~~Proprietary Information and Critical Energy Infrastructure Information~~
~~Withhold from Public Disclosure Under 10 CFR 2.390~~
~~This letter is decontrolled when separated from Enclosure 4~~

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

CNL-16-109

July 13, 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 23, Miscellaneous Updates**

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 TVA to NRC, CNL-16-079, "Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU) - Supplement 15, Responses to Requests for Additional Information," dated May 11, 2016 (ML16133A580)

3. Letter from NRC to TVA, “Browns Ferry Nuclear Plant, Unit 3 - Issuance of Amendment Regarding Modification of Technical Specification 3.4.9, “RCS Pressure and Temperature (P/T) Limits” (CAC No. MF5659),” dated January 7, 2016 (ML15344A321)

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. This supplement provides updates to some of the enclosures previously provided by the Reference 1 letter.

Enclosure 1 of this letter provides a discussion of the changes provided in Enclosures 2 through 7 of this letter.

Enclosure 2 of this letter provides Revision 1 to the BFN EPU Flow Induced Vibration Analysis and Monitoring Program. The due date for transmitting this supplement to the BFN EPU LAR Attachment 45 provided by the Reference 2 letter was June 29, 2016. Due to the time required to complete the necessary analysis revisions and reviews, the due date for this transmittal was extended to July 15, 2016, per communication with the NRC Project Manager. The BFN EPU Flow Induced Vibration Analysis and Monitoring Program is revised to correct the acceptance criteria values provided in Tables 3-1 through 3-3. The EPU projected vibration remains below the revised acceptance criteria value at all monitoring locations. Enclosure 2 supersedes and replaces Attachment 45 of the BFN EPU LAR (Reference 1).

Enclosure 3 of this letter provides a supplement to the BFN EPU LAR Evaluation of Proposed Change. The BFN EPU LAR Evaluation of Proposed Change is revised to reflect the issuance of the BFN Unit 3 License Amendment regarding modification of TS 3.4.9, “RCS Pressure and Temperature (P/T) Limits” (Reference 3). Enclosure 3 supersedes and replaces the enclosure entitled, “Evaluation of Proposed Change” of the BFN EPU LAR (Reference 1).

Enclosure 4 of this letter provides a supplement to Section 2.1.2 of the Power Uprate Safety Analysis Report (PUSAR) (NEDC-33860P, Revision 0). The supplement is required to reflect the issuance of the BFN Unit 3 License Amendment regarding modification of TS 3.4.9, “RCS Pressure and Temperature (P/T) Limits” (Reference 3). Enclosure 4 also provides a supplement to PUSAR Table 2.7-1, “EPU Effect on Ventilation Systems.” PUSAR Table 2.7-1 stated, "The turbine building is not an [Environmental Qualification] EQ zone." This sentence is corrected to read, "The bulk of the turbine building is not an EQ zone.[1]" Footnote 1 states, “The Main Steam Tunnel is an EQ Zone and is located in the Turbine Building.”

GE-Hitachi Nuclear Energy Americas LLC (GEH) and the Electric Power Research Institute (EPRI) separately consider portions of the information provided in Enclosure 4 of this letter to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390, Public inspections, exemptions, requests for withholding. Affidavits for withholding information, executed by GEH and EPRI, are provided in Enclosure 8 and Enclosure 9, respectively. Enclosure 5 is a non-proprietary version of the supplement to the PUSAR provided in Enclosure 4. Therefore, on behalf of GEH and EPRI, TVA requests that Enclosure 4 be withheld from public disclosure in accordance with the GEH and EPRI affidavits and the provisions of 10 CFR 2.390. Enclosures 4 and 5 supersede and replace Section 2.1.2 and Table 2.7-1 of Attachments 6 and 7, respectively, of the BFN EPU LAR (Reference 1), dated September 21, 2015.

Enclosure 6 of this letter provides a markup of TS Bases Section B 3.6.1.4, "Drywell Air Temperature." Enclosure 7 of this letter provides a retype of TS Bases Section B 3.6.1.4. Enclosures 6 and 7 of this letter supplement the markup and the retype of affected TS Bases pages provided in Attachments 4 and 5, respectively, of the BFN EPU LAR (Reference 1).

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 critical energy infrastructure information or 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 13th day of July 2016.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosures

cc: See Page 4

ENCLOSURE 1

BFN Extended Power Uprate (EPU) - Supplement 23 - Discussion of Changes

ENCLOSURE 1

BFN Extended Power Uprate (EPU) - Supplement 23 - Discussion of Changes

Letter Enclosure	Discussion of Changes
Enclosure 2 - BFN EPU LAR, Attachment 45, Flow Induced Vibration, Revision 1	Enclosure 2 of this letter provides Revision 1 to the BFN EPU Flow Induced Vibration Analysis and Monitoring Program. The BFN EPU Flow Induced Vibration Analysis and Monitoring Program is revised to correct the acceptance criteria values provided in Tables 3-1 through 3-3. The acceptance criteria values provided in Tables 3-1 through 3-3 have been corrected and updated based on further reviews of the results obtained from the time history analyses described in Section 4.2.1 of BFN EPU LAR, Attachment 45. The projected percent of acceptance criteria values are also updated as a result of the updates to the acceptance criteria values. The EPU projected vibration remains below the revised acceptance criteria value at all monitoring locations. This condition, requiring the revision of BFN EPU LAR Attachment 45, has been entered into the Corrective Action Program.
Enclosure 3 - Supplement to BFN EPU LAR, Evaluation of Proposed Change	Enclosure 3 of this letter provides a supplement to the BFN EPU LAR Evaluation of Proposed Change which was transmitted by 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). BFN EPU LAR Evaluation of Proposed Change, Section 2.5, "TS Containing Changes That Have Already Been Made," is revised to reflect the issuance of the BFN Unit 3 License Amendment regarding modification of Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits" (Reference 3).
Enclosure 4 - Supplement to BFN EPU LAR, Attachment 6, NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate, Section 2.1.2 and Table 2.7-1 (Proprietary version)	Enclosure 4 of this letter provides a supplement to Section 2.1.2 of the Power Uprate Safety Analysis Report (PUSAR) (NEDC-33860P, Revision 0). The supplement is required to reflect the issuance of the BFN Unit 3 License Amendment regarding modification of Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits" (Reference 3). Enclosure 4 also provides a supplement to PUSAR Table 2.7-1, "EPU Effect on Ventilation Systems." PUSAR Table 2.7-1 stated, "The turbine building is not an [Environmental Qualification] EQ zone." This sentence is corrected to read, "The bulk of the turbine building is not an EQ zone.[1]" Footnote 1 states, "The Main Steam Tunnel is an EQ Zone and is located in the Turbine Building." The change is made to accurately reflect the BFN design and analysis basis. The condition, requiring the revision of BFN EPU LAR PUSAR Table 2.7-1, has been entered into the Corrective Action Program.

Letter Enclosure	Discussion of Changes
Enclosure 5 - Supplement to BFN EPU LAR, Attachment 6, NEDO-33860, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate, Section 2.1.2 and Table 2.7-1 (Non-proprietary version)	The same change discussion as provided for the Enclosure 4 change above applies to this change.
Enclosure 6 - Supplement to BFN EPU LAR, Attachment 4, Proposed Technical Specification Bases Changes (Markups)	Enclosure 6 of this letter provides a markup of Technical Specification (TS) Bases section B 3.6.1.4, "Drywell Air Temperature." TS Bases B 3.6.1.4 (typical all three units) provides a peak drywell temperature of 336°F under the "Applicable Safety Analyses" section. Table 2.6-1 of the EPU LAR PUSAR (NEDC-33860P) provides a peak drywell temperature of 336.9 °F. The TS Bases B 3.6.1.4 peak drywell temperature of 336°F is changed to "337°F" to reflect the PUSAR (NEDC-33860P) peak drywell temperature. The TS Bases reference (i.e., "Reference 2") for the source of the 337°F value is also changed for consistency. The condition, requiring the revision of TS Bases B 3.6.1.4, has been entered into the Corrective Action Program.
Enclosure 7 - Supplement to BFN EPU LAR, Attachment 5, Retyped Proposed Technical Specification Bases Changes	Enclosure 7 of this letter provides a retype of TS Bases Section B 3.6.1.4. The same change discussion as provided for the Enclosure 6 change above applies to this change.

ENCLOSURE 2

BFN EPU LAR, Attachment 45, Flow Induced Vibration, Revision 1

Attachment 45

Flow Induced Vibration Analysis and Monitoring Program

Revision 1

Pages changed by Revision 1 are as follows:

Page 45-5, Section 3.0, Results from Previous Vibration Test Programs and EPU Projects,
Page 45-6, Table 3-1, CLTP Results and EPU Projections for Piping Monitoring Locations Inside Containment,
Page 45-7, Table 3-2, CLTP Results and EPU Projections for Large Bore Piping Monitoring Locations Outside Containment,
Page 45-8, Table 3-3, CLTP Results and EPU Projections for Small Bore Piping Monitoring Locations Outside Containment

1.0 INTRODUCTION

This Attachment to the submittal provides a detailed discussion of the analyses and testing program undertaken to provide assurance that unacceptable flow induced vibration (FIV) issues are not experienced at Browns Ferry Nuclear Plant (BFN) due to extended power uprate (EPU) implementation for affected piping systems.

Increased flow rates and flow velocities during operation at EPU conditions are expected to produce increased FIV levels in some systems. As discussed in Section 3.4.1 of Licensing Topical Report (LTR) NEDC-33004P-A, Revision 4, “Constant Pressure Power Uprate,” the Main Steam (MS) and Feedwater (FW) system piping vibration levels should be monitored because their system flow rates will be significantly increased (Reference 1).

In December 2008, the Boiling Water Reactor Owners’ Group (BWROG) issued NEDO-33159, Revision 2, “Extended Power Uprate (EPU) Lessons Learned and Recommendations,” based on operating experience (OE) and evaluations from Boiling Water Reactor (BWR) plants that have previously implemented EPUs and from plants currently performing pre-EPU evaluations. NEDO-33159 (Reference 2) states:

> “Since the majority of EPU-related component failures involve flow induced vibration, the BWROG EPU Committee held a vibration monitoring and evaluation information exchange meeting of industry experts in June 2004. The committee determined with the current process of monitoring large bore piping systems in accordance with the requirements of ASMEO&M Part 3 is sufficient to preclude challenges to safe shutdown. Increases in large bore piping vibration levels are a precursor to increased vibration levels in attached small bore piping and components.”

Regulatory Guide (RG) 1.20 (Reference 3), “Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing,” was revised in 2007 to Revision 3. In addition to guidance for vibration assessment of reactor internals, this regulatory guide provides helpful information on methods for evaluating the potential adverse effects from pressure fluctuations and vibrations in piping systems for boiling water reactor (BWR) nuclear power plants. However, additional guidance is provided with regard to piping vibration. The guidance is primarily directed to initial start-up of new plants, with general guidance interpreted for use in power uprate power ascension testing. Where applicable, this guidance has been incorporated into the EPU monitoring program for piping vibration at BFN.

In addition to MS and FW, the related Extraction Steam (ES), Condensate (CD) and Heater Drain (HD) systems also experience similar flow increases under EPU conditions and are included in the EPU vibration monitoring program. Other systems experience insignificant or no increase in flow and; therefore, are not included in this program.

Review of power ascension vibration data collected during initial restart of BFN Unit 1 indicates vibration levels well within acceptable limits at current licensed thermal power (CLTP). Extrapolation of this earlier data to EPU power levels indicates that vibration of piping and components will not be adversely affected by EPU operation.

This document describes the piping vibration monitoring program to be implemented at TVA during power ascension to confirm acceptable vibration levels at EPU power. It compares previously collected vibration data to conservative projections for EPU vibration levels based on increases in vibration being proportional to increases in flow rate squared. It addresses systems impacted by EPU and identifies locations on those systems where monitoring equipment will be installed. This document also describes the techniques to be used for collecting and storing the vibration data.

2.0 SUSCEPTIBILITY AND MONITORING

The MS and FW piping will experience higher mass flow rates and flow velocities under EPU conditions. When power is increased, steady state FIV levels are conservatively expected to increase in proportion to the flow velocity squared. Thus, the vibration levels of the MS and FW piping are expected to increase by approximately 35% from CLTP to EPU conditions and 58.5% from OLTP to EPU conditions, based on flow increases of up to 16% for CLTP and 23% for OLTP. Other possible sources of increased vibration, such as flow instabilities or acoustic resonance as a result of increased flow velocities, may contribute to EPU vibration levels. It is noted that acoustic vibration suppressors have been installed on the MS system at BFN to reduce vibration susceptibility of piping and components.

Flow rates in portions of the CD, ES and HD systems increase similarly to MS and FW, and are, therefore, susceptible to increased vibration at EPU conditions.

Based on the potential for significantly increased vibrations on the systems identified above, a confirmatory test program will be implemented to monitor piping and attached component vibration levels on the identified systems during initial power ascension to EPU conditions. The test program will incorporate the guidance and OE discussed in Section 1.0, industry experience from recently implemented EPU FIV monitoring programs and other industry OE related to FIV issues experienced in piping and attached components.

Piping inside containment and inaccessible piping outside containment will be monitored using vibration sensors (accelerometers or displacement transducers) installed at selected locations on the piping and attached components. The vibration sensors will be wired to remote data acquisition systems located in the reactor and turbine buildings. Piping outside containment that is included in the monitoring program and is accessible during plant operation will be monitored either remotely or by performing visual observations or taking vibration measurements using hand-held vibration instruments during power ascension to EPU conditions.

Small bore branch piping is susceptible to the effects of the associated large bore piping FIV. Modifications to small bore branch piping to reduce susceptibility to header-induced vibrations have been made as a result of BFN operating experience. Small bore piping assessments, supplemented by confirmatory walkdowns, will be performed during the refueling outage prior to the EPU implementation outage for each unit to identify any additional potentially susceptible configurations. Any necessary small bore line modifications will be made prior to EPU power ascension. Selected small bore branch lines will be monitored for vibration during EPU power ascension to confirm that vibrations are within acceptable limits.

3.0 RESULTS FROM PREVIOUS VIBRATION TEST PROGRAMS AND EPU PROJECTS

Vibration levels at CLTP (3458 MWt) were obtained as part of the BFN Unit 1 restart in 2007, with additional CLTP data obtained in 2008, for MS and FW piping and components. The Unit 1 CLTP vibration monitoring results are part of the basis for the vibration monitoring to be performed during EPU power ascension for BFN Units 1, 2 and 3. The Unit 1 baseline vibration monitoring results are used to demonstrate that projected vibrations are anticipated to be acceptable. This conclusion is applicable to the other units based on the general similarity of the three units. For the analyses performed to determine monitoring locations and acceptance criteria, the unit specific piping and support configurations are taken into account.

The MS and FW monitoring locations included in the 2007 and 2008 monitoring scope are summarized below:

<u>Inside Containment</u>

MS Piping: 7 monitoring locations, 12 measurements (1 or 2 directions per location)

FW Piping: 9 monitoring locations, 14 measurements (1 or 2 directions per location)

MS Components: 8 monitoring locations, 24 measurements (3 directions per location)

<u>Outside Containment</u>

MS piping: 11 monitoring locations, 20 measurements (1 to 3 directions per location)

FW piping: 13 monitoring locations, 23 measurements (1 or 2 directions per location)

The CLTP measured vibration levels, projected EPU vibration levels and comparisons of EPU projections with acceptance criteria are summarized in Tables 3-1 through 3-4 of this attachment. The projected EPU vibration levels are calculated using the following equation:

$$\text{EPU vibration level} = (\text{CLTP vibration level}) * (\text{EPU flow rate} / \text{CLTP flow rate})^2$$

The acceptance criteria were developed using the methodology described in Section 4.2.

The results presented in Tables 3-1 through 3-4 of this attachment illustrate the acceptability of previously-measured vibrations. Based on conservative projections, vibrations at EPU conditions are expected to remain within acceptable limits.

Updates to Tables 3-1 through 3-3

The acceptance criteria values provided in Tables 3-1 through 3-3 have been updated based on further reviews of the results obtained from the time history analyses described in Section 4.2.1. One or more of the following adjustments were made to the acceptance criteria values as applicable to each specific analysis and monitoring location:

a. The time history analysis output stresses were originally interpreted as being equal to the nominal stress multiplied by the stress intensification factor 'i'. It was later determined that the analysis output stresses actually correspond to the nominal stress multiplied by 0.75i, where 0.75i cannot be less than 1.0. Therefore, the analysis output stresses have been adjusted to determine stress values corresponding to the nominal stress multiplied by 'i'. This adjustment, when applied by itself, can result in an acceptance criteria value at a given location that is either less than or equal to the original acceptance criteria value. The issue relating to the misinterpretation of the time history analysis output stresses has been entered into the TVA Corrective Action Program.

b. An additional adjustment to the acceptance criteria values was made to account for differences between the modulus of elasticity values used in the analyses and those used as the bases for the ASME fatigue curves. This adjustment was made to provide consistency with guidance provided in revisions to Reference 4 made after determination of the original acceptance criteria. This adjustment, when applied by itself, can result in an acceptance criteria value at a given location that is either less than, equal to or greater than the original acceptance criteria value.

c. For some monitoring locations, the location of the governing stress used to calculate the acceptance criteria value changed. This change occurred as a result of the adjustments made in Items a and b and examination of the maximum stresses in the proximity of the monitoring location. This adjustment resulted in an acceptance criteria value at a given location that was either less than or greater than the original acceptance criteria value.

d. At some locations, based on the symmetry of the piping configuration in two monitoring directions, it is appropriate to combine the acceptance criteria values in the two monitoring directions by the square root of the sum of the squares (SRSS) to obtain an SRSS acceptance criteria value. At these locations, the measured vibrations are acceptable if the SRSS value of the measured vibrations in the two directions is less than the SRSS acceptance criteria value. This method of evaluation was used for monitoring location G99 in previous Table 3-3.

The net effect of the adjustments made in Items a through d above is that the updated acceptance criteria values are lower than the originally provided acceptance criteria values at all but two monitoring locations, where the values increased (monitoring location 246 in Table 3-1 and monitoring location A310 in Table 3-2). Note that the original acceptance criteria values in the x and z directions at monitoring location E30/E40 in Table 3-3 were transposed, so the acceptance criteria values are actually reduced in both directions at that location.

The projected % of acceptance criteria values are also updated as a result of the updates to the acceptance criteria values. The EPU projected vibration remains below the revised acceptance criteria value at all locations.

Table 3-1
CLTP Results and EPU Projections for Piping Monitoring Locations Inside Containment

System	Piping Identifier	Monitoring Location-Direction	CLTP Measured Vibration (Note 1)	EPU Projected Vibration (Note 1)	Acceptance Criteria (Note 1)	Projected % of Acceptance Criteria
MS	MS Line B	A3A-T	15	20	~~67~~ 51	~~30~~ 39
		A3A-R	10	14	~~55~~ 41	~~25~~ 34
MS	MS Line A	15-T	14	19	~~100~~ 76	~~19~~ 25
		15-R	7	9	~~100~~ 76	~~9~~ 12
MS	MS Line C	246-T	12	16	~~25~~ 30	~~64~~ 53
		246-R	18	24	~~40~~ 48	~~60~~ 50
MS	MS Ring H	85-R	3	4	~~84~~ 50	~~5~~ 8
		85-Y	15	20	~~84~~ 50	~~24~~ 40
MS	MS Line B	19A-Y	2	3	~~98~~ 55	~~3~~ 5
MS	MS Line C	40-X	5	7	~~31~~ 12	~~23~~ 58
		40-Z	2	3	~~61~~ 23	~~5~~ 13
MS	MS Line C	36,37-Y	5	7	~~106~~ 40	~~7~~ 18
FW	FW Nozzle B	BT-X	0.27	0.36	~~2.71~~ 2.53	~~13~~ 14
		BT-Z	0.17	0.23	~~1.53~~ 1.43	~~15~~ 16
FW	FW Nozzle C	16-R	0.28	0.38	~~2.23~~ 2.08	~~17~~ 18
		16-T	0.15	0.2	~~0.93~~ 0.87	~~22~~ 23
FW	FW Nozzle A	ATA-R	0.2	0.27	~~3.83~~ 3.56	~~7~~ 8
FW	FW Ring Header	19A-Y	0.17	0.23	~~1.77~~ 1.65	~~13~~ 14
FW	FW Nozzle F	8A-R	0.15	0.2	~~3.23~~ 3.01	~~6~~ 7
FW	FW Nozzle E	24A-X	0.25	0.34	~~1.59~~ 1.48	~~21~~ 23
		24A-Z	0.16	0.22	~~5.02~~ 4.67	~~4~~ 5
FW	FW Ring Header	15D-Y	0.24	0.32	~~1.31~~ 1.22	~~24~~ 26
FW	FW Nozzle D	42A-R	0.37	0.5	~~3.07~~ 2.40	~~16~~ 21
		42A-T	0.21	0.28	~~3.05~~ 2.39	~~9~~ 12
FW	FCV-3-562 FW	55BQ-V	0.42	0.57	~~0.90~~ 0.84	~~63~~ 68
		55BQ-T	0.12	0.16	~~0.47~~ 0.44	~~34~~ 36

Note 1: Vibration values shown are in terms of displacement (mils pk-pk) for MS and acceleration (g's peak) for FW.

Table 3-2
CLTP Results and EPU Projections for Large Bore Piping Monitoring Locations Outside Containment

System	Piping Identifier	Monitoring Location-Direction	CLTP Measured Vibration (mils pk-pk)	EPU Projected Vibration (mils pk-pk)	Acceptance Criteria (mils pk-pk)	Projected % of Acceptance Criteria
MS	Main Steam Line B 24"	B125-X	42	57	~~75~~ 70	~~76~~ 81
MS	Main Steam Line D 24"	D125-X	45	61	~~106~~ 99	~~58~~ 62
MS	Bypass Valves 8" Line	L75-Y	24	32	~~122~~ 114	~~26~~ 28
		L75-Z	23	31	~~100~~ 94	~~31~~ 33
MS	Main Steam Line A 28"	A310-Z	79	107	~~167~~ 293	~~64~~ 37
		A310-X	53	72	~~87~~ 152	~~83~~ 47
		A310-Y	13	18	~~80~~ 141	~~23~~ 13
MS	Main Steam Line C 28"	C290-X	31	42	~~66~~ 62	~~64~~ 68
		C290-Y	2	3	~~160~~ 149	2
		C290-Z	44	59	~~247~~ 230	~~24~~ 26
FW	RFP 1A 18" Discharge	A38-Y	9	12	~~52~~ 41	~~23~~ 29
		A38-X	7	9	~~104~~ 82	~~9~~ 11
FW	RFP 1A 18" Discharge	47-Z	16	22	~~311~~ 244	~~7~~ 9
FW	RFP 1B 18" Discharge	142A-Y	2	3	~~129~~ 102	~~2~~ 3
		142A-X	2	3	~~108~~ 85	~~3~~ 4
FW	RFP 1B 18" Discharge	132A-Z	11	15	~~324~~ 254	~~5~~ 6
FW	RFP 1C 18" Discharge	80A-Y	2	3	~~129~~ 102	~~2~~ 3
FW	Heater String A2 18" Line	215B-Z	15	20	~~187~~ 131	~~11~~ 15
		215B-X	4	5	~~120~~ 84	4 6
FW	Heater String A1 18" Line	95A-Y	1	1	~~33~~ 23	~~3~~ 4
		95A-X	1	1	~~58~~ 40	~~2~~ 3
FW	Heater String C1 18" Line	32-Y	3	4	~~37~~ 26	~~11~~ 15
		32-Z	3	4	~~46~~ 32	~~9~~ 13
FW	RFW 24" Disch Return	135A-X	10	14	~~45~~ 31	~~31~~ 45
		135A-Z	1	1	~~48~~ 34	~~2~~ 3

Table 3-3
CLTP Results and EPU Projections for Small Bore Piping Monitoring Locations Outside Containment

System	Piping Identifier	Monitoring Location-Direction	CLTP Measured Vibration (mils pk-pk)	EPU Projected Vibration (mils pk-pk)	Acceptance Criteria (mils pk-pk)	Projected % of Acceptance Criteria
MS	Main Steam Line A 1"[1]	M30-X	139	(1)	(1)	(1)
		M30-Z	97	(1)	(1)	(1)
MS	Main Steam Line C 1"[1]	N30-X	44	(1)	(1)	(1)
		N30-Z	70	(1)	(1)	(1)
MS	Stop Valve 1C	F37-X	37	50	~~222~~ 159	~~23~~ 31
		F37-Z	11	15	~~272~~ 195	~~6~~ 8
MS	Control Valve 1A 1" Line	G99-~~X~~SRSS(X,Y)	62	84	~~101~~ 100	~~83~~ 84
		~~G99-Y~~ N/A	~~4~~ N/A	~~5~~ N/A	~~95~~ N/A	~~5~~ N/A
MS	Control Valve 1C 2.5" Line	G55-Z	3	4	~~121~~ 87	~~3~~ 5
MS	Control Valve 1D 1" Line	G22-X	11	15	~~99~~ 71	~~15~~ 21
FW	RFP 1A .5" Discharge	E30/E40-X	4	5	~~1165~~ 446	~~<1~~ 1
		E30/E40-Z	5	7	~~555~~ 936	1
FW	RFP 1A 1" Vent	F20/F40-X	3	4	~~95~~ 73	4 5
		F20/F40-Z	20	27	~~377~~ 287	~~7~~ 9
FW	RFP 1C 1" Vent	G20/G40-X	4	5	~~22~~ 17	~~23~~ 29
		G20/G40-Z	29	39	~~59~~ 45	~~66~~ 87
FW	RFP 1C 1.5" Vent	H31-Z	2	3	~~41~~ 30	~~7~~ 10
		H31-Y	4	5	~~102~~ 73	~~5~~ 7

Note 1: Tie-back support installed after CLTP measurements to mitigate header-induced vibration effects.

Table 3-4
CLTP Results and EPU Projections for Main Steam Valve Monitoring Locations

Valve ID	Valve Description	Monitoring Direction	CLTP Measured Vibration (g's rms)	EPU Projected Vibration (g's rms)	Acceptance Criteria (g's rms)	Projected % of Acceptance Criteria
FCV-1-14	MSIV	X	(1)	N/A	0.260	N/A
FCV-1-14	MSIV	Y	(1)	N/A	0.136	N/A
FCV-1-14	MSIV	Z	0.10	0.14	0.386	36
FCV-1-55	MS Drain	X	0.06	0.08	0.166	48
FCV-1-55	MS Drain	Y	0.04	0.05	0.214	23
FCV-1-55	MS Drain	Z	(1)	N/A	0.157	N/A
FCV-71-2	RCIC	X	0.06	0.08	0.166	48
FCV-71-2	RCIC	Y	0.04	0.05	0.215	23
FCV-71-2	RCIC	Z	0.05	0.07	0.157	45
FCV-73-2	HPCI	X	0.04	0.05	0.374	13
FCV-73-2	HPCI	Y	(1)	N/A	0.234	N/A
FCV-73-2	HPCI	Z	0.06	0.08	0.234	34
PCV-1-4	SRV	X	0.09	0.12	0.69	17
PCV-1-4	SRV	Y	0.09	0.12	0.90	13
PCV-1-4	SRV	Z	0.08	0.11	0.40	28
PCV-1-34	SRV	X	0.12	0.16	0.69	23
PCV-1-34	SRV	Y	0.10	0.14	0.90	16
PCV-1-34	SRV	Z	0.15	0.2	0.40	50
PCV-1-22	SRV	X	0.08	0.11	0.69	16
PCV-1-22	SRV	Y	0.11	0.15	0.90	17
PCV-1-22	SRV	Z	0.05	0.07	0.40	18
PCV-1-180	SRV	X	0.07	0.09	0.69	13
PCV-1-180	SRV	Y	0.10	0.14	0.90	16
PCV-1-180	SRV	Z	0.10	0.14	0.40	35

Note 1: Inoperable sensor.

4.0 EPU VIBRATION MONITORING PROGRAM

4.1 Overview

The portions of the MS, FW, CD, HD and ES systems included in the EPU vibration monitoring program have been selected based on evaluation of the flow increases resulting from EPU implementation. The specific EPU vibration monitoring locations and acceptance criteria are established using detailed analysis methods, as described in Section 4.2. The EPU flow increase evaluation and vibration analysis results form the bases for EPU vibration monitoring.

Several MS-associated components will also be monitored. Although BFN does not have a history of safety-relief valve maintenance issues due to vibration, selected safety-relief valves will be instrumented with accelerometers, as well as four other power-operated valves. This is in response to industry OE from an earlier EPU project. A representative sample of valves were selected to monitor the effect of EPU flow changes on the vibration levels at the primary valves in the system with symmetry between trains, loops and units considered to remove unnecessary redundancies.

4.2 Vibration Monitoring Locations and Acceptance Criteria Development

4.2.1 MS and FW Piping (Inside and Outside Containment)

Hydraulic and structural models of the MS and FW piping were created for determination of the vibration monitoring locations and development of the vibration acceptance criteria. The hydraulic analyses were performed to generate piping leg force time histories simulating loading due to dynamic pressure fluctuations that cause piping steady-state vibrations. The generated force time histories were used as input for force time history analyses performed to provide piping structural responses. The intent of the hydraulic and structural dynamic analyses was to apply loading that is similar to the loading due to steady-state vibration, and generate responses that are based on the piping system acoustic and structural properties. Because the exact forcing functions are unknown, the analytical responses are not predicted responses. However, the deflected shape of the piping and the resulting stress distribution will correspond to the appropriate type of loading.

The vibration monitoring locations were selected where, based on the structural time history analysis results, significant displacements occurred relative to other locations. The measurement locations were also selected such that the general overall piping response would be reflected in the data and it would not be likely that significant vibrations would be missed. Where applicable, symmetry between trains or loops was considered to reduce the overall number of monitoring locations. The EPU vibration monitoring locations determined for the MS and FW piping from the analyses are summarized in Tables 4-1 through 4-3 of this attachment.

Allowable displacement (mils pk-pk) and acceleration (g's-pk) limits at the selected measurement locations were calculated based on the analysis results and ASME code fatigue stress limits for steady state vibration consistent with ASME OM-S/G, Part 3 (OM-3)

(Reference 4). The primary acceptance criteria are in terms of displacement, which is directly proportional to pipe stress. Secondary acceptance criteria in terms of acceleration were determined for locations where accelerometers are used for monitoring.

The displacement limits for MS and FW are applicable for vibration frequencies up to 50 Hz, which corresponds to the frequency range in which the most significant structural displacement responses are expected. Piping displacements due to excitation frequencies above 50 Hz are typically insignificant relative to the lower frequency displacements. Secondary acceleration limits established for the FW piping inside containment are also applicable for frequencies up to 50 Hz, since significant forcing frequencies and structural responses above 50 Hz are not expected in the FW system.

Small bore piping attached to the MS and FW piping were reviewed for potential susceptibility to header-induced vibrations. The lines determined to be most susceptible were selected for monitoring and acceptance criteria were developed accordingly. The following factors were considered for the small bore line evaluations:

- The presence or absence of a tie-back support. Tie-back supports are added to reduce the influence of header-induced vibrations on small bore lines. Therefore, lines with tie-back supports are generally not susceptible to header-induced vibrations.
- The routing and support configurations of the small bore lines. Lines with unsupported concentrated masses or long, unsupported runs are generally most susceptible to header-induced vibrations.
- The expected amplitudes of the header vibrations. The more rigidly supported the header piping is in the vicinity of the branch connection, the lower the amplitudes of the header vibrations. The expected relative amplitudes of the header vibrations are checked in the header time history analyses.
- Small bore lines included in the large bore piping models. In these cases, the time history analysis results are used to determine the susceptibility of the small bore lines to the header-induced vibrations.

Table 4-1
EPU Monitoring Locations for MS and FW Piping (Inside Containment)[1]

System	Location	Direction	Description
MS	A3A	T	MS Line B – El. 620.50’
MS	A3A	R	MS Line B – El. 620.50’
MS	15	T	MS Line A – El. 621.00’
MS	15	R	MS Line A – El. 621.00’
MS	246	T	MS Line C – El. 621.00’
MS	246	R	MS Line C – El. 621.00’
MS	85	R	MS Ring H – El. 586.58’
MS	85	Y	MS Ring H – El. 586.58’
MS	19A	Y	MS Line B – El. 578.08’
MS	40	X	MS Line C – El. 575.26’
MS	40	Z	MS Line C – El. 575.26’
MS	36,37	Y	MS Line C – El. 584.33’/578.22’
FW	BT	X	FW Nozzle B – El. 613.41’
FW	BT	Z	FW Nozzle B – El. 613.41’
FW	I6	R	FW Nozzle C – El. 610.00’
FW	I6	T	FW Nozzle C – El. 610.00’
FW	ATA	R	FW Nozzle A – El. 611.32’
FW	19A	Y	FW Ring Header – EL. 587.00’
FW	8A	R	FW Nozzle F – El. 611.55’
FW	24A	X	FW Nozzle E – El. 611.42’
FW	24A	Z	FW Nozzle E – El. 611.42’
FW	15D	Y	FW Ring Header – El. 587.09’
FW	42A	R	FW Nozzle D – El. 611.64’
FW	42A	T	FW Nozzle D – El. 611.64’
FW	55BQ	V	FCV-3-562 FW
FW	55BQ	T	FCV-3-562 FW

Note 1: The specific node numbers and locations listed in Table 4-1 correspond to BFN Unit 1. The equivalent locations in BFN Units 2 and 3, as applicable, will also be monitored.

Table 4-2
EPU Monitoring Locations for MS and FW Large Bore Piping (Outside Containment)[1]

System	Location	Direction	Description
MS	B125	X	MS Line B 24”
MS	D125	X	MS Line D 24”
MS	L75	Y	Bypass Valves 8” Line
MS	L75	Z	Bypass Valves 8” Line
MS	A310	Z	MS Line A 28”
MS	A310	X	MS Line A 28”
MS	A310	Y	MS Line A 28”
MS	C290	X	MS Line C 28”
MS	C290	Y	MS Line C 28”
MS	C290	Z	MS Line C 28”
FW	A38	Y	RFP 1A 18” Disch.
FW	A38	X	RFP 1A 18” Disch
FW	47	Z	RFP 1A 18” Disch.
FW	142A	Y	RFP 1B 18” Disch.
FW	142A	X	RFP 1B 18” Disch.
FW	132A	Z	RFP 1B 18” Disch.
FW	80A	Y	RFP 1C 18” Disch.
FW	215B	Z	Heater String A2 18” Line
FW	215B	X	Heater String A2 18” Line
FW	95A	Y	Heater String A1 18” Line
FW	95A	X	Heater String A1 18” Line
FW	32	Y	Heater String C1 18” Line
FW	32	Z	Heater String C1 18” Line
FW	135A	X	RFW 24” Disch. Return
FW	135A	Z	RFW 24” Disch. Return

Note 1: The specific node numbers and locations listed in Table 4-2 correspond to BFN Unit 1. The equivalent locations in BFN Units 2 and 3, as applicable, will also be monitored.

Table 4-3
EPU Monitoring Locations for MS and FW Small Bore Piping (Outside Containment)[1]

System	Location	Direction	Description
MS	M30	X	Main Steam Line A 1”
		Z	
MS	N30	X	Main Steam Line C 1”
		Z	
MS	F37	X	Stop Valve 1C
		Z	
MS	G99	X	Control Valve 1A 1” Line
		Y	
MS	G55	Z	Control Valve 1C 2.5” Line
MS	G22	X	Control Valve 1D 1” Line
FW	E30/E40	X	RFP 1A .5” Line
		Z	
FW	F20/F40	X	RFP 1A 1” Vent
		Z	
FW	G20/G40	X	RFP 1C 1” Vent
		Z	
FW	H31	Z	RFP 1C 1.5” Vent
		Y	

Note 1: The specific node numbers and locations listed in Table 4-3 correspond to BFN Unit 1. The equivalent locations in BFN Units 2 and 3, as applicable, will also be monitored.

4.2.2 CD, ES and HD Piping (Outside Containment)

Significant flow increases occur in portions of the condensate, extraction steam and heater drain systems as a result of EPU. The portions of the systems selected for monitoring were based on the percent flow increase due to EPU, projected EPU flow rates, a review of the piping configurations and similarities between trains and units. Determination of specific monitoring locations and acceptance criteria will be based on analysis methodologies consistent with ASME OM-3.

Condensate:

The condensate system will experience a flow increase of approximately 16% as a result of EPU. The piping between the 3rd stage feedwater heaters and the reactor feedwater pumps (RFPs) as well as the piping between the 4th stage feedwater heaters and the 3rd stage feedwater heaters were selected for EPU vibration monitoring.

Extraction Steam:

The extraction steam system will experience flow increases in the piping from the high pressure (HP) turbine to the 1st stage feedwater heaters and the piping from the low pressure (LP) turbine to the 2nd stage feedwater heaters of approximately 22% and 20%, respectively, as a result of EPU. The piping in these two portions of the extraction steam system was selected for EPU vibration monitoring.

Heater Drain:

The heater drain system will experience flow increases in the normal drain piping between the 1st and 2nd stage feedwater heaters and between the 2nd and 3rd stage feedwater heaters of approximately 22% and 20%, respectively, as a result of EPU. Based on a review of the piping configurations for these two portions of the heater drain system, the piping between the 2nd and 3rd stage feedwater heaters was selected for EPU vibration monitoring.

The portions of the CD, ES and HD systems selected for EPU vibration monitoring are summarized in Table 4-4.

Table 4-4
EPU Monitoring Locations for CD, ES and HD, BFN Units 1, 2 and 3

System	Description
CD	Piping from FW Heaters 3A/B/C to RFPs 1A/B/C
CD	Piping from FW Heaters 4A/B/C to FW Heaters 3A/B/C
ES	Piping from HP Turbine to FW Heaters 1A/B/C
ES	Piping from LP Turbine to FW Heaters 2A/B/C
HD	Piping from FW Heaters 2A/B/C to FW Heaters 3A/B/C

4.2.3 MS Components (Inside Containment)

BFN operating history indicates that excessive component vibrations are not expected at EPU conditions. In order to provide confirmation that component vibrations will be within acceptable limits at EPU conditions, selected components will be instrumented with accelerometers. The selected components include four safety-relief valves (SRV), one main steam isolation valve (MSIV), the inboard isolation valve for the MS drain piping, the inboard isolation valve for the reactor core isolation cooling (RCIC) turbine steam supply line and the inboard isolation valve for the high pressure coolant injection (HPCI) turbine steam supply line. Both the RCIC and HPCI lines are attached to the MS piping. The EPU component vibration monitoring locations are summarized in Table 4-5.

Component vibration acceptance criteria are based on the dynamic characteristics of the specific components, the frequency content of the excitation vibrations, including acoustic vibration; and industry experience for similar valves.

Table 4-5
EPU Component Monitoring Locations, BFN Units 1, 2 and 3

System	Valve ID	Direction	Description
MS	FCV-1-14	X	MS Line A Inboard Isolation Valve
MS		Y	
MS		Z	
MS	FCV-1-55	X	MS Drain Header Inboard Isolation Valve
MS		Y	
MS		Z	
RCIC	FCV-71-2	X	RCIC Steam Supply Line Inboard Isolation Valve
RCIC		Y	
RCIC		Z	
HPCI	FCV-73-2	X	HPCI Steam Supply Line Inboard Isolation Valve
HPCI		Y	
HPCI		Z	
MS	PCV-1-4	X	MS Line A SRV
MS		Y	
MS		Z	
MS	PCV-1-34	X	MS Line B SRV
MS		Y	
MS		Z	
MS	PCV-1-22	X	MS Line C SRV
MS		Y	
MS		Z	
MS	PCV-1-180	X	MS LINE D SRV
MS		Y	
MS		Z	

4.3 Data Acquisition and Reduction Methodology

The vibration data will be collected during EPU power ascension at pre-determined power levels using PC-based digital data acquisition systems (DAS). Each data set will be recorded using a minimum sample rate of 2000 samples per second per channel for a minimum duration of one minute.

The raw time history data for each power level will be processed for comparison to applicable acceptance criteria. The data processing will include integration, determination of peak, peak-to-peak and root mean square (rms) values, and high and low pass filtering, as applicable for specific monitoring locations, sensor types and acceptance criteria bases. Additional data processing, such as frequency analysis, will be performed to aid data analysis, as required.

4.4 Required Actions for Test Exceptions

The FIV data collected at each test plateau above CLTP will be processed and compared to the established acceptance criteria to demonstrate acceptability of the monitored piping and components. Level 1 and Level 2 criteria are established to aid in evaluation of the data and decision making during power ascension. A test exception will be generated if either Level 1 or Level 2 criteria are not satisfied.

The Level 1 criteria correspond to the calculated vibration limits. If a Level 1 criterion is not met, the plant will be placed in a safe condition until the issue can be resolved. This is accomplished by reducing power to the last power level where the Level 1 criteria were met. Once the issue is resolved, testing will be repeated at the applicable test plateau to verify that the Level 1 criteria are satisfied.

The Level 2 criteria are set at some percentage of the calculated vibration limits to provide sufficient warning that a Level 1 limit may be exceeded before the next test plateau. If a Level 2 criterion is not met, power will not be increased above the current power level until the issue is resolved. An evaluation will need to be completed to demonstrate that Level 1 criteria will still be satisfied at the next test plateau. Data may need to be retaken at the current test plateau depending on the resolution.

5.0 SUMMARY

Review of previous vibration data collected during BFN Unit 1 restart power ascension testing, as discussed in Section 3, indicates CLTP vibration levels well within acceptable limits. Extrapolation of the CLTP data to EPU power levels indicates that vibration of piping and components will not be adversely affected by EPU operation.

A confirmatory test program will be implemented to perform vibration monitoring during power ascension to EPU conditions. Piping and attached components on systems experiencing significant flow increases as a result of EPU will be included in the monitoring program. Piping vibration acceptance criteria will be based on ASME OM-3. Component vibration acceptance criteria will be based on component-specific dynamic characteristics and industry experience. Small bore piping assessments will be performed to identify potentially susceptible configurations, and any modifications required to reduce vibration susceptibility will be made prior to EPU power ascension.

Monitoring of inaccessible piping and components will be accomplished using vibration sensors wired to remote data acquisition systems. Accessible piping included in the monitoring program will be monitored either remotely or by performing visual observations or by taking vibration measurements using hand-held vibration instruments during power ascension to EPU conditions.

6.0 REFERENCES

1. GE Nuclear Energy, “Constant Pressure Power Uprate,” Licensing Topical Report NEDC- 33004P-A, Revision 4, Class III, July 2003.

2. BWR Owners’ Group EPU Committee, “Extended Power Uprate (EPU) Lessons Learned and Recommendations,” NEDO-33159 Revision 2, December 2008, BWR Owners’ Group EPU Committee.

3. U.S. Nuclear Regulatory Commission Regulatory Guide 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” Rev. 3.

4. ASME OM-2009, “Operation and Maintenance of Nuclear Power Plants,” Division 2, Part 3, “Vibration Testing of Piping Systems.”

ENCLOSURE 3

Supplement to BFN EPU LAR, Evaluation of Proposed Change

Enclosure

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

License Amendment Request TS-505 – Extended Power Uprate

EVALUATION OF PROPOSED CHANGES

TABLE OF CONTENTS

SUBJECT: Extended Power Uprate

1.0 SUMMARY DESCRIPTION

The Browns Ferry Nuclear Power Plants (BFN) Units 1, 2, and 3 Renewed Operating Licenses (OLs) specify the Maximum Power Level at which BFN Units 1, 2, and 3 may be operated. The proposed amendment increases the Maximum Power Level authorized from 3458 megawatts thermal (MWt) to 3952 MWt. This amendment request includes revision of the OL and Technical Specifications (TS) to support the increased power level. The new Maximum Power represents an increase of approximately 20% above the original rated thermal power (RTP) of 3293 MWt and an increase of approximately 14% above the Current Licensed Thermal Power (CLTP) level of 3458 MWt. The CLTP level for BFN Unit 1 was approved on March 6, 2007 by Amendment No. 269 (Reference 1). The CLTP level for BFN Units 2 and 3 were approved on September 8, 1998, by Amendment Nos. 254 and 214, respectively (Reference 2).

2.0 DETAILED DESCRIPTION

2.1 <u>Power Uprate Safety Analysis Report</u>

NEDC-33860P, "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate" (also called the Power Uprate Safety Analysis Report or PUSAR) is provided in Attachment 6 (proprietary version) and Attachment 7 (non-proprietary version) of this submittal.

The GE-Hitachi Nuclear Energy Americas LLC (GEH) licensing topical report NEDC-33004P-A, "Constant Pressure Power Uprate," Revision 4, dated July 2003 (Reference 3), hereafter referred to as the CLTR, provides an NRC-accepted approach for performing constant pressure power uprates (CPPU). The CPPU approach has been used as the basis for multiple power uprate license amendment requests submitted to and approved by the NRC. As the name suggests, the CPPU approach maintains a plant's current maximum operating reactor pressure. The constant pressure constraint along with other required limitations and restrictions discussed in the CLTR, allows a simplified approach to power uprate analyses and evaluations.

The evaluation methods and conclusions of the CLTR were approved for GE fuel up to and including GE14 fuel assemblies. Because BFN uses a mix of fuel types, the CLTR is not applicable for the fuel design-dependent topics and the associated analyses performed in support of the generic disposition in the CLTR are not applicable. Therefore, for fuel-dependent topics, the PUSAR follows the NRC-approved generic content for BWR extended power uprate (EPU) licensing reports documented in NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" (Reference 4), which is commonly called "ELTR1." ELTR1 provides the process for evaluating safety issues that are plant-specific. For issues that are evaluated generically, the PUSAR follows the NRC-approved generic evaluations in NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (Reference 5), which is commonly called the "ELTR2."

The Office of Nuclear Reactor Regulation document, "Review Standard for Extended Power Uprates," RS-001, dated December 2003 (Reference 6), provides guidance to the NRC Staff when performing reviews of EPU applications. The review standard was developed to enhance the consistency, quality, and completeness of the staff's reviews and to inform licensees of the guidance documents the Staff would use when reviewing EPU applications.

PUSAR Section 2, "Safety Evaluation," follows the format and guidance delineated in RS-001 (Reference 6), Section 3.2, to the extent that the review standard is consistent with the BFN design basis. To facilitate the NRC staff's review of this application, Attachment 48 provides a redline-strikeout mark-up of the matrices contained in RS-001 to identify differences between the review standard and the BFN design bases. Attachment 49 provides a re-type of the RS-001 safety evaluation template.

The PUSAR, as supplemented by ANP-3403P, "Fuel Uprate Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate," hereafter referred to as the FUSAR, is provided in Attachment 8 (proprietary version) and Attachment 9 (non-proprietary version) of this submittal, provides an integrated summary of the results of the safety analyses and evaluations performed in accordance with the CLTR, ELTR1, and ELTR2. The FUSAR supports operation of BFN Units 1, 2, and 3 at EPU conditions with AREVA's ATRIUM 10XM fuel.

These analyses and evaluations support the proposed increase to the maximum power level at BFN to 3952 MWt. These safety analyses also support elimination of the reliance on Containment Accident Pressure (CAP) credit in demonstrating adequate Net Positive Suction Head (NPSH) for the Emergency Core Cooling System (ECCS) pumps.

In developing the PUSAR, the Tennessee Valley Authority (TVA) identified certain evaluations that, due to size, level of detail, and/or subject matter, were more appropriately broken out as separate Attachments to this submittal. These areas include the Steam Dryer Analysis Report (Attachment 40, proprietary version, and Attachment 41, non-proprietary version), Transmission Stability Evaluation (Attachment 43), the Probabilistic Risk Assessment (Attachment 44), the Flow-Induced Vibration Analysis and Monitoring Program (Attachment 45), and the Startup Test Plan (Attachment 46). These evaluations support the appropriate PUSAR Technical Evaluations.

2.2 <u>Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3, and fuel related reports</u>

The FUSAR (ANP-3403) is provided in Attachment 8 (proprietary version) and Attachment 9 (non-proprietary version).

The fuel-related reports, proprietary and non-proprietary versions, where applicable, included in Attachments 10 through 38 are as follows:

- ANP-3377, Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU)

- ANP-3378, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel (EPU)

- ANP-3384, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10 Fuel (EPU)

- ANP-3342, Browns Ferry EPU (120% OLTP) Equilibrium Fuel Cycle Design

- ANP-3372, Browns Ferry Unit 3 Cycle 19 EPU (120% OLTP) LAR Reference Fuel Cycle Design

- ANP-3404, Browns Ferry Unit 3 Cycle 19 Representative Reload Analysis at Extended Power Uprate

- ANP-3343, Nuclear Fuel Design Report Browns Ferry EPU (120% OLTP) Equilibrium Cycle ATRIUM 10XM Fuel

- ANP-3386, Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM 10XM Fuel Assemblies

- ANP-3385, Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM 10 Fuel Assemblies

- ANP-3388, Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Extended Power Uprate

- ANP-3327, Evaluation of AREVA Fuel Thermal-Hydraulic Performance for Browns Ferry at EPU

- FS1-0019629/30, Browns Ferry Unit 3 Cycle 19 MCPR Safety Limit Analysis With SAFLIM3D Methodology

- ANP-2860 Revision 2, Supplement 2, Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information, Extension for Use of ATRIUM 10XM Fuel for Extended Power Uprate

- ANP-2637, Boiling Water Reactor Licensing Methodology Compendium

- ANP-3409, Fuel-Related Emergent Regulatory Issues

The FUSAR and the fuel related reports provide summaries of the results of the analyses addressing the effect of operation of BFN Units 1, 2, and 3 at EPU conditions with ATRIUM 10XM fuel.

2.3 Renewed Operating License and Technical Specifications

The following OL and TS sections, and associated TS Bases, are affected by the proposed EPU for the three BFN Units, except as noted:

- Maximum Power Level (Operating License Section 2.C.(1))

- Potential Adverse Flow Effects (Operating License Section 2.C(4) for Units 2 and 3, and Section 2.C(6) for Unit 1)

- Definitions - Rated Thermal Power (RTP) (TS 1.1)
- Reactor Core Safety Limits (TS 2.1.1)
- Standby Liquid Control (SLC) System (TS 3.1.7)
- Average Planar Linear Heat Generation Rate (APLHGR) (TS 3.2.1)
- Minimum Critical Power Ratio (MCPR) (TS 3.2.2)
- Linear Heat Generation Rate (LHGR) (TS 3.2.3)
- Reactor Protection System (RPS) Instrumentation (TS 3.3.1.1)
- Feedwater and Main Turbine High Water Level Trip Instrumentation (TS 3.3.2.2)
- End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation (TS 3.3.4.1)
- Jet Pumps (TS 3.4.2)
- Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS) (TS 3.7.1) [BFN Units 2 and 3 only]
- Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS) (TS 3.7.2) [BFN Units 2 and 3 only]
- Main Turbine Bypass System (TS 3.7.5)
- Primary Containment Leakage Rate Testing Program (TS 5.5.12)

Section 3.1 of this Enclosure provides the details of the above changes along with the associated technical justification. Attachment 2 contains the proposed TS Change Markups. Attachment 3 contains the retyped proposed TS changes. Associated proposed changes to the TS Bases are provided for information only in Attachment 4 (markups) and Attachment 5 (retyped pages). In addition, some editorial changes, such as removal of outdated footnotes and errant punctuation marks, were also made, as reflected in Attachments 2 and 3, but are not specifically described in this Enclosure. These editorial changes are administrative in nature and do not involve technical changes to the TS.

2.4 <u>TS Containing Percentage of Rated Thermal Power That Are Not Affected</u>

Many of the TS listed above contain criteria or requirements expressed in terms of percent rated thermal power (% RTP) that are re-scaled or otherwise adjusted for the EPU. However, there are several other TS with such criteria that do not require revision to support EPU. The CLTR, Section 11.1, discussed this situation of the TSs expressed in terms of % RTP that may not require a change based on EPU. To ensure clarity, the CLTR provided Table 11-1, which included all % RTP TS. Each TS was dispositioned as to whether it required a change or not. Similarly, to avoid any misunderstanding, TVA provides below the BFN-specific TSs that are expressed in terms of % RTP and are not changing. A brief explanation as to why a revision is unnecessary is included.

1. Control Rod Operability (TS 3.1.3)

   The current TS 3.1.3 Condition D includes a note stating the Condition is not applicable when thermal power is greater than 10% RTP. The stated % RTP is conservatively maintained at the same % RTP as the CLTP. The 10% RTP power level is the power level below which the control rod drop accident (CRDA) analyses assume the reactor operator follows prescribed rod withdrawal sequences (i.e., complies with Banked Position Withdrawal Sequence (BPWS) requirements).

   The BPWS requires control rods to be moved in groups, with all control rods assigned to a specific group within specified banked positions. The banked positions are established to minimize the maximum incremental control rod worth. Analyses demonstrate that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation.

   Maintaining BPWS requirements in effect until 10% RTP of the EPU power level will result in a larger range in terms of absolute power when BPWS requirements apply. Therefore, not revising the TS 3.1.3 Condition D note is conservative for EPU.

2. Control Rod Scram Times (TS 3.1.4)

   Current Surveillance Requirement (SR) 3.1.4.1 and SR 3.1.4.4 Frequencies require verification that control rod scram times are within applicable limits prior to exceeding 40% RTP. The stated % RTP does not change. The 40% RTP provides a reasonable time to complete the scram time testing following a shutdown. As such, this is a timing consideration to allow for the testing to be completed and does not affect the operation or operability of the control rods. Thus, it is acceptable to maintain the current 40% RTP.

3. Rod Pattern Control (TS 3.1.6)

   The applicability of current TS 3.1.6 requirements for BPWS is MODES 1 and 2 with THERMAL POWER ≤ 10% RTP. The stated % RTP is conservatively maintained at the same % RTP as the CLTP. The 10% RTP power level is the power level below which the CRDA analyses assume the reactor operator follows prescribed rod withdrawal sequences (i.e., complies with BPWS requirements).

   The BPWS requires control rods to be moved in groups, with all control rods assigned to a specific group within specified banked positions. The banked positions are established to minimize the maximum incremental control rod worth. Analyses demonstrate that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation.

   Therefore, maintaining the TS 3.1.6 applicability of MODES 1 and 2 with THERMAL POWER ≤ 10% RTP of the EPU power level will result in a larger range in terms of absolute power when BPWS requirements apply and will continue to prevent exceeding the 280 cal/gm fuel design limit during a CRDA.

4. Reactor Protection System (RPS) Instrumentation (TS 3.3.1.1)

   Current Table 3.3.1.1-1, Function 2.b (APRM - Flow Biased Simulated Thermal Power - High) and Function 2.c (APRM - Neutron Flux - High) provide an allowable value of ≤ 120% RTP. Although the APRM – Flow Biased Simulated Thermal Power – High setpoint is changed, the clamped high value remains the same in terms of % RTP.

   Both Function 2.b and Function 2.c will perform the same under EPU as CLTP to the high neutron flux trip setpoint clamp setting. The APRM - Flow Biased Simulated Thermal Power - High trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is slightly lower than or equal to the fixed APRM Neutron Flux – High function allowable value. Because of the increase in RTP from CLTP to EPU, the clamped high value setting is re-scaled to ≤ 120% of the uprated RTP, consistent with the assumptions used in the revised safety analyses. (Refer to PUSAR Section 2.4.1.3.)

5. Control Rod Block Instrumentation (TS 3.3.2.1)

   Current Table 3.3.2.1-1 Function 2, "Rod Worth Minimizer," (SR 3.3.2.1.2, SR 3.3.2.1.3, SR 3.3.2.1.5, and Table 3.3.2.1-1 note (c)) is required to be Applicable, in part, in MODES 1 and 2 with THERMAL POWER ≤ 10% RTP. The stated % RTP is conservatively maintained at the same % RTP as the CLTP. The 10% RTP power level is the power level below which the CRDA analyses assume the reactor operator follows prescribed rod withdrawal sequences (i.e., complies with BPWS requirements).

   The BPWS requires control rods to be moved in groups, with all control rods assigned to a specific group within specified banked positions. The banked positions are established to minimize the maximum incremental control rod worth. Analyses demonstrate that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The Rod Worth Minimizer (RWM) functions to enforce the BPWS requirements.

   Therefore, maintaining the TS 3.3.2.1 Table 3.3.2.1-1 Function 2 (SR 3.3.2.1.2, SR 3.3.2.1.3, SR 3.3.2.1.5, and Table 3.3.2.1-1 note (c)) Applicability of MODES 1 and 2 with THERMAL POWER ≤ 10% RTP of the EPU power level will result in a larger range in terms of absolute power when BPWS and RWM requirements apply and will continue to prevent exceeding the 280 cal/gm fuel damage limit during a CRDA.

   SR 3.3.2.1.8 and Notes (a), (b), (f), (g), and (h) of Table 3.3.2.1-1 Analytical Limit (AL) associated with the Analytical Value power levels for the various ranges of Rod Block Monitor operability are unchanged in terms of percent power for EPU, thus no setpoint change is required. The power-dependent MCPR multiplier at each AL are verified on a cycle specific basis in order to determine if the multiplier is bounding.

6. Drywell-to-Suppression Chamber Differential Pressure (TS 3.6.2.6)

   The current Limiting Condition for Operation (LCO), APPLICABILITY, and REQUIRED ACTIONS for TS 3.6.2.6 include reference to 15% RTP. The applicability for TS LCO 3.6.2.6 is “MODE 1 during the time period from 24 hours after Thermal Power is > 15% RTP following startup, to 24 hours prior to reducing Thermal Power to < 15% RTP prior to the next scheduled reactor shutdown.” In accordance with CLTR Table 11-1 regarding the drywell-to-suppression chamber differential pressure TS, this value does not change for EPU.

   The drywell-to-suppression chamber differential pressure is an assumption in the containment analysis. The drywell-to-suppression chamber differential pressure establishes a MODE 1 operating condition with the drywell at a higher pressure than the suppression chamber. During a postulated design basis loss-of-coolant accident (LOCA), the increasing drywell pressure will discharge mass and energy, including non-condensables, into the wetwell vent header and downcomers. The drywell-to-suppression chamber differential pressure reduces the resultant hydrodynamic load on the suppression chamber during the LOCA blowdown.

   Although the absolute thermal power increases for EPU at 15% RTP, the effects on containment hydrodynamic loads due to a LOCA have been evaluated and remain within specified limits. (Refer to PUSAR Section 2.6.1.2.)

7. Primary Containment Oxygen Concentration (TS 3.6.3.2)

   The Applicability of current TS LCO 3.6.3.2 is MODE 1 during the time period from 24 hours after Thermal Power is greater than 15% RTP following startup, to 24 hours prior to reducing Thermal Power to less than 15% RTP prior to the next scheduled reactor shutdown. The TS LCO 3.6.3.2 Applicability value of 15% RTP is a historical value for requiring containment inerting. The current TS Bases do not reference analyses supporting the 15% power level. Maintaining this LCO power level value at 15% EPU RTP from the current 15% RTP results in an insignificant change in the hydrogen source due to a LOCA and the potential for a fire or explosion is unchanged at EPU conditions. In accordance with CLTR Table 11-1 regarding the applicability of the primary containment oxygen concentration TS, this value does not change for EPU.

2.5 TS Containing Changes That Have Already Been Made

This section describes EPU-related TS changes that have previously been NRC-approved for at least one BFN unit. This information is provided to support NRC staff review of the effects the proposed EPU may have in related areas.

1. Standby Liquid Control (SLC) System (TS 3.1.7)

   Several changes to TS 3.1.7 support EPU:

   a. For BFN Unit 1 only, in SR 3.1.7.5, the value for the minimum quantity of Boron-10 in the SLC System solution tank has changed to greater than or equal to 203 pounds. This change incorporated EPU conditions and was approved by the NRC on March 6, 2007, by Amendment 269, "Five Percent

Uprate," (Reference 1) to the Renewed Facility Operating License for BFN Unit 1. Proposed changes (described in Section 3.1 below) to the BFN Unit 2 and BFN Unit 3 Technical Specifications will make TS 3.1.7 similar for all three BFN units.

b. The borated solution volume in the storage tank must be maintained for reactivity control and Post-LOCA suppression pool pH control. The tank volume requirement for reactivity control is encompassed by the requirement for post-LOCA pH control. The amount of available sodium pentaborate required in SR 3.1.7.1 (greater than or equal to 4000 gallons) does not change for EPU. The volumes provided in the calculation for SLC System Boron-10 requirements demonstrate that EPU requirements are bounded by the volumes calculated for the Alternative Source Term (AST). The AST requirements, at EPU values, were approved by the NRC for BFN Units 1, 2, and 3 on September 27, 2004 (Reference 8). The analyses performed to support these changes were performed at EPU conditions.

c. Other TS changes were approved as part of the AST license change. These include SLC parameters for meeting Anticipated Transients Without Scram (ATWS) concerns. The other changes to the SLC System were:

   i. Changed the SLC Mode of Applicability to require SLC to be operable in Mode 3. Commensurate with the change to the SLC Mode of Applicability, a Required Action and associated Completion Time was added to place the reactor in Mode 4 within 36 hours if the Required Action and associated Completion Time of Actions A or B are not met.

   ii. SR 3.1.7.1 was changed to increase the available volume of sodium pentaborate solution (SPB) from greater than or equal to 3007 gallons to greater than or equal to 4000 gallons.

   iii. SR 3.1.7.3 was added to perform a verification that the SPB concentration is greater than or equal to 8.0% by weight every 31 days and once within 24 hours after water or boron is added to the solution.

2. Primary Containment Isolation Valves (PCIVs) (TS 3.6.1.3)

   Changes to the main steam isolation valve (MSIV) leakage rate limits (TS SR 3.6.1.3.10) were previously approved for BFN Units 2 and 3 by Amendment Nos. 263 and 223, respectively, dated March 14, 2000 (Reference 15). A similar change for BFN Unit 1 was approved by Amendment No. 261 on September 27, 2006 (Reference 16). The radiological consequences based on the MSIV leakage limits under EPU conditions and the acceptability of the alternate leakage treatment (ALT) system for BFN Units 1, 2, and 3 were previously approved by the NRC staff as documented in the safety evaluation for Amendment Nos. 251, 290, and 249, respectively, (full-scope implementation of AST), dated September 27, 2004 (Reference 8).

3. RCS Pressure and Temperature (P/T) Limits (TS 3.4.9)

The Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits for BFN Units 1, 2, and 3 have been developed for EPU conditions and have been submitted to the NRC for approval as follows:

a. The BFN Unit 1 change was submitted to the NRC on December 18, 2013 (Reference 10) and approved in Amendment No. 287 on February 2, 2015 (Reference 13).

b. The BFN Unit 2 change was submitted to the NRC on June 19, 2014 (Reference 11) and approved in Amendment No. 314 on June 2, 2015 (Reference 17).

c. The ~~current~~ BFN Unit 3 change ~~P/T Limits were submitted to the NRC on September 18, 2003 (Reference 20) and are based on EPU conditions. NRC approved the current P/T Limits in Amendment 247 on March 10, 2004 (Reference 14). A revision to the BFN Unit 3 P/T limits~~ was submitted to the NRC on January 27, 2015, ~~to address operation beyond the period of the original 40-year operating license and is currently under NRC review~~ (Reference 12) and approved in Amendment No. 278 on January 7, 2016 (Reference 21). ~~These revised P/T limits have also been developed for EPU conditions.~~

2.6 Elimination of Containment Accident Pressure Credit

TVA is eliminating the need to rely on containment accident pressure (CAP) for specific event sequences associated with the proposed EPU. The elimination of the need for CAP credit is consistent with guidance contained in NRC Regulatory Guide (RG) 1.82, "Water Source for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (Reference 7). RG 1.82 recommends minimizing reliance on CAP credit to demonstrate adequate pump net positive suction head (NPSH) margins to the extent possible. Therefore, TVA has pursued elimination of reliance on CAP credit for BFN's EPU. The elimination of CAP credit is accomplished through plant modifications, analysis methodology changes, and revised safety analyses, and therefore, is integrated into the EPU technical basis. Refer to PUSAR Section 2.6.5.2 and LAR Attachment 39 for additional information regarding the elimination of CAP credit at BFN.

2.7 Plant Modifications

TVA is also making various physical plant changes required to support EPU conditions. Some modifications are necessary to support efficient electrical output of the units to maximize the benefits from the increase in RTP. Other modifications are necessary to support or compensate for changes in analysis using plant-specific EPU parameters. These modifications were evaluated for impact to the Technical Specifications and Bases, as noted by proposed changes in Section 2.0 above and the technical evaluation in Section 3.0 of this attachment. The Updated Final Safety Analysis Report (UFSAR) will be updated, as required when the EPU application is approved. The detailed description of the plant changes are addressed in Attachment 47 of this amendment request. The steam dryer is also being replaced. Technical information regarding the Replacement Steam Dryer (RSD) is located in Attachments 40 (proprietary) and 41 (non-proprietary).

2.8 Methodology Changes

Containment Accident Pressure (CAP) Credit Elimination Methodologies

The following changes to analytical assumptions are applied to the EPU analyses for design basis and special events:

1. Rather than using limiting values, nominal (or realistic) values are used in the analysis of special events (beyond design basis events) that include Station Blackout, ATWS and Fire events. This change is discussed in more detail in PUSAR Section 2.6.5.2.

2. Credit is taken for passive heat sinks in the suppression pool temperature response to certain design basis and special events. Although this change in methodology is applicable to the containment analyses, the resultant change in suppression pool temperature response is a key input towards elimination of CAP credit in NPSH analyses. Heat sinks are also credited in minimizing containment pressure

The evaluation of ECCS pump NPSH margin for specified design basis and special events is described in PUSAR Section 2.6.5.2. Additional event-specific details, including the methodologies used to perform the analyses, are provided in the following PUSAR sections:

- 2.3.5 Station Blackout
- 2.5.1.4.2 Fire Event
- 2.6.1.1 Containment Pressure and Temperature Response
- 2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents
- 2.8.5.7 Anticipated Transients Without Scram

3.0 TECHNICAL EVALUATION

3.1 Renewed Operating Licensing and Technical Specification Changes

The following OL and TS Changes are required to support EPU and the associated elimination of reliance on CAP credit in the licensing basis. In addition, all actions that will not be completed prior to EPU power ascension will be contained in license conditions.

1. Renewed Operating License Paragraph 2.C(1)

   The proposed change supports an increase in the authorized Maximum Power Level from 3458 MWt to 3952 MWt. The analyses and evaluations presented in the Attachments to this license amendment request support this proposed change.

2. Renewed Operating License Paragraph 2.C(4) for Units 2 and 3 and 2.C(6) for Unit 1

   The proposed change provides requirements for monitoring and evaluating potential adverse flow effects as a result of power uprate operation, including verifying the continued structural integrity of the replacement steam dryers. Also, during the first two scheduled refueling outages after reaching EPU conditions, requirements are provided for performing visual inspections of the replacement steam dryers. Refer to LAR Attachment 2 for additional information.

3. Definitions (TS 1.1)

   The proposed change revises the definition of RATED THERMAL POWER (RTP) from the current value of 3458 MWt to 3952 MWt. The analyses and evaluations presented in the Attachments to this license amendment request support this request. This change is reflective of the Renewed Operating License change discussed above.

4. Reactor Core Safety Limits (TS 2.1.1)

   The current TS 2.1.1.1 states that thermal power shall be less than or equal to 25% RTP when the reactor steam pressure is less than 785 psig or core flow is less than 10% rated core flow. The proposed change revises the less than or equal to 25% RTP limit to less than or equal to 23% RTP. The revision to the RTP limit is based on the fuel thermal limit monitoring threshold. (Refer to PUSAR Section 2.8.2.1.1.)

   The current TS 2.1.1.2 for Unit 3 states that the Safety Limit Minimum Critical Power Ratio (SLMCPR) shall be greater than or equal to 1.09 for two recirculation loop operation or greater than or equal to 1.11 for single loop operation. A proposed change to the BFN Unit 3 SLMCPR was submitted to the NRC on March 6, 2015 (Reference 19). This proposed change is also reflected in Attachments 2 and 3. The proposed change modifies the TS 2.1.1.2 value of the SLMCPR for two-loop operation to 1.06 and the SLMCPR for single loop operation to 1.08. The revised SLMCPR values reflect a reduction from the current values, supported by the application of the SAFLIM3D methodology previously approved for BFN. In support of the proposed TS change, AREVA has performed a BFN Unit 3 specific evaluation based on a representative Cycle 19 core design to demonstrate that the proposed SLMCPR values are conservative for EPU conditions (Refer to Attachment 32).

5. Standby Liquid Control (SLC) System (TS 3.1.7)

   The following SRs are being revised to support EPU:

   a. The proposed change to TS SR 3.1.7.5 revises the value of the minimum quantity of Boron-10 (B-10) in the SLC System solution tank from 186 pounds to 203 pounds for BFN Units 2 and 3. The requirement for 203 pounds of B-10 reflects the change in the required boron concentration. (Refer to

PUSAR Section 2.8.4.5.1.) As previously stated, the BFN Unit 1 change to the minimum quantity of B-10 (greater than or equal to 203 pounds) in SR 3.1.7.5 was approved by the NRC on March 6, 2007, by Amendment 269, "Five Percent Uprate," (Reference 1) to the Renewed Facility Operating License for BFN Unit 1.

b. The SLC system is required to inject borated water solution into the reactor pressure vessel to control reactor power in the event of an ATWS event in accordance with the requirements of 10 CFR 50.62(c)(4). By meeting the conditions specified in SR 3.1.7.6, the SLC System provides a combination of flow capacity and B-10 content equivalent in control capacity to 86 gpm of 13 weight percent (wt. %) natural sodium pentaborate solution.

The proposed change to TS SR 3.1.7.6 provides a more rapid shutdown of the reactor during an ATWS event and considers the increase in heat generated due to EPU. The reduction in the peak suppression pool temperature for EPU is a result of modified plant parameters in the ATWS safety analysis, including an increase in SLC System B-10 enrichment, an increase in the credited SLC storage tank boron concentration, and an increase in the credited SLC flow rate. As a result, the total integrated heat load added to the suppression pool during an ATWS event is reduced, which provides additional NPSH margin for the credited ECCS pumps.

For EPU, the B-10 enrichment is increased to a nominal 94 atom-percent. However, the equation specified in SR 3.1.7.6 can be satisfied by a lower B-10 enrichment by increasing the other variables (i.e., boron concentration and/or pump flow rate).

For EPU, the equivalency requirement can be demonstrated if the following relationship is satisfied:

$$\frac{(C)(Q)(E)}{(8.7 \text{ wt. \%})(50 \text{ gpm})(94 \text{ atom \%})} \geq 1$$

where,

C = sodium pentaborate solution concentration (wt. %)
Q = pump flow rate (gpm)
E = B-10 enrichment (atom % B-10)

If the result of the above equation is numerically greater than or equal to one, the SLC System is capable of shutting down the reactor with significant margin to the acceptance criteria for suppression pool temperature. (Refer to PUSAR Section 2.8.4.5 for an evaluation of the SLC System for EPU, and PUSAR Section 2.8.5.7 for the ATWS evaluation under EPU conditions.)

6. <u>Average Planar Linear Heat Generation Rate (APLHGR) (TS 3.2.1)</u>

TS 3.2.1 APLHGR Applicability, Required Action B.1, and SR 3.2.1.1 Frequency include requirements associated with a thermal power limit of 25% RTP. The proposed change revises the 25% RTP to 23% RTP. The revision to the % RTP is based on the fuel thermal limit monitoring threshold. (Refer to FUSAR Section 2.8.2.1.2.)

7. Minimum Critical Power Ratio (MCPR) (TS 3.2.2)

   TS 3.2.2 MCPR Applicability, Required Action B.1 and SR 3.2.2.1 Frequency include requirements associated with a thermal power limit of 25%. The proposed change revises the 25% RTP to 23% RTP. The revision to the % RTP is based on the fuel thermal limit monitoring threshold. (Refer to FUSAR Section 2.8.2.1.2.)

8. Linear Heat Generation Rate (LHGR) (TS 3.2.3)

   TS 3.2.3 LHGR Applicability, Required Action B.1 and SR 3.2.3.1 Frequency include requirements associated with a thermal power limit of 25%. The proposed change revises the 25% RTP to 23% RTP. The revision to the % RTP is based on the fuel thermal limit monitoring threshold. (Refer to FUSAR Section 2.8.2.1.2.)

9. Reactor Protection System (RPS) Instrumentation (TS 3.3.1.1)

   The following Actions and SRs are being revised to support the EPU:

   a. The proposed change revises the RTP level value of TS 3.3.1.1 Required Action E.1 from 30% RTP to 26% RTP for arming the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions. Rescaling the % RTP maintains the same absolute thermal power level that was evaluated and approved for CLTP. (Refer to PUSAR Section 2.4.1.3.2 and Table 2.4-1.)

   b. The proposed change revises the Average Power Range Monitor (APRM) channel check RTP thermal monitoring threshold value of TS SR 3.3.1.1.2 and the associated Note from 25% RTP to 23% RTP. The revision to the % RTP is based on the fuel thermal limit monitoring threshold. (Refer to FUSAR Section 2.8.2.1.2.)

   c. The proposed change revises the RTP level value for arming the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions in TS SR 3.3.1.1.15 from 30% RTP to 26% RTP . Rescaling the % RTP maintains the same absolute thermal power level that was evaluated and approved for CLTP. (Refer to PUSAR Section 2.4.1.3.2 and Table 2.4-1.)

   d. TS SR 3.3.1.1.17 to the Reactor Protection System (RPS) Instrumentation ensures the OPRM Upscale Function will not be inadvertently bypassed in the region of power and flow operation if thermal hydraulic oscillations occur. Entry into this region is indicated by APRM Simulated Thermal Power ≥ 23% RTP and recirculation drive flow < 60% of rated flow. The proposed change revises the SR RTP level value from 25% RTP to 23% RTP to maintain the same absolute thermal power level that was previously approved for CLTP. The revision to the RTP is based on the fuel thermal limit monitoring threshold. (Refer to FUSAR Sections 2.8.2.1.2 and 2.8.3.1.1.)

e. The proposed change revises the Allowable Value of Table 3.3.1.1-1, Function 2.a, APRM Neutron Flux - High (Setdown), from less than or equal to 15% RTP to less than or equal to 13% RTP. Rescaling the % RTP maintains the same absolute thermal power level authorized for CLTP in terms of megawatts thermal. The APRM Neutron Flux - High (Setdown) function is not credited in the accident or transient analysis. (Refer to PUSAR Section 2.4.1.3 and Table 2.4-1.)

f. The proposed change revises the Allowable Value of Table 3.3.1.1-1, Function 2.b, APRM Flow Biased Simulated Thermal Power - High, from ≤ 0.66 W + 66% RTP to ≤ 0.55 W + 65.5% RTP for two loop operation. The proposed change also revises Footnote (c) from [0.66 W + 66% - 0.66 Δ W] RTP to [0.55 W + 65.5% - 0.55 Δ W] RTP for single loop operation. The Allowable Value is based on the proposed changes in power level. The APRM Flow Biased Simulated Thermal Power - High function is not credited in the accident or transient analysis for BFN. The calculated value follows the methodology that the NRC approved by Amendment Nos. 257, 296, and 254 for BFN Units 1, 2, and 3, respectively, dated September 14, 2006. (Reference 9) (Refer to PUSAR Section 2.4.1.3 and Table 2.4-1.)

g. The proposed change revises the Applicable Modes or Other Specified Conditions of Table 3.3.1.1-1, Function 8, Turbine Stop Valve - Closure, from 30% RTP to 26% RTP. Rescaling the % RTP maintains the same absolute thermal power level that was evaluated and approved for CLTP. (Refer to PUSAR Section 2.4.1.3 and Table 2.4-1.)

h. The proposed change revises the Applicable Modes or Other Specified Conditions of Table 3.3.1.1-1, Function 9, Turbine Control Valve Fast Closure, Trip Oil Pressure - Low, from 30% RTP to 26% RTP. Rescaling the % RTP maintains the same absolute thermal power level that was evaluated and approved for CLTP. (Refer to PUSAR Section 2.4.1.3 and Table 2.4-1.)

10. <u>Feedwater and Main Turbine High Water Level Trip Instrumentation (TS 3.3.2.2)</u>

TS 3.3.2.2 Applicability and Required Action C.1 include requirements corresponding to thermal power limits of 25% RTP. The proposed change to LCO 3.3.2.2 revises the 25% RTP to 23% RTP. The revision to the % RTP is based on the fuel thermal limit monitoring threshold. (Refer to FUSAR Section 2.8.2.1.2.)

11. <u>End of Cycle Recirculation Pump Trip (EOC - RTP) Instrumentation (TS 3.3.4.1)</u>

TS 3.3.4.1 Applicability, Required Action C.1, and SR 3.3.4.1.2 include requirements corresponding to thermal power limits of 30% RTP. The proposed change to LCO 3.3.2.2 revises the 30% RTP to 26% RTP. Rescaling the % RTP maintains the same absolute thermal power level that was evaluated and authorized for CLTP. (Refer to PUSAR Section 2.4.1.3 and Table 2.4-1.)

12. <u>Jet Pumps (TS 3.4.2)</u>

TS SR 3.4.2.1, Note 2 states that the surveillance is not required to be performed until 24 hours after > 25% RTP. The 25% RTP in the note is being changed to

23% RTP. The revision to the % RTP is conservative, providing consistency with the other proposed changes from 25% RTP to 23% RTP that are associated with the fuel thermal limit monitoring threshold. (Refer to FUSAR Section 2.8.2.1.2.)

13. Containment Atmosphere Dilution (CAD) System (TS 3.6.3.1)

    Current TS SR 3.6.3.1.1 requires that at least 2500 gallons of liquid nitrogen be stored in each nitrogen storage tank. This volume is being increased to 2615 gallons as a result of the increased production rate of radiolytic gas following a postulated LOCA under EPU conditions. The revised TS value represents the analytical limit assumed in the analysis of the primary containment atmosphere following a postulated LOCA, and does not include allowance for potential nitrogen boil-off and tank level instrumentation inaccuracies. Implementing procedures will include the appropriate margin in tank volume to account for uncertainties. (Refer to FUSAR Section 2.6.4.)

14. Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS) (TS 3.7.1) [BFN Units 2 and 3 only]

    For BFN Units 2 and 3 only, TS 3.7.1 is being revised to remove requirements for the ultimate heat sink (UHS), which are included in TS 3.7.2. Reference to the UHS is being deleted from TS 3.7.1 and the section title is being changed to "Residual Heat Removal Service Water (RHRSW) System." These changes will make TS 3.7.1 alike for all three BFN units. Specifically:

    a. The page headings for TS 3.7.1 is being changed from "RHRSW System and UHS" to "RHRSW System." TS requirements for the UHS are contained in TS 3.7.2.

    b. TS LCO 3.7.1 is being revised to remove the requirement for the UHS to be OPERABLE in MODES 1, 2, and 3. This requirement is redundant and already included in TS LCO 3.7.2.

    c. TS LCO 3.7.1 ACTION G is being revised to remove the requirement to be in MODE 3 within 12 hours and MODE 4 within 36 hours when the UHS is inoperable. This requirement is redundant and already included in TS LCO 3.7.2 ACTION B.

    d. TS SR 3.7.1.2 and Figure 3.7.1-1 are being deleted because there is no longer a restriction for the UHS average water temperature to be in accordance with the limits specified in Figure 3.7.1-1. When the average water temperature of the UHS is at or below 95°F, there is no longer a need to make any reduction in rated thermal power for the UHS to be OPERABLE. The provisions of TS SR 3.7.1.2 and Figure 3.7.1-1 are not contained in the BFN Unit 1 TS.

    The service water and UHS temperature limit for all three BFN units is specified in TS SR 3.7.2.1 as less than or equal to 95°F. The EPU design basis analyses for design basis events, including the long term primary containment response after a design basis LOCA, assume a UHS temperature equal to 95°F.

The evaluation supporting this change is described in PUSAR Sections 2.5.3.4 and 2.6.5.1, applies to the UHS service water temperature for all three BFN units, and provides the basis for the revised service water temperature limit used in the safety analyses.

15. Emergency Equipment Cooling Water (ECCS) System and Ultimate Heat Sink (UHS) (TS 3.7.2) [BFN Units 2 and 3 only]

    The Note referring to TS SR 3.7.2.1 for additional requirements related to the UHS in the BFN Units 2 and 3 TS is being deleted. This note is being deleted because the UHS requirements in TS 3.7.1 are being deleted. (Refer to PUSAR Section 2.5.3.4.)

16. Main Turbine Bypass System (TS 3.7.5)

    TS 3.7.5 Applicability and Required Action B.1 include requirements corresponding to thermal power limits of 25% RTP. The proposed change revises the 25% RTP to 23% RTP. The revision to the % RTP is conservative, providing consistency with the other proposed changes from 25% RTP to 23% RTP that are associated with the fuel thermal limit monitoring threshold. (Refer to FUSAR Section 2.8.2.1.2.)

17. Primary Containment Leakage Rate Testing Program (TS 5.5.12)

    The peak calculated containment internal pressure for the design basis accident (DBA) loss of coolant accident (P_a) is being revised from 50.6 psig to 49.1 psig for BFN Units 2 and 3. For BFN Unit 1, the peak calculated containment internal pressure for the DBA loss of coolant accident (P_a) is being revised from 48.5 psig to 49.1 psig. The revised event initial conditions for EPU, the selection of mass and energy inputs for Units 2 and 3 to be consistent with the current licensing basis for Unit 1, and uniform modeling in the containment analysis for all three BFN units account for these changes. The same analytical inputs and assumptions are now used in the containment analysis for all three BFN units. (Refer to PUSAR Sections 2.2.4.1 and 2.6.3.1 and Table 2.6-1.)

3.2 Elimination of Containment Accident Pressure Credit

The current licensing basis for all three BFN units includes credit for CAP in determining available NPSH for ECCS pumps. As part of the proposed EPU, TVA is eliminating CAP credit assumptions in the BFN safety analyses. The elimination of CAP credit from the licensing basis is accomplished through system modifications and analytical assumption changes that are factored into the safety analyses.

The increase in core power due to EPU and the increased reactor steam flow rates were examined for the effect on heat loads to the suppression pool following postulated events. In order to maintain or improve suppression pool temperature margin, several changes are being made to the licensing basis. As discussed in Section 3.0, this change to the licensing basis includes changes that rely on more realistic analytical assumptions.

As part of the EPU, TVA is proposing a modification to increase the isotopic B-10 enrichment provided by the SLC System. Raising the boron-10 enrichment for EPU increases the rate of negative reactivity inserted by the SLC system and results in a faster shut down of the reactor during the ATWS event. This results in a reduced heat load input into the suppression pool; therefore, the suppression pool temperature is lower. SLC system shutdown requirements will continue to be evaluated on a cycle-specific basis using NRC-approved methods.

Containment heat removal and suppression pool temperature response was evaluated in accordance with the guidelines in NRC-approved licensing topical reports using NRC-approved methodologies. (Refer to PUSAR Section 2.6.5). The revised containment safety analysis, when combined with reduced heat exchanger fouling resistance (discussed in Attachment 39), decreases peak suppression pool temperatures further below current design limits.

The acceptability of ECCS pump NPSH based on the containment analysis suppression pool temperature response and without CAP credit is provided in PUSAR Section 2.6.5.2. NPSH evaluations are described in PUSAR Section 2.6.5.2 for the following events:

- Large Break LOCA Short-Term Phase
- Large Break LOCA Long-Term Phase
- Small Break LOCA
- Loss of Residual Heat Removal Shutdown Cooling
- Stuck Open Relief Valve with Reactor Pressure Vessel Isolation
- Fire Event
- Station Blackout
- ATWS
- Shutdown of the Non-Accident Unit Following Loss of Offsite Power and Accident in the Accident Unit

The ECCS pumps have been analyzed for plant-specific conditions and have sufficient NPSH margin to perform satisfactorily under postulated accident and transient conditions.

3.3 Plant Modifications Supporting Extended Power Uprate

The evaluations performed to support EPU identified that changes are required to certain safety and non-safety related systems, including minor equipment changes, replacements, and setpoint or alarm point changes. These changes will be made in accordance with the requirements of 10 CFR 50.59, "Changes, tests, and experiments," and do not require prior NRC approval through this EPU License

Amendment Request. Some modifications have been implemented, as reflected in Attachment 47. The remaining modifications will be implemented prior to escalation above CLTP. Attachment 47 provides a status and listing of these modifications.

Any aspects of the modifications (i.e., associated TS or methodology changes) that require prior NRC approval are summarized in this Enclosure, in the TS changes, or methodology change sections.

Modifications that specifically address CAP credit elimination are listed in Section 3.2 of this Enclosure.

Additionally, the steam dryer in each unit is being replaced. Technical information regarding the replacement steam dryers is located in Attachment 40 (proprietary) and Attachment 41 (non-proprietary). The replacement of the steam dryers have been addressed for EPU and have no impact on other evaluations contained in the PUSAR. Additional discussion is provided below in Section 3.4.

3.4 Replacement Steam Dryers

TVA evaluated the existing BFN original equipment manufacturer steam dryers and determined that the steam dryers would not be suitable for EPU conditions without modifications. Therefore, TVA is replacing the existing BFN original equipment manufacturer steam dryers with replacement steam dryers manufactured by GEH. Refer to Attachments 40 (proprietary) and 41 (non-proprietary) for additional information regarding the replacement steam dryer design and analyses.

4.0 REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

TVA has determined that the proposed changes do not require any exemptions or relief from regulatory requirements and do not affect conformance with any General Design Criterion (GDC) differently than described in the UFSAR.

NEDC-33860P, "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate," is provided as Attachment 6 (proprietary) and Attachment 7 (non-proprietary). Each PUSAR section contains a regulatory evaluation that describes the relevant regulatory requirements and criteria. A technical evaluation is also included that explains the EPU changes and how the applicable regulatory requirements are met.

The PUSAR follows the format and guidance outlined in RS-001, "Review Standard for Extended Power Uprates," Revision 0 (Reference 6) to the extent that the review standard is consistent with the BFN design basis. For differences between plant-specific design bases and RS-001 regulatory evaluation sections, the corresponding PUSAR regulatory evaluation section was revised to reflect the BFN design basis.

The proposed EPU is based on the approaches described in the following documents:

- GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A (CLTR), Revision 4, dated July 2003

- GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A (ELTR1), dated February 1999

- GE Nuclear Energy, "Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A (ELTR2), dated February 1999

The PUSAR uses GEH GE14 fuel as the principal reference fuel type for the evaluation of the impact of EPU. However, the BFN units will utilize AREVA ATRIUM 10XM fuel, with some legacy ATRIUM 10 fuel, under EPU conditions. Therefore, the AREVA Fuel Uprate Safety Analysis Report (FUSAR) for Browns Ferry Units 1, 2, and 3 (Attachments 8 and 9) and fuel related reports are provided to supplement the PUSAR and address the effect of EPU conditions on the AREVA fuel in the BFN units.

The fuel-related reports are included in Attachments 10 through 38 and are as follows:

- ANP-3377, Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU)

- ANP-3378, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel (EPU)

- ANP-3384, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10 Fuel (EPU)

- ANP-3342, Browns Ferry EPU (120% OLTP) Equilibrium Fuel Cycle Design

- ANP-3372, Browns Ferry Unit 3 Cycle 19 EPU (120% OLTP) LAR Reference Fuel Cycle Design

- ANP-3404, Browns Ferry Unit 3 Cycle 19 Representative Reload Analysis at Extended Power Uprate

- ANP-3343, Nuclear Fuel Design Report Browns Ferry EPU (120% OLTP) Equilibrium Cycle ATRIUM 10XM Fuel

- ANP-3386, Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM 10XM Fuel Assemblies

- ANP-3385, Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM 10 Fuel Assemblies

- ANP-3388, Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Extended Power Uprate

- ANP-3327, Evaluation of AREVA Fuel Thermal-Hydraulic Performance for Browns Ferry at EPU

- FS1-0019629/30, Browns Ferry Unit 3 Cycle 19 MCPR Safety Limit Analysis With SAFLIM3D Methodology
- ANP-2860 Revision 2, Supplement 2, Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information, Extension for Use of ATRIUM 10XM Fuel for Extended Power Uprate
- ANP-2637, Boiling Water Reactor Licensing Methodology Compendium
- ANP-3409, Fuel-Related Emergent Regulatory Issues

4.2 Precedent

The following approved extended power uprates were reviewed for precedent to the BFN request:

Precedent Relating to Extended Power Uprate (EPU)	
Precedent	Relevance and Deltas to EPU Proposed Licensing Action:
Peach Bottom Atomic Power Station 2 and 3 8/25/14 (ML14133A046) Amendment Nos. 293 and 296, respectively	Extended Power Uprate amendment increasing reactor core thermal power approximately 12.4%. Peach Bottom Atomic Power Station Units 2 and 3 are BWR 4s with Mark I containments similar to BFN Units 1, 2, and 3.
Grand Gulf 1, 07/18/12 (ML121210020) Amendment No. 191	Extended Power Uprate amendment increasing reactor core thermal power approximately 15%. Grand Gulf is a BWR 6 with a Mark III containment compared to BFN Units 1, 2, and 3, BWR 4s with Mark I containments.
St Lucie 1, 07/09/12 (ML12191A220) Amendment No. 213	Extended Power Uprate and Measurement Uncertainty Recapture (MUR) amendment increasing total reactor core thermal power approximately 11.9% (10.0% EPU and 1.7% MUR). St Lucie is a Pressurized Water Reactor (PWR) and of dissimilar design compared to BFN's BWR design.
Turkey Point 3 and 4, 06/15/12 (ML11293A359) Amendment Nos. 249 and 245, respectively	Extended Power Uprate and MUR amendment increasing total reactor core thermal power approximately 15% (13.0% EPU and 1.7% MUR). Turkey Point Units 3 and 4 are PWRs and of dissimilar design compared to BFN's BWR design.

Precedent Relating to Extended Power Uprate (EPU)	
Precedent	Relevance and Deltas to EPU Proposed Licensing Action:
Nine Mile Point 2, 12/22/11 (ML113300040) Amendment No. 140	Extended Power Uprate amendment increasing reactor core thermal power approximately 15%. Nine Mile Point 2 is a BWR 5 with a Mark II containment compared to BFN Units 1, 2, and 3 - BWR 4 with a Mark I containment.
Hope Creek, 05/14/08 (ML081230540) Amendment No. 174	Extended Power Uprate amendment increasing reactor core thermal power approximately 15%. Hope Creek is a BWR 4 with a Mark I containment similar to BFN Units 1, 2, and 3.
Susquehanna 1 and 2, 01/30/08 (ML081050530) Amendment Nos. 246 and 224, respectively	Extended Power Uprate amendment increasing reactor core thermal power approximately 13%. Susquehanna Units 1 and 2 are BWR 4s with Mark II containments. BFN Units 1, 2, and 3 are similar BWR 4 plants and vessel design with a difference in containment designs (Mark I versus Mark II).

Precedent Relating to Replacement Steam Dryers (RSD)	
Precedent	Relevance and Deltas to EPU Proposed Licensing Action:
Peach Bottom Atomic Power Station 2 and 3 8/25/14 (ML14133A046) Amendment 293 and 296, respectively	Extended Power Uprate increasing reactor core thermal power approximately 12.4%. The Peach Bottom Atomic Power Station Units 2 and 3 application included RSDs. The Peach Bottom Atomic Power Station Units 2 and 3 RSDs are of Westinghouse design.
Grand Gulf 1, 07/18/12 (ML121210020) Amendment 191	Extended Power Uprate increasing reactor core thermal power approximately 15%. The Grand Gulf application included an RSD. The Grand Gulf RSD is of GEH design.
Susquehanna 1 and 2, 01/30/08 (ML081050530) Amendment Nos. 246 and 224, respectively	Extended Power Uprate amendment increasing reactor core thermal power approximately 13%. The Susquehanna Units 1 and 2 application included RSDs. The Susquehanna RSDs are of GEH design.

Precedent Relating to Elimination of Containment Accident Pressure (CAP) Credit	
Precedent and Date	Relevance and Deltas to EPU Proposed Licensing Action:
Peach Bottom Atomic Power Station 2 and 3, 8/25/14 (ML14133A046) Amendment 293 and 296, respectively	Extended Power Uprate amendment increasing reactor core thermal power approximately 12.4% and eliminating credit for CAP to ensure adequate ECCS pump NPSH.

4.3 Significant Hazards Consideration

The Tennessee Valley Authority (TVA) is submitting an amendment request to Renewed Facility Operating Licenses DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3, respectively, Technical Specifications (TS) and licensing bases. The proposed change revises the TS and licensing bases to support safe operation of BFN Units 1, 2, and 3 at an increased licensed reactor thermal power (RTP) of 3952 megawatts thermal (MWt); this is approximately 20% above the original licensed thermal power (OLTP) and approximately 14% above the current licensed thermal power (CLTP) of 3458 MWt. For BFN Units 2 and 3 only, because of proposed Extended Power Uprate (EPU) conditions, TVA is requesting a change in the maximum service water temperature and ultimate heat sink water temperature. Also, as part of modifications supporting EPU, TVA is proposing elimination of the credit for containment accident pressure in certain analyses.

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1) Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change increases the maximum authorized core power level for BFN from the current licensed thermal power (CLTP) of 3458 MWt to 3952 MWt. Evaluations and analysis of the nuclear steam supply system (NSSS) and balance of plant (BOP) structures, systems, and components (SSCs) that could be affected by the power uprate were performed in accordance with the approaches described in the following.

- GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A (CLTR), Revision 4, dated July 2003
- GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A (ELTR1), dated February 1999

- GE Nuclear Energy, "Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A (ELTR2), dated February 1999

The Power Uprate Safety Analysis Report (PUSAR) summarizes the results of safety evaluations performed that justify uprating the licensed thermal power at BFN. The PUSAR uses GEH GE14 fuel as the principal reference fuel type for the evaluation of the impact of EPU. However, the BFN units will utilize AREVA ATRIUM 10XM fuel, with some legacy ATRIUM 10 fuel, under EPU conditions. Therefore, the AREVA Fuel Uprate Safety Analysis Report (FUSAR) for Browns Ferry Units 1, 2, and 3 and fuel related reports are provided to supplement the PUSAR and address the impact of EPU conditions on the AREVA fuel in the BFN units. The AREVA analyses contained in the FUSAR have provided disposition of the critical characteristics of the GE14 fuel and have been shown to bound ATRIUM 10XM and ATRIUM 10 fuel.

The fuel-related reports are as follows:

- ANP-3377, Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU)
- ANP-3378, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel (EPU)
- ANP-3384, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10 Fuel (EPU)
- ANP-3342, Browns Ferry EPU (120% OLTP) Equilibrium Fuel Cycle Design
- ANP-3372, Browns Ferry Unit 3 Cycle 19 EPU (120% OLTP) LAR Reference Fuel Cycle Design
- ANP-3404, Browns Ferry Unit 3 Cycle 19 Representative Reload Analysis at Extended Power Uprate
- ANP-3343, Nuclear Fuel Design Report Browns Ferry EPU (120% OLTP) Equilibrium Cycle ATRIUM 10XM Fuel
- ANP-3386, Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM 10XM Fuel Assemblies
- ANP-3385, Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM 10 Fuel Assemblies
- ANP-3388, Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Extended Power Uprate
- ANP-3327, Evaluation of AREVA Fuel Thermal-Hydraulic Performance for Browns Ferry at EPU

- FS1-0019629/30, Browns Ferry Unit 3 Cycle 19 MCPR Safety Limit Analysis With SAFLIM3D Methodology

- ANP-2860 Revision 2, Supplement 2, Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information, Extension for Use of ATRIUM 10XM Fuel for Extended Power Uprate

- ANP-2637, Boiling Water Reactor Licensing Methodology Compendium

- ANP-3409, Fuel-Related Emergent Regulatory Issues

The evaluations concluded that all plant components, as modified, will continue to be capable of performing their design function at the proposed uprated core power level.

The BFN licensing and design bases, including BFN accident analysis, were also evaluated for the effect of the proposed power increase. The evaluation concluded that the applicable analysis acceptance criteria continue to be met.

Power level is not an initiator of any transient or accident; it is used as an input assumption to equipment design and accident analyses. The proposed change does not affect the release paths or the frequency of release for any accident previously evaluated in the FSAR. SSCs required to mitigate transients remain capable of performing their design functions considering radiological consequences associated with the effect of the proposed EPU. The source terms used to evaluate the radiological consequences were reviewed and were determined to bound operation at EPU power levels. The results of EPU accident evaluations do not exceed NRC-approved acceptance limits.

The spectrum of postulated accidents and transients were reviewed and were shown to meet the regulatory criteria to which BFN is currently licensed. In the area of fuel and core design, the Safety Limit Minimum Critical Power Ratio (SLMCPR) and other Specified Acceptable Fuel Design Limits (SAFDLs) are still met. Continued compliance with the SLMPCR and other SAFDLs is confirmed on a cycle specific basis consistent with the criteria accepted by the NRC.

Challenges to the reactor coolant pressure boundary were evaluated at the EPU conditions of pressure, temperature, flow, and radiation and found to meet the acceptance criteria for allowable stresses. Adequate overpressure margin is maintained.

Challenges to the containment were also evaluated. The containment and its associated cooling system continue to meet applicable regulatory requirements. The calculated post event suppression pool temperatures remain within design limits, while ensuring adequate net positive suction head is maintained for required emergency core cooling system pumps.

Radiological releases were evaluated and found to be within the regulatory limits of 10 CFR 50.67, Accident Source Terms.

The modifications and methodology associated with the elimination of containment accident pressure credit do not change the design functions of the systems. By maintaining these functions, they do not significantly increase the probability or consequences of an accident previously evaluated.

The non-safety-related Replacement Steam Dryer (RSD) must function to maintain structural integrity and avoid generation of loose parts that may affect other SSCs. The RSD analyses demonstrate the structural integrity of the steam dryer is maintained at EPU conditions. Therefore, the RSD does not significantly increase the probability or consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change increases the maximum authorized core power level for BFN from the current licensed thermal power (CLTP) of 3458 MWt to 3952 MWt. An evaluation of the equipment that could be affected by the power uprate has been performed. No new accident scenarios or equipment failure modes were identified. The full spectrum of accident considerations was evaluated and no new or different kinds of accidents were identified. For BFN, the standard evaluation methods outlined in the CLTR, ELTR1, ELTR2, PUSAR, FUSAR, and fuel related reports were applied to the capability of existing or modified safety-related plant equipment. No new accidents or event precursors were identified.

All SSCs previously required for mitigation of a transient remain capable of fulfilling their intended design functions. The proposed increase in power does not adversely affect safety-related systems or components and does not challenge the performance or integrity of any safety-related systems. The change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than was previously evaluated. Operating at the proposed EPU power level does not create any new accident initiators or precursors.

The modifications and methodology associated with the elimination of containment accident pressure credit do not change the design functions of the systems. The systems are not accident initiators and by maintaining their current function they do not create the possibility of a new or different kind of accident.

The new RSD does not have any new design functions. RSD analyses demonstrate that the RSD will be capable of performing the design function of maintaining structural integrity. Therefore, there are no new or different kinds of accidents from those previously evaluated.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) **Does the proposed change involve a significant reduction in a margin of safety?**

Response: No.

Based on the analyses of the proposed power increase, the relevant design and safety acceptance criteria will be met without significant adverse effects or reduction in margins of safety. The analyses supporting EPU have demonstrated that the BFN SSCs are capable of safely performing at EPU conditions. The analyses identified and defined the major input parameters to the NSSS, and NSSS design transients, and evaluated the capability of the primary containment, NSSS fluid systems, NSSS and BOP components, as appropriate. Radiological consequences of design basis events remain within regulatory limits and are not increased significantly. The analyses confirmed that NSSS and BOP SSCs are capable of achieving EPU conditions without significant reduction in margins of safety, with the modifications discussed in this application.

Analyses have shown that the integrity of primary fission product barriers will not be significantly affected as a result of the power increase. Calculated loads on SSCs important to safety have been shown to remain within design allowables under EPU conditions for all design basis event categories. Plant response to transients and accidents do not result in exceeding acceptance criteria.

As appropriate, the evaluations that demonstrate acceptability of EPU have been performed using methods that have either been reviewed and approved by the NRC staff, or that are in compliance with regulatory review guidance and standards established for maintaining adequate margins of safety. These evaluations demonstrate that there are no significant reductions in the margins of safety.

Maximum power level is one of the inherent inputs that determine the safe operating range defined by the accident analyses. The Technical Specifications ensure that BFN is operated within the bounds of the inputs and assumptions used in the accident analyses. The acceptance criteria for the accident analyses are conservative with respect to the operating conditions defined by the Technical Specifications. The engineering reviews performed for the constant pressure EPU confirm that the accident analyses criteria are met at the revised maximum allowed thermal power of 3952 MWt. Therefore, the adequacy of the renewed Facility Operating License and Technical Specifications to maintain the plant in a safe operating range is also confirmed, and the increase in maximum allowable power level does not involve a significant decrease in a margin of safety.

The modifications and methodology associated with the elimination of containment accident pressure credit do not change the design functions within the applicable limits. The credit is associated with accident or event response and does not significantly affect accident initiators by maintaining their current functions and does not create the possibility of a new or different kind of accident. The proposed Technical Specifications associated with these modifications ensure that BFN is operated within the bounds of the inputs and assumptions used in the accident analyses.

The steam dryer is being replaced in order to ensure adequate margin to the established structural requirements is maintained. The new RSD does not have any new design functions and an analysis was performed to confirm it will be capable of maintaining its structural integrity. The power ascension test plan will verify that the RSD conservatively meets the vibration and stress requirements.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendments do not involve a significant hazard consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations evaluation is contained in Attachment 42, Supplemental Environmental Report. It concludes that EPU will not result in a significant change in non-radiological impacts on land use, water use, waste discharges, terrestrial and aquatic biota, transmission facilities, or social and economic factors, and will have no non-radiological environmental impacts other than those evaluated in the Supplemental Environmental Report. The Supplemental Environmental Report further concludes that EPU will not introduce any new radiological release pathways, will not result in a significant increase in occupational or public radiation exposures, and will not result in significant additional fuel cycle environmental impacts.

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Five Percent Power Uprate (TAC No. MD3048) (TS-431)," dated March 6, 2007; (ADAMS Accession No. ML063350404).

2. NRC Letter to TVA, "Issuance of Amendments RE: Power Uprate - Browns Ferry Plant, Units 2 and 3 - (TAC Nos. M99711 and M99712)," dated September 8, 1998; (ADAMS Accession No. ML020100022)

3. GE Nuclear Energy, "Constant Pressure Power Uprate," NEDC-33004P-A (CLTR), Revision 4, dated July 2003.

4. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A (ELTR1), dated February 1999.

5. GE Nuclear Energy, "Generic Evaluation of General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32523P-A (ELTR2), dated February 1999.

6. NRC, Office of Nuclear Reactor Regulation, Review Standard RS-001, "Review Standard for Extended Power Uprates," Revision 0, dated December 2003.

7. NRC Regulatory Guide (RG) 1.82, "Water Source for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 4, March 2012; (ADAMS Accession No. ML111330278).

8. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Issuance of Amendments Regarding Full-Scope Implementation of Alternative Source Term (TAC Nos. MB5733, MB5734, MB5735, MC0156, MC0157 and MC0158) (TS-405)," dated September 27, 2004; (ADAMS Accession No. ML042730028).

9. NRC Letter to 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)," dated September 14, 2006; (ADAMS Accession No. ML061680008).

10. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN), Unit 1 - Application to Modify Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits (BFN TS-484)," dated December 18, 2013; (ADAMS Accession No. ML13358A067).

11. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN), Unit 2 - Application to Modify Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits (BFN TS-491)," dated June 19, 2014; (ADAMS Accession No. ML14175A307).

12. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN), Unit 3 - Application to Modify Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits" (BFN TS-494)," dated January 27, 2015; (ADAMS Accession No. ML15040A698).

13. NRC Letter to TVA, “Browns Ferry Nuclear Plant, Unit 1 – Issuance of Amendment Revising Pressure and Temperature Limit Curves (TAC No. MF3260),” dated February 2, 2015; (ADAMS Accession No. ML14325A501).

14. ~~NRC Letter to TVA, "Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Regarding Pressure-Temperature Limit Curves (TAC Nos. MC0807 and MC0808)," dated March 10, 2004; (ADAMS Accession Nos. ML040480013, ML040750188, and ML040750194)~~Not used.

15. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Units 2 and 3 – Issuance of Amendments Regarding Limits on Main Steam Isolation Valve Leakage (TAC Nos. MA6405 and MA6406)," dated March 14, 2000; (ADAMS Accession No. ML003693000).

16. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Unit 1 – Issuance of Amendment Regarding Limits on Main Steam Isolation Valve Leakage (TAC No. MC3813)," dated September 27, 2006; (ADAMS Accession No. ML062210458).

17. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Unit 2 – Issuance of Amendment Revising Pressure and Temperature Limit Curves (TAC No. MF4303)," dated June 2, 2015; (ADAMS Accession No. ML15065A049).

18. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Technical Specification (TS) Change TS-478 Addition of Analytical Methodologies to TS 5.6.5 and Revision of TS 2.1.1.2 for Unit 2 (TAC Nos. MF0877, MF0878, and MF0879)," dated July 31, 2014; (ADAMS Accession No. ML14113A286).

19. TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN), Unit 3 – Application to Modify Technical Specification 2.1.1.2, Reactor Core Minimum Critical Power Ratio Safety Limits (TS-499)," dated March 6, 2015; (ADAMS Accession No. ML15090A436).

20. ~~TVA Letter to NRC, "Browns Ferry Nuclear Plant (BFN), Units 2 and 3 – Technical Specifications (TS) Change TS-441 Revision 1 – Update of Pressure-Temperature (P-T) Curves," dated September 18, 2003; (ADAMS Accession No. ML032750278).~~Not used.

21. NRC letter to TVA, "Browns Ferry Nuclear Plant, Unit 3 - Issuance of Amendment Regarding Modification of Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (CAC No. MF5659), dated January 7, 2016; (ADAMS Accession No. ML15344A321).

ENCLOSURE 5

Supplement to BFN EPU LAR, Attachment 6, NEDO-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate, Section 2.1.2 and Table 2.7-1

(Non-proprietary version)

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

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-9. Final GDC-31 is applicable to Browns Ferry as described in "Browns Ferry Nuclear Plant (BFN), Unit 1- Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-484)," dated December 18, 2013 (Reference 8), "Browns Ferry Nuclear Plant (BFN), Unit 2 - Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-491)," dated June 19, 2014 (Reference 9), and "Browns Ferry Nuclear Plant, Unit 3 - Application to Modify Technical Specification 3.4.9, 'RCS Pressure and Temperature (P/T) Limits' (BFN TS-494)," dated January 27, 2015 (Reference 10).

The Pressure-Temperature Limits and Upper Shelf Energy is described in Browns Ferry UFSAR Section 4.2, "Reactor Vessel and Appurtenances Mechanical Design," and the Bases to TS 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

In addition to the evaluations described in the Browns Ferry UFSAR, Browns Ferry's systems and components were evaluated for license renewal. 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 license renewal evaluations associated with Pressure-Temperature Limits and Upper-Shelf Energy are documented in NUREG-1843, Sections 4.2.1 and 4.2.5.

RCS Pressure and Temperature (P/T) Limits

The Reactor Coolant System (RCS) Pressure and Temperature (P/T) Limits have been developed for EPU conditions and have been submitted to the NRC for approval as follows:

a. The Browns Ferry Unit 1 change was submitted to the NRC on December 18, 2013 and approved in License Amendment No. 287 on February 2, 2015.
b. The Browns Ferry Unit 2 change was submitted to the NRC on June 19, 2014 and approved in License Amendment No. 314 on June 2, 2015.
c. The ~~current~~ Browns Ferry Unit 3 ~~P/T limits are based on EPU conditions and were approved by the NRC in License Amendment 247 on March 10, 2004. A revision to the~~

~~Browns Ferry Unit 3 P/T limits~~ was submitted to the NRC on January 27, 2015, ~~to address operation beyond the period of the original 40 year operating license and is currently under NRC review. These revised P/T limits have also been developed for EPU conditions.~~

and approved in License Amendment No. 278 on January 7, 2016.

Technical Evaluation

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 EPUs. Section 3.2.1 of the CLTR addresses the effect of EPU on Pressure-Temperature (P-T) Limits and Upper-Shelf Energy (USE). The results of this evaluation are described below.

As explicitly stated in Section 3.2.1 of the CLTR, EPU may result in a higher operating neutron flux at the vessel wall, consequently increasing the integrated flux over time (neutron fluence). The neutron fluence is recalculated using the NRC-approved GEH neutron fluence methodology (Reference 12). This method is consistent with Regulatory Guide (RG) 1.190 (Reference 13) and utilizes a more representative fluence than previous methods. Browns Ferry meets all CLTR dispositions.

AREVA fuel will be used at Browns Ferry when EPU is implemented; however, the basis for the RPV flux is the GEH analysis using GE14 fuel. AREVA independently evaluates the bounding nature of the GEH results for the peak flux values for RPV inner diameter, and internals (shroud diameter, top guide, core plate) in FUSAR Section 2.1.2.

The topics addressed in this evaluation are:

Topic	CLTR Disposition	Browns Ferry Result
Fracture Toughness	Plant Specific	Meets CLTR Disposition

The revised fluence is used to evaluate the vessel against the requirements of 10 CFR 50, Appendix G. The results of these evaluations indicate that:

(a) The reduction in USE, using Equivalent Margin methods, demonstrates that there is an equivalent margin of safety against fracture for RPV materials such that it will remain qualified with respect to 10 CFR 50 Appendix G criterion for the design life of the vessel. The maximum decrease in USE for the beltline plate materials is 16% ([[ ]]) for Unit 2 at 48 EFPY. The maximum decrease in USE for the beltline weld materials is 33.5% ([[ ]]) for Unit 1 at 38 EFPY. These values are provided in Tables 2.1-1a through 2.1-1c.

(b) The beltline material Reference Temperature of Nil-Ductility Transition (RT_{NDT}) remains below 200°F. The N-16 water level instrumentation nozzle is included in the evaluation.

(c) The Technical Specification P-T curves were revised to incorporate the methodology of the GEH P-T curve LTR (Reference 14) and the ISP Browns Ferry Unit 2 second surveillance capsule results. The fracture toughness evaluation included the effects of the N-16 water level instrumentation nozzle that occurs within the beltline region. The hydro test pressure for EPU is the minimum nominal operating pressure.

(d) The end of life (EOL) shift is increased, and consequently, results in an increase in the Adjusted Reference Temperature (ART), which is the initial RT_{NDT} plus the shift. These values are provided in Tables 2.1-2a through 2.1-2c.

(e) The EOL beltline circumferential weld material mean RT_{NDT} remains bounded by the requirements of Generic Letter (GL) 98-05 (Reference 15), BWRVIP-05 (References 16 and 17), and BWRVIP 74-A (Reference 18). This comparison is provided in Table 2.1-3.

(f) GEH P-T limit curves include an adjustment for the column of water in a full RPV. The Browns Ferry EPU is a constant pressure power uprate, which, by definition, does not change the pressure from that considered for CLTP. The pressure head for Browns Ferry for a full vessel is 31.6 psig.

(g) ISP plate and weld materials have been considered in development of the beltline ART as defined in BWRVIP-135. In accordance with the guidance from BWRVIP-135 and the methodology provided in RG 1.99 Revision 2 (Reference 19), the surveillance materials are considered in the development of the P-T limit curves for Units 1 and 2, but are not considered in the development of the P-T limit curves for Unit 3.

(h) The generic pressure test P-T limit curve is based on dimensions cited in NEDC-33178P-A, Revision 1 (Reference 14). GEH P-T limit curves are considered acceptable for plant-specific application when it is demonstrated that the plant-specific dimensions are bounded by the generic dimensions, as is the case for Browns Ferry Units 1, 2, and 3.

(i) Ferritic piping within the RCPB has not been replaced since plant start-up.

Therefore, Browns Ferry meets all CLTR dispositions for fracture toughness.

Conclusion

TVA has evaluated the effects of the proposed EPU on the P-T limits for the plant and addressed changes in neutron fluence and their effects on the P-T limits. ~~Revised P-T curves have been generated and submitted per 10 CFR 50.90 consistent with the guidance of the GE CLTR as a separate license amendment request.~~

Insert 1

2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCS).

INSERT 1

Revised P-T curves have been approved by the NRC per 10 CFR 50.90. As such, TVA concludes that the changes in neutron fluence and their effects on the P-T limits have been adequately addressed. TVA further concludes it has demonstrated the validity of the proposed PT limits for operation under the proposed EPU conditions. Based on this, TVA concludes the P-T limits will continue to meet the requirements of 10 CFR 50, Appendix G, and 10 CFR 50.60 and will enable Browns Ferry to continue to comply with the current licensing basis following implementation of the proposed EPU. Therefore, TVA finds the proposed EPU acceptable with respect to the proposed P-T limits.

zone.[1]

Table 2.7-1 EPU Effect on Ventilation Systems

bulk of the

System	EPU Effect
Turbine Building Ventilation System	Increases in process temperatures results in slight temperature increase. The turbine building is not an EQ ~~zone.~~ The design of the Turbine Building HVAC system is adequate to handle the increase in heat load.
Reactor Building Ventilation System	EPU does not result in significant temperature increases in areas of the Reactor Building. The expected increase in the Main Steam Tunnel is < 0.5°F, which is not significant. The temperature of the General Floor Area at El 639 will increase to a peak of 128.7°F for the most limiting Reactor Building room. The design of the HVAC system is adequate for EPU.
Drywell Ventilation System	EPU will not result in a significant increase in drywell heat load or area temperature increases (< 0.5°F). The drywell HVAC system is adequate to handle the small increase in heat load.
Radwaste Building Ventilation System	Negligible effect due to EPU.
Ventilation Systems for Miscellaneous Rooms and Buildings	Core Spray Pump room temperature will increase to a bounding 118.2°F. RHR Pump room temperature will increase to a bounding 131.3°F. The RHR heat exchanger rooms temperature will increase to a bounding 131.0°F. The bounding temperature is the Browns Ferry Unit 1, Unit 2, or Unit 3 highest temperature prediction for the respective room.
Control Room HVAC	Negligible effect due to EPU. No process temperature changes in the Control Room/Control Building.
Emergency Ventilating Systems	Negligible effect due to EPU. Some electrical operational loads may increase slightly, but will stay below design loads.

1. The Main Steam Tunnel is an EQ Zone and is located in the Turbine Building.

ENCLOSURE 6

Supplement to BFN EPU LAR, Attachment 4, Proposed Technical Specification Bases Changes (Markups)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak drywell temperature does not exceed the maximum allowable temperature of ~~336°F~~ 337°F (Ref. 2). Exceeding this temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.

Drywell air temperature satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 14.6.3.
2. ~~TVA Drawing 47E225-101-1~~.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate, dated September 2015.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

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(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 14.6.3.

2. ~~TVA Drawing 47E225-101-1.~~

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

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(continued)

BASES (continued)

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1. FSAR, Section 14.6.3.

2. ~~TVA Drawing 47E225-101-1.~~

NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate, dated September 2015.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

ENCLOSURE 7

Supplement to BFN EPU LAR, Attachment 5,
Retyped Proposed Technical Specification Bases Changes

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.
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(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 14.6.3.
2. NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate, dated September 2015.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

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BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

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(continued)

BASES (continued)

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1. FSAR, Section 14.6.3.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

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(continued)

BASES (continued)

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1. FSAR, Section 14.6.3.

2. NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate, dated September 2015.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

ENCLOSURE 8

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.

James F. Harrison

James F. Harrison
Vice President, Fuel Licensing
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Road
Wilmington, NC 28401
James.Harrison@ge.com

ENCLOSURE 9

EPRI Affidavit for NEDC-33860P, Revision 0

ELECTRIC POWER
RESEARCH INSTITUTE

AFFIDAVIT

RE: Request for Withholding of the Following Proprietary Information Included In:

GE Hitachi Nuclear Energy Report, "Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2 and 3 Extended Power Uprate". NEDC-33860P, Revision 0

I, Kurt Edsinger, being duly sworn, depose and state as follows:

I am the Director of PWR and BWR Materials at Electric Power Research Institute, Inc. whose principal office is located at 3420 Hillview Avenue, Palo Alto, CA. ("EPRI") and I have been specifically delegated responsibility for the above-listed report that contains EPRI Proprietary Information that is sought under this Affidavit to be withheld "Proprietary Information". I am authorized to apply to the U.S. Nuclear Regulatory Commission ("NRC") for the withholding of the Proprietary Information on behalf of EPRI.

EPRI Information is identified in solid underline inside double square brackets. An example of such identification is as follows:

[[This sentence is an example.{E}]]

Tables containing EPRI proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation {E} refers to the EPRI affidavit, which provides the basis for the proprietary determination.

EPRI requests that the Proprietary Information be withheld from the public on the following bases:

Withholding Based Upon Privileged And Confidential Trade Secrets Or Commercial Or Financial Information (see e.g., 10 C.F.R. § 2.390(a)(4)):

a. The Proprietary Information is owned by EPRI and has been held in confidence by EPRI. All entities accepting copies of the Proprietary Information do so subject to written agreements imposing an obligation upon the recipient to maintain the confidentiality of the Proprietary Information. The Proprietary Information is disclosed only to parties who agree, in writing, to preserve the confidentiality thereof.

b. EPRI considers the Proprietary Information contained therein to constitute trade secrets of EPRI. As such, EPRI holds the Information in confidence and disclosure thereof is strictly limited to individuals and entities who have agreed, in writing, to maintain the confidentiality of the Information.

c. The information sought to be withheld is considered to be proprietary for the following reasons. EPRI made a substantial economic investment to develop the Proprietary Information and, by prohibiting public disclosure, EPRI derives an economic benefit in the form of licensing royalties and other additional fees from the confidential nature of the Proprietary Information. If the Proprietary Information were publicly available to consultants and/or other businesses providing services in the electric and/or nuclear power industry, they would be able to use the Proprietary Information for their own commercial benefit and profit and without expending the substantial economic resources required of EPRI to develop the Proprietary Information.

d. EPRI's classification of the Proprietary Information as trade secrets is justified by the Uniform Trade Secrets Act which California adopted in 1984 and a version of which has been adopted by over forty states. The California Uniform Trade Secrets Act, California Civil Code §§3426 – 3426.11, defines a "trade secret" as follows:

> "'Trade secret' means information, including a formula, pattern, compilation, program device, method, technique, or process, that:
>
> (1) Derives independent economic value, actual or potential, from not being generally known to the public or to other persons who can obtain economic value from its disclosure or use; and
>
> (2) Is the subject of efforts that are reasonable under the circumstances to maintain its secrecy."

e. The Proprietary Information contained therein are not generally known or available to the public. EPRI developed the Information only after making a determination that the Proprietary Information was not available from public sources. EPRI made a substantial investment of both money and employee hours in the development of the Proprietary Information. EPRI was required to devote these resources and effort to derive the Proprietary Information. As a result of such effort and cost, both in terms of dollars spent and dedicated employee time, the Proprietary Information is highly valuable to EPRI.

f. A public disclosure of the Proprietary Information would be highly likely to cause substantial harm to EPRI's competitive position and the ability of EPRI to license the Proprietary Information both domestically and internationally. The Proprietary Information can only be acquired and/or duplicated by others using an equivalent investment of time and effort.

I have read the foregoing and the matters stated herein are true and correct to the best of my knowledge, information and belief. I make this affidavit under penalty of perjury under the laws of the United States of America and under the laws of the State of California.

Executed at 3420 Hillview Avenue, Palo Alto, CA. being the premises and place of business of Electric Power Research Institute, Inc.

Date: 9/14/2015

Kurt Edsinger

CALIFORNIA ALL-PURPOSE ACKNOWLEDGMENT **CIVIL CODE § 1189**

A notary public or other officer completing this certificate verifies only the identity of the individual who signed the document to which this certificate is attached, and not the truthfulness, accuracy, or validity of that document.

State of California)
County of Santa Clara)

On September 14, 2015 before me, Sandra K Tokunaga, Notary Public,
Date *Here Insert Name and Title of the Officer*

personally appeared Kurt Ward Edsinger
Name(s) of Signer(s)

who proved to me on the basis of satisfactory evidence to be the person(~~s~~) whose name(~~s~~) is/~~are~~ subscribed to the within instrument and acknowledged to me that he/~~she/they~~ executed the same in his/~~her/their~~ authorized capacity(~~ies~~), and that by his/~~her/their~~ signature(~~s~~) on the instrument the person(s), or the entity upon behalf of which the person(~~s~~) acted, executed the instrument.

SANDRA K. TOKUNAGA
Commission # 2051548
Notary Public - California
San Mateo County
My Comm. Expires Dec 12, 2017

I certify under PENALTY OF PERJURY under the laws of the State of California that the foregoing paragraph is true and correct.

WITNESS my hand and official seal.

Signature Sandra K Tokunaga
Signature of Notary Public

Place Notary Seal Above

OPTIONAL

Though this section is optional, completing this information can deter alteration of the document or fraudulent reattachment of this form to an unintended document.

Description of Attached Document

Title or Type of Document: ________ Document Date: ________

Number of Pages: ____ Signer(s) Other Than Named Above: ________

Capacity(ies) Claimed by Signer(s)

Signer's Name: ________
☐ Corporate Officer — Title(s): ________
☐ Partner — ☐ Limited ☐ General
☐ Individual ☐ Attorney in Fact
☐ Trustee ☐ Guardian or Conservator
☐ Other: ________
Signer Is Representing: ________

Signer's Name: ________
☐ Corporate Officer — Title(s): ________
☐ Partner — ☐ Limited ☐ General
☐ Individual ☐ Attorney in Fact
☐ Trustee ☐ Guardian or Conservator
☐ Other: ________
Signer Is Representing: ________