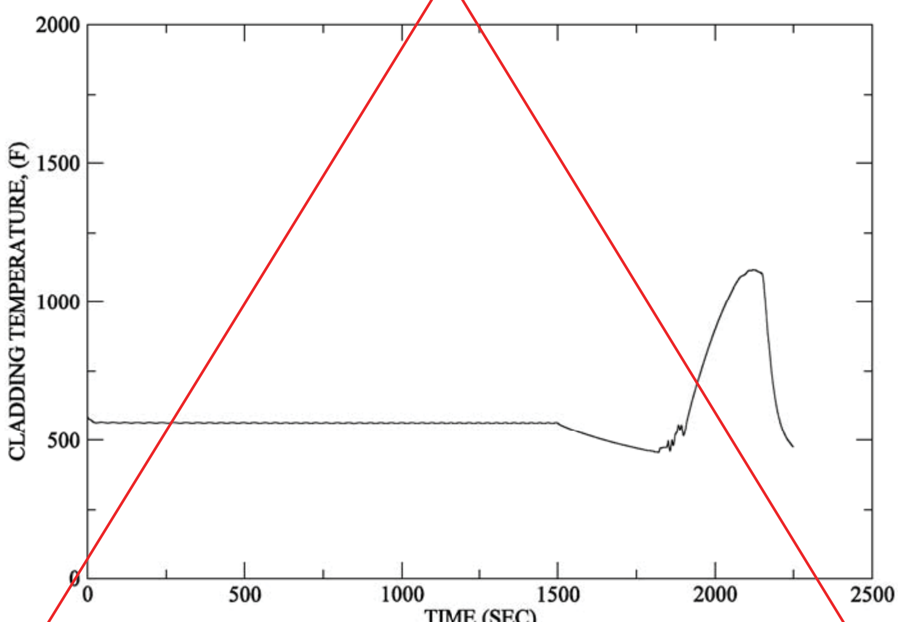

Figure 2.5-2 Appendix R PCT Rod Surface Temperature Case 1 - 3 MSRVs Opened at 25 Minutes

Enclosure 4

AREVA Affidavit for ANP-3403P, Revision 2

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the report ANP-3403P, Revision 2, "Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3," dated August 2015 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 19th day of August, 2015.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY PUBLIC
01-14-16
STATE OF WASHINGTON

Enclosure 5

GE Hitachi Nuclear Energy Affidavit for NEDC-33860P, Revision 0

GE-Hitachi Nuclear Energy Americas LLC
AFFIDAVIT

I, James F. Harrison, state as follows:

(1) I am Vice President, Fuel Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC ("GEH"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH proprietary report, NEDC-33860P, *Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate*, Revision 0, dated September 2015. GEH proprietary information within text is identified by a dotted underline within double square brackets. [[This sentence is an example.{3}]] Figures and large objects containing GEH proprietary information are identified with double square brackets before and after the object. In all cases, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2.d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2.d 1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GEH.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions regarding supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of the analysis for a GEH Boiling Water Reactor (BWR). The analysis utilized analytical models and methods, including computer codes, which GEH has developed, obtained NRC approval of, and applied to perform evaluations of Power Uprates for a GEH BWR. The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical

methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on this 17th day of September 2015.

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