Proprietary Information
Withhold from Public Disclosure Under 10 CFR 2.390
This letter is decontrolled when separated from Enclosure 1

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

CNL-16-056

April 4, 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 9, Responses to Requests for Additional Information**

References:
1. Letter from TVA to NRC, CNL-15-169, "Proposed Technical Specifications (TS) Change TS-505 - Request for License Amendments - Extended Power Uprate (EPU)," dated September 21, 2015 (ML15282A152)
2. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Request for Additional Information Related to License Amendment Request Regarding Extended Power Uprate (CAC Nos. MF6741, MF6742, and MF6743)," dated March 15, 2016 (ML16062A072)

By the Reference 1 letter, Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) for the Extended Power Uprate (EPU) of Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3. The proposed LAR modifies the renewed operating licenses to increase the maximum authorized core thermal power level from the current licensed thermal power of 3458 megawatts to 3952 megawatts. During their technical review of the LAR, the Nuclear Regulatory Commission (NRC) identified the need for additional information. The Reference 2 letter provided NRC Requests for Additional Information (RAIs) from the Reactor Systems Branch (SRXB). The due date for the responses to these NRC RAIs is April 4, 2016. The enclosures to this letter provide the responses to each of the NRC RAIs provided in the Reference 2 letter.

Enclosure 1 provides the responses to the NRC RAIs SRXB-RAI 1 to 5, 8, 10, 12, 13, 16 to 18, and 20 from Reference 2. AREVA considers portions of the information provided in Enclosure 1 to this letter to be proprietary and, therefore, exempt from public disclosure pursuant to 10 CFR 2.390, Public inspections, exemptions, requests for withholding. An affidavit for withholding information, executed by AREVA, is provided in Enclosure 4. A non-proprietary version of the RAI and response is provided in Enclosure 2. Therefore, on behalf of AREVA, TVA requests that Enclosure 1 be withheld from public disclosure in accordance with the associated AREVA affidavit and the provisions of 10 CFR 2.390.

Enclosure 3 provides the response to the NRC RAIs SRXB-RAI 6, 7, 9, 11, 14, 15, 19, 21, 22, and 23 from Reference 2.

TVA has reviewed the information supporting a finding of no significant hazards consideration and the environmental consideration provided to the NRC in the Reference 1 letter. The supplemental information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration. In addition, the supplemental information in this submittal does not affect the bases for concluding that neither an environmental impact statement nor an environmental assessment needs to be prepared in connection with the proposed license amendment. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter, without the proprietary information, to the Alabama State Department of Public Health.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 4th day of April 2016.

Respectfully,

J. W. Shea

Digitally signed by J. W. Shea
DN: cn=J. W. Shea, o=Tennessee Valley Authority, ou=Nuclear Licensing, email=jwshea@tva.gov, c=US
Date: 2016.04.04 13:15:00 -04'00'

J. W. Shea
Vice President, Nuclear Licensing

Enclosures:

1. ANP-3473P, Response to RAIs for Browns Ferry Nuclear Plant EPU Submittal - Reactor Systems Branch Questions (Proprietary version)
2. ANP-3473NP, Response to RAIs for Browns Ferry Nuclear Plant EPU Submittal - Reactor Systems Branch Questions (Non-proprietary version)
3. Response to NRC RAIs SRXB-RAI 6, 7, 9, 11, 14, 15, 19, 21, 22, and 23
4. AREVA Affidavit

cc: See Page 3

cc:(Enclosures)

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
State Health Officer, Alabama Department of Public Health (w/o Enclosure 1)

Withhold from Public Disclosure Under 10 CFR 2.390

ENCLOSURE 1

ANP-3473P, Response to RAIs for Browns Ferry Nuclear Plant EPU Submittal - Reactor Systems Branch Questions

(Proprietary version)

ENCLOSURE 2

ANP-3473NP, Response to RAIs for Browns Ferry Nuclear Plant EPU Submittal - Reactor Systems Branch Questions

(Non-proprietary version)

Responses to RAIs for Browns Ferry Nuclear Plant EPU Submittal - Reactor Systems Branch Questions

ANP-3473NP
Revision 0

March 2016

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial issue

Contents

Tables

Figures

NOMENCLATURE

Abbreviation	Description
2PT	2 Recirculation Pump Trip (aka, 2RPT)
BFN	Browns Ferry Nuclear Plant
BSP	Backup Stability Protection
BWROG	Boiling Water Reactor Owners Group
CLTP	Current Licensed Thermal Power (3458 MWt)
DIVOM	Delta over Initial MCPR Versus Oscillation Magnitude
EOOS	Equipment Out Of Service
EOP	Emergency Operating Procedures
EPG	Emergency Procedure Guidelines
EPU	Extended Power Uprate (3952 MWt)
FWHOOS	Feedwater Heaters Out Of Service
HCOM	Hot Channel Oscillation Magnitude
HSBW	Hot Shutdown Boron Worth
kW/ft	Kilo-Watt per linear foot
LAR	License Amendment Request
$LHGRFAC_f$	Flow-Dependent Linear Heat Generation Rate Multipliers
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
MSBWP	Maximum Subcritical Banked Withdrawal Position
MWt	Mega-Watt thermal
OLMCPR	Operating Limit MCPR
OLTP	Original Licensed Thermal Power (3293 MWt)
OPRM	Oscillation Power Range Monitor
PLU	Power Load Unbalance
RMOV	Reactor Motor Operator Valve
RO	Reactor Operator
SAG	Severe Accident Guidelines
SLMCPR	Safety Limit MCPR
SLO	Single Loop Operation
SRO	Senior Reactor Operator

Abbreviation	Description
SS	Steady-State
TIPs	Traversing In-core Probes
TS	Technical Specifications
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
XB-hot-nat	Hot shutdown boron concentration requirement for naturally occurring boron (ppm)

1.0 Introduction

In Reference 1, the Tennessee Valley Authority (TVA) submitted a license amendment request (LAR) to modify the operating license for the Browns Ferry Nuclear Plant (BFN) for an extended power uprate (EPU). The amendment, if approved, would allow for an increase in the licensed reactor thermal power from the current licensed thermal power (CLTP) of 3458 MWt to a new licensed thermal power of 3952 MWt, approximately 120% of the original licensed thermal power (OLTP) of 3293 MWt. Reference 2 provided supplemental information including revised versions of some of the Enclosures supporting the LAR.

The NRC staff has determined that additional information is needed to complete their review of the BFN EPU LAR (Reference 3). This document contains only the responses to the Request for Additional Information (RAI) that contains AREVA content. Portions of some of the responses in this document have been provided by TVA. Where indicated, other responses are provided by TVA separate from this document.

References

1. Letter, JW Shea (TVA) to USNRC, “Proposed Technical Specifications Change to TS-505 – Request for License Amendments – Extended Power Uprate”, CNL-15-169, September 21, 2015. (Accession Number ML15282A152)

2. Letter, JW Shea (TVA) to USNRC, “Proposed Technical Specifications (TS) Change TS-505 – Request for License Amendments – Extended Power Uprate (EPU) – Supplement 2, MICROBURN-B2 Information”, CNL-15-250, December 15, 2015. (Accession Number ML15351A113)

3. Letter, FE Saba (USNRC) to JW Shea (TVA), “Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Request for Additional Information Related to License Amendment Request Regarding Extended Power Uprate (CAC NOs. MF6741, MF6742, and MF6743)”, March 15, 2016. (Accession Number ML16062A072)

4. FS1-0024528 Revision 1.0, *Browns Ferry EPU Low Flow Issues Impact*, AREVA Inc., November 9, 2015.

5. Letter, JW Shea (TVA) to USNRC, “Proposed Technical Specifications Change TS-505 – Request for License Amendments – Extended Power Uprate – Supplemental Information (CAC Nos. MF4851, MF4852, MF4853)”, CNL-15-240, November 13, 2015. (Accession Number ML15317A361)

6. EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,* Siemens Power Corporation, October 1999.

7. NEDO-32465-A, *Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications*, BWR Owners Group, August 1996.

8. ANP-2860P, Revision 2, “Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information”, AREVA NP Inc., October 2009.

9. NEDC-33435P Revision 1, “Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus,” dated December 2009. (Attachment 3 to Monticello MELLLA+ LAR Accession Number ML100280558, a non-proprietary version of this report is NEDO-33435 Accession Number ML100280557).

2.0 RAIs and Responses

2.1 ***<u>SRXB-RAI 1</u>*** ***<u>MICROBURN-B2 (MB2) Non-Convergence Issue</u>***

The licensee, in Tables 3 and 4 of Enclosure 1, "Browns Ferry EPU [Extended Power Uprate] Low Flow Issues Impact (Proprietary), [1] *AREVA FS1-0024528, Revision 1.0," of the letter dated November 13, 2015, provided the revised operating limit minimum critical power ratio (OLMCPR) two recirculation pump trip (2RPT) limits that are calculated with the revised version of MB2. The changes between the original and revised OLMCPR values are significant (almost a factor of 2). However, (1) the delta over initial versus oscillation magnitude curve is essentially unchanged (see Figure 1), and (2) the 2RPT event is initiated from full power, where MB2 did not have any convergence issue. Provide an explanation for the significant changes in OLMCPR.*

<u>AREVA Response</u>

Tables 3 and 4 of FS1-0024528 (Reference 4, included as Enclosure 1* to Reference 5) address the impacts of the previously identified low-flow issues in the MICROBURN-B2 (Reference 6) code on the BFN Oscillation Power Range Monitor (OPRM) setpoint results provided in the Reference 1 LAR. There are three components to the OPRM setpoint calculation that can affect the results; (1) Hot Channel Oscillation Magnitude (HCOM), (2) Delta over Initial MCPR Versus Oscillation Magnitude (DIVOM), and (3) the change in MCPR during a two Recirculation Pump Trip (2PT).

The HCOM is a plant-specific input parameter that is dependent upon the specific OPRM configuration (i.e. LPRM assignments to OPRM cells) and other plant dependent inputs. This parameter is not recalculated for each cycle and the BFN HCOM values were not generated using MICROBURN-B2; therefore, the HCOM is not impacted by the reported low-flow issues.

As noted in the request above, the DIVOM was essentially unchanged. Because the cycle-specific DIVOM values fall below the generic slope of 0.45 found in NEDO-32465-A (Reference 7), the OPRM setpoints were based upon the more conservative generic slope. Because there was no change in the assumed DIVOM value, the steady-state results provided in the

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1 *Enclosure 2 (FS1-0024530) contains a non-proprietary version of this document.*

* FS1-0024530 is a non-proprietary version of the same report and was included as Enclosure 2 to Reference 5.

OLMCPR(SS) column were not affected by the low-flow issues and the DIVOM did not contribute to the changes in the OLMCPR(2PT) column.

During a 2PT event, the reactor is assumed to start at rated power at the lowest allowable flow condition, identified as state point P1. After the 2PT, the operating state point is at natural circulation flow with a power consistent with equilibration of the feedwater temperature. This post pump trip condition is identified as state point P2. The OPRM setpoint OLMCPR(2PT) results are affected by the change in the MCPR during the pump trip event. This change in the initial condition of the core is expressed as the MCPR_P1 / MCPR_P2 ratio (generally referred to simply as the P1/P2 ratio). This ratio acts as a direct multiplier on the OLMCPR(SS) for the calculation of the OLMCPR(2PT) results.

The changes identified in the OLMCPR(2PT) columns of Tables 3 and 4 of Reference 4 are a direct result of a corresponding decrease in the calculated P1/P2 ratio. While it is true that the initial state point results (i.e. state point P1) results are not affected by the correction of the low-flow issues, the final state point results were impacted because they correspond to a high power condition at very low flow (i.e. natural circulation). The correction of the low flow issues resulted in a more accurate solution at the extreme low flow condition represented by state point P2. The corresponding change in the final core power and MCPR for state point P2 and the corresponding impact on the P1/P2 ratio is summarized in Table 2.1-1 and Table 2.1-2. In these tables, the values that changed from the original calculation are indicated in dark red font and the limiting P1/P2 MCPR ratios that were used in the setpoint calculation are indicated with a **bold font**.

Table 2.1-1 Equilibrium Cycle OPRM Setpoint P1/P2 Impacts

			Original Calculation			Revised Calculation		
Temperature	**Exposure**	**State Point**	**Power (MWt)**	**MCPR**	**P1/P2**	**Power (MWt)**	**MCPR**	**P1/P2**
Nominal Feedwater Temperature	BOC	P1	3952.0	1.431	0.913	3952.0	1.431	0.845
		P2	1911.6	1.567		1862.8	1.694	
	MOC	P1	3952.0	1.423	0.935	3952.0	1.423	0.867
		P2	1920.2	1.522		1842.7	1.641	
	EOC	P1	3952.0	1.543	0.898	3952.0	1.543	0.828
		P2	1923.9	1.718		1824.8	1.864	
Reduced Feedwater Temperature (FWHOOS)	BOC	P1	3952.0	1.413	0.895	3952.0	1.413	0.830
		P2	1824.5	1.578		1794.5	1.703	
	MOC	P1	3952.0	1.435	**0.963**	3952.0	1.435	**0.885***
		P2	1936.1	1.491		1850.3	1.621	
	EOC	P1	3952.0	1.596	0.822	3952.0	1.596	0.825
		P2	1779.0	1.941		1807.7	1.934	

Table 2.1-2 BFE3-19 Representative Cycle OPRM Setpoint P1/P2 Impacts

			Original Calculation			Revised Calculation		
Temperature	**Exposure**	**State Point**	**Power (MWt)**	**MCPR**	**P1/P2**	**Power (MWt)**	**MCPR**	**P1/P2**
Nominal Feedwater Temperature	BOC	P1	3952.0	1.447	0.954	3952.0	1.447	0.888
		P2	1956.3	1.517		1911.3	1.63	
	MOC	P1	3952.0	1.456	0.960	3952.0	1.456	0.888
		P2	1929.2	1.517		1855.0	1.639	
	EOC	P1	3952.0	1.626	0.958	3952.0	1.626	0.888
		P2	1928.3	1.697		1844.7	1.832	
Reduced Feedwater Temperature (FWHOOS)	BOC	P1	3952.0	1.441	0.965	3952.0	1.441	0.900
		P2	1975.2	1.493		1928.8	1.601	
	MOC	P1	3952.0	1.457	**1.010**	3952.0	1.457	**0.933†**
		P2	1963.2	1.442		1889.3	1.561	
	EOC	P1	3952.0	1.685	0.951	3952.0	1.685	0.879
		P2	1912.5	1.772		1823.4	1.916	

* The revised Equilibrium Cycle OPRM Setpoint calculation conservatively rounded up the limiting P1/P2 result to 0.89.

† The revised Representative Cycle OPRM Setpoint calculation conservatively rounded up the limiting P1/P2 result to 0.94.

2.2 ***SRXB-RAI 2 MB2 Void-Quality Correlation***

Section 2.2 of FS1-0024528 provides information that appears to be contradictory. The solution presented is to remove the feature implemented in the correlation that was found to cause problems and then review the analyses to ensure that the results are acceptable. However, if the results of analyses are acceptable, removing the low flux cutoff should have had no effect. Provide a more detailed explanation of the analysis performed, including the cause and effect of the changes implemented in the correlation for this analysis. Indicate what conditions and transients were analyzed and the impact on the BFN EPU LAR.

AREVA Response

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FS1-0024528 (Reference 4, Enclosure 1 to Reference 5) provides a discussion of all the AREVA calculations affected and evaluated for the identified low flow MICROBURN-B2 issues.

2.3 ***SRXB-RAI 3 Bypass Voids***

Figure 7-7 in the enclosure for Attachment 34,[2]"ANP-2860P, Revision 2," to LAR (ADAMS Accession No. ML15282A152) provides the bypass voids at local power range monitor level elevations, but it does not specify the power-flow condition.

a. Provide the bypass void at D-level at the intercept of the backup stability protection region with the high-flow line

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2 *Attachment 35 contains a non-proprietary version of Attachment 34.*

b. *If greater than 5 percent, is a correction required for decalibration (i.e., "losing the calibration" that was set without voids in the bypass)?*

AREVA Response

Figures 7-5 through 7-7 of ANP-2860P Revision 2 Supplement 2P Revision 1 (Attachment 34* of the Reference 1 EPU LAR) were provided to illustrate the capability of the AREVA MICROBURN-B2 model to predict bypass boiling. The results provided for LPRM locations in Figure 7-7 represent a rated power statepoint. However, as noted in the supporting text in Section 7.3 the void levels were artificially increased by reducing the inlet subcooling. Consequently, the specified figure demonstrates the capability to predict localized bypass boiling but the values do not represent actual void conditions expected at BFN for EPU conditions.

(a) Additional calculations have been performed to quantify the level of bypass boiling at the intersection of the Backup Stability Protection (BSP) regions with the MELLLA high flow control line (HFCL). The statepoints used were consistent with the underlying BSP analyses for both the ATRIUM 10XM EPU equilibrium cycle and the BFE3-19 representative EPU cycle. This analysis determined that the limiting case was the intersection of the scram line with the MELLLA HFCL for the equilibrium cycle which exhibited a maximum of [ ] bypass voiding at the D-level. Figure 2.3-1 is an edit of the calculated LPRM bypass voiding for this limiting state point.

(b) If significant bypass voiding had been calculated there would have been no impact on use of the calculated BSP regions because these are manual operating regions that are monitored by the operator using actual power and flow conditions. Even if the APRMs were utilized for monitoring the power level, there would be no significant impact because each APRM is an average of a number of LPRMs from various locations and axial levels within the core.

The impact of LPRM bypass voiding on operation with the OPRM system was addressed in Section 2.1 of ANP-2860P Revision 2 (Reference 8). This detailed evaluation concluded that the steady state and dynamic effects of bypass boiling on lowering the sensitivity of individual LPRM detectors cause [ ] OPRM signals used for comparison with the OPRM amplitude setpoint. Therefore, no correction is required for de-calibration.

* A non-proprietary version of the same report was included as Attachment 35 of the Reference 1 EPU LAR.

Figure 2.3-1 LPRM Bypass Voiding at BSP Intersection with MELLLA HFCL

2.4 ***SRXB-RAI 4*** ***Emergency Operating Procedures (EOPs)***

What version of the emergency operating guidelines (emergeny procedure gudelines (EPG)) is implemented in the current BFN EOPs? Provide a comparison between the values of the cycle-specific EOP parameters (e.g., hot shutdown boron weight) pre- and post-EPU.

AREVA Response

The emergency operating procedures (EOPs) being implemented are based on emergency operating guidelines (EPG) Revision 3.

Calculations are performed each cycle to confirm the reactivity related EOP parameters supporting both the Hot Shutdown Boron Weight (HSBW) calculation and the Minimum Subcritical Banked Withdrawal Position (MSBWP). For supporting HSBW, a hot natural boron concentration (*XB-hot-nat*) of 522 ppm is confirmed to keep the core subcritical at the most reactive point in the cycle. The MSBWP currently used for CLTP cycles was applied to the EPU Equilibrium Cycle core and to the Unit 3 Cycle 19 Representative Cycle core. As shown in Table 2.4-1, the EPU cores remained subcritical by a substantial margin, similar to the CLTP cycles. The MSBWP used in the Table 2.4-1 evaluations assumed 19 rods are banked at position 02 with the remaining rods fully inserted. The MSBWP can be modified on a cycle specific basis if needed or desired.

Table 2.4-1 BFE EPG Reactivity Parameter Sensitivity to EPU

		Shutdown Margin	
Power Level	**Unit & Cycle**	**HSBW (%Δk/k)**	**MSBWP (%Δk/k)**
EPU (3952 MWt)	Equilibrium Cycle	1.321	1.86
	Unit 3 Cycle 19 (Representative Cycle)	1.363	2.25
CLTP (3458 MWt)	Unit 1 Cycle 11	1.474	3.31
	Unit 2 Cycle 19	0.513	2.67
	Unit 3 Cycle 18	1.016	2.42

2.5 ***SRXB-RAI 5 Safety Limit Minimum Critical Power Ration (SLMCPR) Adders***

The NRC requires that the licensee in the EPU applications provide SLMCPR and OLMCPR adders for operation at EPU conditions. Provide a list of the OLMCPR and SLMCPR adders proposed for BFN using AREVA methods and/or a justification for not applying them.

AREVA Response

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] with EPU conditions. Void-quality uncertainties have been addressed in section 6.7 of ANP-2860P Revision 2 (Reference 8) for both transient delta CPR and the SLMCPR. The transient analysis methodology results in predicted power increases that are bounding relative to benchmark tests. In addition, for licensing calculations a 110% multiplier is applied to the calculated integral power to provide additional conservatism to offset uncertainties in the transient analysis methodology. Therefore, uncertainty in the void-quality correlation is inherently incorporated in the transient analysis methodology.

It was identified that these uncertainties are inherently accounted for in the bundle power uncertainty, which is a direct input to the Safety Limit analysis. The analysis performed in Section 6.7 of ANP-2860P Revision 2 with SAFLIM2 is applicable to SAFLIM3D as well. Because these plants have all installed gamma traversing in-core probes (TIPs) the level of conservatism already present in the analysis using TIP uncertainties based upon neutron TIPs to determine the bundle power uncertainty **[** **]** . A detailed discussion of this conservatism is presented in Section 7.3.1 of ANP-2860P Revision 2 which was included in the LAR submittal for the transition to AREVA fuel in Browns Ferry Unit 1. This discussion is applicable to all three units of Browns Ferry.

Figure 2.5-1 presents the TIP statistics information taken from a more current collection of benchmark data of the CASMO-4/MICROBURN-B2 system as a function of core Power/Flow ratio. This plot shows TIP comparison data of Power/Flow ratios up to **[** **]** for plants with and without EPU. There is no indication of higher uncertainties for higher Power/Flow ratios or EPU operation. **[**
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AREVA has reviewed Appendices A, B, C, and D of NEDC-33435P Revision 1, “Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus,” dated December 2009 (Reference 9). **[**
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Figure 2.5-1 3-D TIP Statistics vs. Core Power/Flow Ratio

2.6 ***SRXB-RAI 6*** ***Core Flow***

The power-flow map in Figure 1.1 of Enclosure 3, "ANP-3404P, Revision 3, Browns Ferry Unit 3 Cycle 19 Representative Reload," of letter dated December 15, 2015, shows Point F with 105 percent flow in the increased core flow region. What is the maximum core flow that can be achieved at BFN? Is this a function of exposure (e.g., bottom-peaked shapes may result in reduced max achievable flow)? Is BFN susceptible to bi-stable flow in the recirculation loops? If so, what is the maximum (or range of) achievable recirculation flow used in normal operation to minimize bi-stable flow concerns?

AREVA Response

TVA will provide a response separate from this document.

2.7 ***SRXB-RAI 7*** ***SRV***

In Attachment 6[3], "NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate [aka, PUSAR]," to the LAR, it is assumed that one main safety relief valve (SRV) is out of service (OOS); however, the LAR is confusing as different assumptions are used for different analyses. Has the number of allowed SRVs OOS been changed as a result of the EPU operating domain extension? Is the 3 percent tolerance value supported by testing results?

AREVA Response

TVA will provide a response separate from this document.

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3 *Attachment 7, "NEDO-33860, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate," contains non-proprietary version of Attachment 6.*

2.8 ***SRXB-RAI 8 ACE Critical Power Ratio (CPR) Correlation Applicability Range***

Section 4 of ANP-2860P, Revision 2 states that approved corrective actions are applied if analyzed conditions in BFN fall outside the ACE CPR correlation range of applicability. The range of applicability is defined in Table 2-1 of ANP-10298P[4], Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," in terms of flow, pressure, inlet sub-cooling, and local peaking. However, the approved corrective actions are defined in Table 3-1 of Supplement 1P to ANP-10298PA in terms exclusively of local peaking factor. Please confirm that the analyzed conditions (in terms of flow, pressure, and inlet sub-cooling) in BFN fall inside the range of applicability of the ACE correlation, and only the local peaking (mostly of controlled bundles) is outside.

AREVA Response

The Browns Ferry approved references for the ACE/ATRIUM 10XM critical power correlation are ANP-10298PA Revision 0 and ANP-3140P Revision 0. The range of applicability of the critical power correlation is provided in Table 2-1 of ANP-10298PA Revision 0. The approved actions for conditions that may fall outside of the range identified in Table 2-1 of ANP-10298PA Revision 0 are provided in Section 5.13 of ANP-10298PA Revision 0. Table 3-1 of ANP-3140P Revision 0 shows the ACE correlation additive constant uncertainties which are applied within the AREVA's Safety Limit methodology. Table 3-1 of ANP-3140P Revision 0 does not define the ACE CPR correlation range of applicability.

When the analyzed conditions fall outside of the range of applicability of the ACE/ATRIUM 10XM CPR correlation, the NRC approved conservative actions are applied. For BFN EPU analysis, the analyzed conditions (in terms of flow, pressure, and inlet subcooling) do not exceed the range of applicability of the ACE/ATRIUM 10XM correlation.

2.9 ***SRXB-RAI 9***

ANP-3404, Revision 3, "BFN Unit 3 Cycle 19 Representative Reload Analysis at Extended Power Uprate (Proprietary)[5]" (Enclosure 3 to letter dated December 15, 2015), is submitted to support the EPU operation of all three units.

a. *Identify any design and operational differences between the units that affect the EPU analyses, if any.*

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[4] *A non-proprietary version of this document is provided at ADAMS Accession No. ML 14183A734.*

[5] *Enclosure 4 (ANP-3404NP) contains a non-proprietary version of this document.*

b. *Justify the application of the Unit 3 analysis at EPU for the other two units and explain in detail that the Unit 3 analysis is representative or bounding for the other two units.*

AREVA Response

(a) TVA will provide a response separate from this document.

(b) TVA will provide a response separate from this document.

2.10 ***SRXB-RAI 10***

ANP-3404P, Table 2.1, "Disposition of Events Summary for EPU and AREVA Fuel at Browns Ferry," in the comments column of page 2-8 for the Final Safety Analysis Report, Sections 14.5.4.3 and 14.5.4.4, "Control Rod Removal Error During Refueling," and "Fuel Assembly Insertion Error During Refueling," stated respectively that, "This event is not credible," and, "An unplanned criticality during refueling due to a single fuel assembly insertion error is not credible." The staff does not agree with the statement that these events are not credible. These events, which may be infrequent events, happened in plants, and hence, these events may not be categorized as "incredible events." Provide an evaluation for these infrequent events in BFN.

AREVA Response

The two events referenced in the above request are reactivity insertion events that could potentially occur during refueling. The events are discussed in UFSAR Section 14.5.4.3 “Control Rod Removal During Refueling, and UFSAR Section 14.5.4.4 “Fuel Assembly Insertion Error During Refueling”. The intent of the disposition for both of these items is addressing that the potential consequence, i.e., an inadvertent criticality due to the event is not credible. The current BFN licensing basis for these events as detailed in these UFSAR sections are not affected and thus remain applicable for EPU.

Additionally, the disposition of events summarized in Table 2.1 of ANP-3404P (Enclosure 3* of Reference 2) also addresses a fuel loading error (FLE), either a mislocated or misoriented fuel assembly. In this event, a FLE occurs that is not caught during the refueling or verification of the core loading. The core subsequently operates with the misload condition during the next cycle. AREVA methodology classifies the FLE as an infrequent event. As summarized in Table 2.1 of ANP-3404P, the mislocated and misoriented fuel assembly event is addressed each reload.

* A non-proprietary version of the same report was included as Enclosure 4 of Reference 2.

To minimize the chances of an unplanned criticality during refueling due to a single fuel assembly insertion error, BFN has both plant design requirements and procedure verification requirements as follows.

- Technical Specification (TS) 3.9.1, Refueling Interlocks, requires the refueling interlocks to be operable during refueling. The refueling interlocks preclude prompt reactivity excursions from occurring during refueling by the prevention of loading fuel into the core with any control rod withdrawn or by preventing withdrawal of a control rod from the core during fuel loading. BFN procedures ensure that the TS 3.9.1 requirements are satisfied.

- The BFN verification process for fuel movement includes the use of the following equipment.
  - A camera and monitoring system for live time verifications of fuel loading.
  - A Bridge and Trolley Tracking System which provides an alternate position indication when performing fuel and fuel related component moves in either the reactor vessel or spent fuel pool.

- The BFN verification process for fuel movements involves the following personnel.
  - Two fuel handlers are located on the refuel bridge and perform concurrent verification on fuel movements.
  - The fuel handling supervisor (Senior Reactor Operator(SRO)) has a control station in close proximity to the fuel operation and also verifies all fuel movements.
  - A Reactor Operator (RO) in the control room is stationed to follow the fuel movements and monitor Source Range Monitor (SRM) count rate.
  - The camera and monitoring system operator also verifies each fuel movement by remote cameras.
  - The Reactor Engineer (RE) also verifies fuel movements and monitors SRM count rate remotely.

- The BFN verification process for control rod manipulations or testing during refueling involves the following personnel.
  - Two ROs, stationed at the control panel, using concurrent verification techniques for control rod selection.
  - A dedicated SRO to oversee reactivity manipulations. In addition to the operators, during a shutdown margin test, a RE is also present.

2.11 ***SRXB-RAI 11***

Chapter 14 of the BFN Updated Final Safety Analysis Report addresses loss of a feedwater heater event. Clarify whether any BFN units had any loss of feedwater event during the life of the plant, and if so, what was the maximum feedwater temperature reduction during the event?

AREVA Response

TVA will provide a response separate from this document.

2.12 ***SRXB-RAI 12***

ANP-3404P, Table 1.1, "EOD and EOOS Operating Conditions," lists "Extended Operating Domain (EOD) Conditions" and "Equipment Out-Of-Service (EOOS) Operating Conditions." Section 5.3 describes EOOS scenarios. Section 5.3 describes operation scenarios with turbine bypass valve OOS, feed water heater OOS, power load unbalance OOS, and different combinations of these scenarios.

a. *Clarify whether TVA is planning to take credit for these analyses when the above specified equipment becomes inoperable. ANP-3404P representative analyses are only for Unit 3, Cycle 19, and hence, may not accurately reflect the actual Unit 1 and Unit 2 plant configuration in future operations.*

b. *Specify the number of turbine bypass valves, feed water heater OOS assumed to be inoperable.*

c. *How many power load unbalance devices are in the plant? Are they located in the control room or outside the control room?*

d. *Minimum critical power ratio limits are given in Tables 8.1 and 8.2 for EOOS. Describe the nominal scram speed and technical specifications scram speed tests performed during the EOOS operation, if they are different from the tests done at normal operation.*

e. *Note below that Table 1.1 states, "SLO [single-loop operation] may be combined with all of the other EOOS conditions." The NRC staff does not agree with this statement. SLO operation is susceptible to thermal-hydraulic stability, and feedwater heater OOS may increase the susceptibility. Verify that analyses have been performed to confirm the TV A statement.*

f. *Note below that Table 1.1 mentions two traversing in-core probe (TIP) machines OOS. How many TIP machines are in BFN units?*

AREVA Response

(a) As noted in the response to question SRXB RAI-9, the results in ANP-3404P are considered to provide representative results for an EPU transition cycle. The specific results in this report will not be used to support EOOS operation in future unit 3 EPU cycles, or to support EPU cycles of the other two units. A specific Reload Report will be prepared for each EPU cycle for each unit, that will include cycle specific limits for the EOOS domains.

(b) The turbine bypass valve out of service (TBOOS) analysis assumes that all nine turbine bypass valves are inoperable. The TBOOS limits would be applied if the bypass system was declared inoperable, and would be conservatively applied to situations where an individual bypass valve (or valves) are declared inoperable.

The feedwater heater out of service (FWHOOS) analysis does not make any assumptions with regard to which specific feedwater heaters, or how many, are out of service. Rather, the analysis is based on an assumed final feedwater temperature reduction relative to the condition of having all heaters in service. Heaters may be removed from service at power as needed for maintenance provided that the unit complies with the assumed temperature reduction assumption. In addition, the final stage heater in each feedwater train is typically removed from service near end of cycle to extend full power cycle operation (final feedwater temperature reduction). The assumed reduction in feedwater temperature used in the analysis was selected to accommodate this mode of operation.

(c) Each BFN unit has one PLU device. There are multiple components to the PLU. Most of the components of the PLU are located outside the control room, while one component is located within the control room. In order to perform its function, the PLU compares the main generator output current to the main turbine intermediate pressure.

The main generator current is monitored by current transformers located at the generator, with each output phase of the generator having its own redundant signal. Cables carry these signals to six signal converters (two for each phase) located in the main control room. The signal converters sum the signals for all three phases, producing two redundant signals of total generator output. These signals are then passed to field bus modules that are located in the Aux Instrument Room, which convert the signals to an analog input to a control processor, which is also located in the Aux Instrument Room. Intermediate pressure transmitters located near the main turbine pass a signal of the intermediate turbine pressure to the same control processor in the Aux Instrument Room.

The control processor compares the generator power and pressure signals to detect a power load unbalance condition. When the processor detects a sufficient mismatch between the two, a PLU trip is initiated. The PLU trip triggers a main turbine trip, that in turn leads to a reactor scram.

(d) There is no difference in the scram timing testing process for EOOS conditions compared to normal operations. The scram time test results cannot be influenced by having the turbine bypass system or power load unbalance function out of service. The use of feedwater temperature reduction will result in a slightly reduced reactor pressure condition, but the resulting pressure is still well above the minimum scram time testing pressure of 800 psig required by Technical Specification 3.1.4. Therefore, the scram time testing procedures do not need to account for EOOS operating conditions.

(e) Core hydrodynamic stability is addressed in Section 4.3 of ANP-3404P (included as Enclosure 3* to Reference 2).

BFN has implemented BWROG Long Term Stability Solution Option III (Oscillation Power Range Monitor-OPRM). In the response to SRXB RAI-1 some detail has been provided with regard to the cycle specific OPRM setpoint calculation. Table 2.1-1 and Table 2.1-2 of the response to SRXB RAI-1 illustrate that the cycle specific OPRM setpoint calculation includes the effect of both nominal and reduced feedwater temperature conditions, such as feedwater heaters out-of-service (FWHOOS).

Single loop operation (SLO) may increase the likelihood of an instability scram occurring due to an increase in system noise, but it does not impact the ability of the Option III OPRM system to respond in time to protect the Safety Limit MCPR (SLMCPR), nor does it have a significant impact on the stability characteristics of the system. The limiting case for OPRM setpoint selection is the two recirculation pump trip (2PT) event because it is compared to the rated power operating limit MCPR (OLMCPR) instead of the off-rated OLMCPR used for the steady state (SS) event. The 2PT event is more limiting than a single pump trip event that could occur during SLO due to the following considerations:

(1) After the pump trip, the reactor ends up operating at natural circulation flow regardless of whether it experiences a single or two pump trip. The condition of the core prior to the onset of an instability event is similar for either scenario.

(2) The 2PT event is designed to protect the SLMCPR with the reactor initially operating at a rated power OLMCPR. In SLO, the core is restricted to a lower operating power level. The OLMCPR for reduced power operation is higher than the OLMCPR for rated power ensuring that a single pump trip initiating during SLO will have more margin to the SLMCPR than the rated power 2PT event.

* A non-proprietary version of the same report is included as Enclosure 4 of Reference 2.

In cases where the OPRM system is declared inoperable, manual protection is provided with the implementation of Backup Stability Protection (BSP) regions. As indicated in Section 4.3 of ANP-3404P, the BSP analyses that define these regions were performed to support both nominal and reduced feedwater temperature conditions such as FWHOOS.

For SLO, the stability characteristics of the system will be comparable to similar conditions under two loop operation. This issue has been discussed in Generic Letter (GL) 86-09, “Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs”, dated March 31, 1986. GL 86-09 actually points to testing completed by TVA at BFN Unit 1 on February 9, 1985, that demonstrated “…SLO has similar stability characteristics as two loop operation under the same power/flow operating conditions.” Therefore, no additional evaluations are needed to support SLO with the OPRM system inoperable.

(f) Each BFN unit has five TIP machines.

2.13 ***SRXB-RAI 13***

PCT for anticipated transient without scram (ATWS) is not provided in the LAR. Provide a qualitative assessment to justify the TVA position that the PCT calculation is not needed for the ATWS scenario.

AREVA Response

For ATWS events, the acceptance criteria for PCT and local cladding oxidation for ECCS, defined in 10 CFR 50.46, are adopted to ensure an ATWS event does not impede core cooling. Coolable core geometry is assured by meeting the 2200°F PCT and the 17% local cladding oxidation acceptance criteria stated in 10 CFR 50.46.

There is no core uncovery associated with the ATWS event, hence the PCT and local cladding oxidation results will be bounded by LOCA. Therefore, the PCT and local cladding oxidation for the BFN ATWS events are qualitatively evaluated to demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

2.14 ***SRXB-RAI 14***

ANP 3403P, Revision 3, "Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3 (Proprietary)," Enclosure 1[6] to the letter dated December 15, 2015, in Section 2.8.5.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," Technical Evaluation, TVA stated, "The event was analyzed at EPU conditions and resulted in an unblocked delta CPR of 0.27." What is meant by "unblocked"?

AREVA Response

TVA will provide a response separate from this document.

2.15 ***SRXB-RAI 15***

ANP 3403P, Section 2.8.5.4.4, "Spectrum of Rod Drop Accidents," Technical Evaluation (page 95) in the Criteria Table, number of failed rods is given as less than 850. Identify the source from where the criterion for number of failed rods was obtained (less than 850 in this case).

[6] *Enclosure 2 contains a non-proprietary version of Enclosure 1.*

AREVA Response

TVA will provide a response separate from this document.

2.16 ***SRXB-RAI 16***

TVA stated, in page 41 Attachment 8[7], "ANP-3403P, Fuel Uprate Safety Analysis Report [FUSAR] for Browns Ferry Units 1, 2, and 3 (proprietary)," LAR, that the average bundle power increases from 4.53 MW/bundle to 5.17 MW/bundle (approximately 14 percent) from pre-EPU to post-EPU, which corresponds to the same percent increase of total core power from current licensed thermal power (CLTP) to EPU. It is assumed in the constant pressure EPU for BWRs that the additional core power is obtained by raising the average bundle power, and that the peak bundle power should remain approximately the same. However, other plant operations at EPU power level have shown that peak bundle power can increase by a limited amount. Provide the current peak bundle power and compare it with the expected value of peak bundle power for EPU operation in BFN.

AREVA Response

Table 2.16-1 provides a comparison of the peak bundle powers for the both EPU and CLTP (i.e. pre-EPU) operation at BFN. The EPU cycles are those supporting the BFN EPU LAR (Reference 1). The comparison CLTP cycles are those that contain one or more reload of the ATRIUM 10XM fuel design at BFN.

The peak bundle power for EPU is approximately 7.3 MWt versus 7.0 MWt for CLTP when averaged over the comparison cycles. This represents an increase of about 4% compared to the corresponding reactor power increase of slightly more than 14%. The difference is due to the radial power flattening that is required to maintain operation within thermal limits.

Table 2.16-2 provides a similar comparison showing the impact on the peak linear heat generation rate (LHGR). The comparison cycles all include the ATRIUM 10XM fuel design and are subject to the same fuel design limit. As shown in this table, there is no significant change in the peak LHGR due to implementation of EPU. The same peak LHGR is maintained by additional localized power flattening within the assembly through a combination of bundle design and choice of operating rod patterns.

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7 *Attachment 9 contains a non-proprietary version of Attachment 8.*

Table 2.16-1 Peak Bundle Power Comparison

Power Level	Cycle Description	Average Bundle Power (MWt)	Peak Bundle Power (MWt)	Average Peak Bundle Power (MWt)
EPU (3952 MWt)	Equilibrium Cycle (ATRIUM 10XM EPU)	5.173	7.361	7.327
	Unit 3 Cycle 19 (Representative Cycle)		7.294	
CLTP (3458 MWt)	Equilibrium Cycle (ATRIUM 10XM CLTP)	4.526	7.034	7.025
	Unit 2 Cycle 19		7.034	
	Unit 3 Cycle 18		7.007	

Table 2.16-2 Peak Linear Heat Generation Rate (LHGR) Comparison

Power Level	Cycle Description	Peak LHGR (kW/ft)	Average Peak LHGR (kW/ft)
EPU (3952 MWt)	Equilibrium Cycle (ATRIUM 10XM EPU)	[ ]	[ ]
	Unit 3 Cycle 19 (Representative Cycle)	[ ]	
CLTP (3458 MWt)	Equilibrium Cycle (ATRIUM 10XM CLTP)	[ ]	[ ]
	Unit 2 Cycle 19	[ ]	
	Unit 3 Cycle 18	[ ]	

2.17 ***SRXB-RAI 17***

Stress corrosion cracking (SCC) and pellet-clad-interaction (PCI) phenomena can cause clad perforation, resulting in leaking fuel bundles and resultant increased reactor coolant activity. Provide the following additional information regarding PCI/SCC for BFN at EPU conditions:

a. *Describe whether ATRIUM-10 and XM fuel designs have barrier cladding that has built-in PCI resistance.*

b. *Describe any differences in operating procedures associated with PCI/SCC at EPU conditions versus pre-EPU operations.*

c. *From the standpoint of PCI/SCC, discuss which of the anticipated operational occurrences (AOOs), if not mitigated, would most affect operational limitations associated with PCI/SCC.*

d. *For the AOOs in part c. above, discuss the differences between the type of required operator action, if any, and the time to take mitigating actions between pre-EPU and EPU operations.*

e. *If the EPU core will include fuel designs with non-barrier cladding that have less built-in PCI resistance, demonstrate by plant-specific analyses that the peak clad stresses at EPU conditions will be comparable to those calculated for the current operating conditions.*

f. *Describe operator training on PCI/SCC operating guidelines.*

AREVA Response

(a) All fuel being loaded in the BFN cores have barrier clad.

(b) The BFN units utilize the advanced XEDOR methodology to analyze cladding hoop stress for every pin segment in the core. This will continue to be used with operation at EPU conditions. There is no difference in operating procedures for EPU operation. This methodology precludes experiencing high cladding hoop stresses during normal operating conditions.

While TVA currently uses XEDOR, both XEDOR and REMACCX are acceptable methods to monitor fuel for PCI risk at EPU conditions.

(c) Previous analysis for other plants identified the Loss of FeedWater Heaters (LFWH) event as the most restrictive event from the standpoint of a large number of PCI/SCC fuel failures. This analysis demonstrated that the number of rods at risk is not significantly different for operation at EPU conditions relative to operation at the Original

Licensed Thermal Power (OLTP). This study was published in the 2008 “Water Reactor Fuel Performance” conference under the title “XEDOR Evaluation of PCI Fuel Failure Risk due to Loss of Feedwater Heating in Boiling Water Reactors,” Paper No. 8142.

(d) The operator actions and timing for a loss of feedwater heating event are the same for both pre EPU and EPU operation. The BFN plant procedures for response to a loss of feedwater heating event require that the operator take immediate action to reduce the reactor power to a level 5% or more below the reactor power level that existed prior to the start of the event. This prompt action ensures that the fuel rod power changes in the lower portion of the core do not result in a Pellet Clad Interaction (PCI) overpower condition due to the increased peaking in the lower portion of the core (which results from the higher core inlet subcooling). The relative change in core power and local power peaking from the loss of feedwater heating event is primarily a function of the change in feedwater temperature, rather than the initial power from which the event is initiated. The change in the core power and power shape would be similar between pre EPU and EPU conditions, because the change in feedwater temperature from the event is the same for both initial power levels. The similarity in core response and fuel PCI response to a loss of feedwater heating event initiated from the two initial power levels was confirmed in the studies noted in response to part (c) of this RAI.

Therefore, use of the same operator actions and timing for a loss of feedwater heating event for both EPU and pre EPU conditions is appropriate.

(e) The fuel designs include barrier cladding.

(f) With regard to operator training on PCI/SCC operational guidance, the operators are provided with an overview of the fuel conditioning monitoring requirements as part of their requalification training. For each new cycle of a unit, a lesson plan is developed to provide training on the design of the new cycle core, an overview of changes relative to the prior cycle, the effect of the new cycle core design and reload safety analyses on reactor operations, and operational considerations for the new cycle. One of the learning objectives related to operational considerations are the requirements with regard to fuel conditioning.

The training on fuel conditioning provides the operators with a high level overview of the fuel conditioning model contained within the core monitoring system. The fuel conditioning figure of merit output by the core monitoring system is reviewed, including the limits on the figure of merit to ensure compliance with the vendor fuel preconditioning guidance. The PCI/SCC portion of the lesson also provides a discussion of the PCI/SCC resistant features of the fuel, notably the use of liner (barrier) fuel cladding.

2.18 ***SRXB-RAI 18***

TVA stated in pages 5-6 and 2-14 of ANP-3404P that CASMO-4/MICROBURN-B2 code was used to calculate flow dependent linear heat generation rate operating limits to establish the fuel failure criteria for the recirculation flow control failure - increasing flow event. The analysis of the event assumes recirculation flow increases slowly along the limiting rod line to the maximum flow physically attainable by the equipment. Discuss whether there was any impact of MICROBURN-B2 low flow issues on this slow flow event.

AREVA Response

The flow-dependent linear heat generation rate multipliers ($LHGRFAC_f$) provided in Table 8.7 of ANP-3404P (Enclosure 3 of Reference 2) are based upon analyses performed with the MICROBURN-B2 (Reference 6) computer program. These analyses were evaluated as part of AREVA's corrective action program for both the low flow convergence and low flow void quality issues. This review concluded that the analyses were not impacted and the provided $LHGRFAC_f$ values in ANP-3404P are not impacted. **[**

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2.19 ***SRXB-RAI 19***

TVA stated in page 2-410 of the PUSAR that the standby liquid control system (SLCS) relief valve setpoint margin is 33 pounds per square inch (psi) for EPU. Please provide the following additional information:

a. What is the allowable value (AV) for SLCS relief valve setpoint drift for BFN?

b. Whether the 33 psi AV account for relief valve setpoint drift, and justify that the available margin remains conservative at EPU conditions.

c. The relief valve setpoint margin for CLTP after accounting for the AV setpoint drift.

d. The minimum margin required for the SLCS pump discharge relief valves to remain closed during system injection.

AREVA Response

(a) TVA will provide a response separate from this document.

(b) TVA will provide a response separate from this document.

(c) TVA will provide a response separate from this document.

(d) TVA will provide a response separate from this document.

2.20 **SRXB-RAI 20**

In page 107 of the FUSAR, TVA stated that BFN reactors are small-break limited, as determined by AREVA's EXEM BWR-2000 evaluation model, and that this trend in the break spectrum analysis was not impacted by EPU. Provide the following additional information regarding the loss-of-coolant accident (LOCA) analysis performed in support of EPU operation at BFN:

a. In page 109 of the FUSAR, limiting LOCA break characteristics (location, type/size, single failure, axial power shape, and initial state) were provided for EPU condition. Compare these characteristics of EPU with that of pre-EPU condition. Discuss if there was any significant change in these characteristics as a result of power uprate from CLTP to EPU, and explain why.

b. Provide the total number of automatic depressurization system (ADS) valves at BFN, and the number of ADS valves assumed to be OOS for limiting small-break LOCAs for CLTP and EPU. Explain the difference, if any.

c. The limiting PCT for EPU is provided in FUSAR, page 109, as 2,086 degrees Fahrenheit (°F) for ATRIUM-10 and 2008 °F for XM fuels. Provide the limiting PCT for CLTP condition, including the type of fuel, and explain any significant change between the PCTs before and after EPU.

d. Provide the break location, type/size, single failure, axial power shape, initial state, and the PCT at EPU condition for the worst large-break LOCA case.

AREVA Response

(a) The limiting LOCA characteristics for CLTP and EPU conditions are provided in the table below. There is no significant change in these characteristics as a result of power uprate.

	EPU Conditions	CLTP Conditions
Location	Recirculation discharge pipe	Recirculation discharge pipe
Type / size	Split / 0.23 ft^2	Split / 0.20 ft^2
Single failure	Battery (DC) power, board A	Battery (DC) power, board A
Axial power shape	Top-peaked	Top-peaked
Initial state	102% power / **[ ]**	102% power / **[ ]**

(b) The total number of ADS valves at Browns Ferry is 6. For the limiting single failure condition, which is battery failure affecting Reactor Motor Operator Valve (RMOV) board A, there are 6 available ADS valves remaining. The same number of valves are available for both CLTP and EPU conditions.

(c) For CLTP conditions, the limiting PCT for ATRIUM-10 fuel is 2029°F and the limiting PCT for ATRIUM 10XM fuel is 1985°F. The EPU and CLTP analyses were consistent applications of AREVA's NRC approved EXEM BWR-2000 ECCS Evaluation Model. The MAPLHGR set in the hot channel analysis was the same in both analyses and is a key parameter in the prediction of similar limiting results. The difference in the EPU and CLTP results is expected to be the result of the differences in EPU and CLTP core conditions. The core conditions that change to support EPU include higher core stored energy, higher core void fraction, and flatter core power distribution.

(d) The PCT results for the full range of break characteristics analyzed in support of EPU are provided in Table 6.3, Table 6.6, Appendix B, and Appendix C of ANP-3377P Revision 3, which is Attachment 10 of Reference 1.

2.21 ***SRXB-RAI 21***

Provide a brief description of how the stability mitigation actions (e.g., immediate water level reduction and early boron injection) are implemented in BFN. Discuss if operation at constant pressure power uprate conditions requires modification of any operator instruction, including the EOP and the EPGs and severe accident guidelines (EPG/SAG) that are applicable to BFN.

AREVA Response

TVA will provide a response separate from this document.

2.22 ***SRXB-RAI 22***

Table 2.6-3 of PUSAR provides BFN peak suppression pool (SP) temperature for postulated ATWS, station blackout, and Appendix R fire events. Compare peak SP temperatures for these events with that of pre-EPU condition, and explain any significant change.

AREVA Response

TVA will provide a response separate from this document.

2.23 ***SRXB-RAI 23***

Provide a brief description of the plant training simulator neurotic core model. Provide the schedule as to when the BFN plant training simulator is upgraded for EPU conditions.

AREVA Response

TVA will provide a response separate from this document.

ENCLOSURE 3

Response to NRC RAIs SRXB-RAI 6, 7, 9, 11, 14, 15, 19, 21, 22, and 23

SRXB-RAI 6

The power-flow map in Figure 1.1 of Enclosure 3, "ANP-3404P, Revision 3, Browns Ferry Unit 3 Cycle 19 Representative Reload," of letter dated December 15, 2015, shows Point F with 105 percent flow in the increased core flow region. What is the maximum core flow that can be achieved at BFN? Is this a function of exposure (e.g., bottom-peaked shapes may result in reduced max achievable flow)? Is BFN susceptible to bi-stable flow in the recirculation loops? If so, what is the maximum (or range of) achievable recirculation flow used in normal operation to minimize bi-stable flow concerns?

TVA Response:

The Reactor Recirculation System (RRS) evaluation is contained in Section 2.8.4.6 of Attachment 6 to the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 Extended Power Uprate (EPU) LAR. At EPU conditions, the RRS and its components in all three BFN units are fully capable of supporting 105% of rated core flow (RCF) at all core exposures and power shapes. 105% of RCF is the current maximum licensed core flow for BFN, which is unchanged for the proposed EPU. Therefore, a maximum recirculation system capability evaluation has not been performed to support operation at greater than 105% of RCF.

BFN Units 1, 2, and 3 are not susceptible to bi-stable flow in the recirculation loops. The small increase in recirculation system drive flow and the small changes in reactor water temperature and density associated with EPU do not result in any significant increase in the susceptibility to bi-stable flow in the recirculation loops. BFN periodically monitors the recirculation system and plant parameters, e.g., power and electrical output, for fluctuations that could be indicative of bi-stable recirculation flow. No change to this monitoring is required for EPU.

SRXB-RAI 7

In Attachment 6[3], "NEDC-33860P, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate [aka, PUSAR]," to the LAR, it is assumed that one main safety relief valve (SRV) is out of service (OOS); however, the LAR is confusing, as different assumptions are used for different analyses. Has the number of allowed SRVs OOS been changed as a result of the EPU operating domain extension? Is the 3 percent tolerance value supported by testing results?

TVA Response:

The maximum number of Main Steam Relief Valves out of service (MSRV OOS) is unchanged for the analyses performed to support the Browns Ferry Nuclear Plant (BFN) Extended Power Uprate. The maximum number of MSRV OOS is one. A three-percent (3%) MSRV setpoint tolerance is supported by BFN plant-specific testing of MSRVs.

[3]Attachment 7, "NEDO-33860, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate," contains a non-proprietary version of Attachment 6.

ENCLOSURE 3

SRXB-RAI 9

ANP-3404, Revision 3, "BFN Unit 3 Cycle 19 Representative Reload Analysis at Extended Power Uprate (Proprietary)[5]" (Enclosure 3 to letter dated December 15, 2015), is submitted to support the EPU operation of all three units.

a. *Identify any design and operational differences between the units that affect the EPU analyses, if any.*

b. *Justify the application of the Unit 3 analysis at EPU for the other two units and explain in detail that the Unit 3 analysis is representative or bounding for the other two units.*

TVA Response:

Response to Item a):

There are no design and operational differences between the units which affect the Extended Power Uprate (EPU) analyses.

The three Browns Ferry Nuclear Plant (BFN) units are essentially the same in terms of design characteristics that would affect the results in an EPU Reload Report. The core size and layout are identical between the BFN units. The pressure vessels are the same size and the reactor internals have the same geometry. The geometry of the main steam lines is comparable between the units. Therefore, the modeling and nodalization used in the EPU transient analysis model is the same for each unit.

Each unit has twenty jet pumps, and the recirculation pump design and maximum recirculation flow are the same between the BFN units. Because these characteristics are common, the transient response to a recirculation flow runup event at EPU will be similar for all three BFN units.

The number of steam line safety relief valves is the same for each unit, and the opening and closing setpoints of the relief valves are the same for all three BFN units. Coupled with the reactor vessel and steam line geometries being comparable, the reactor vessel pressure response at EPU to transients involving over-pressurization conditions will be very similar.

The modeled EPU plant heat balance is the same for the three BFN units. The rated core flow, steam dome pressure as a function of core power, and the feedwater temperature as a function of power are the same between the BFN units. The licensed power-flow map for EPU operations is common between the three BFN units. Therefore, the range of initial conditions for the events considered in an EPU Reload Report are the same for the three BFN units.

The design of the feedwater heating system is the same for each BFN unit in terms of number of feedwater trains and number of stages of heating, and the temperature change across each stage of feedwater heating. Each BFN unit has three feedwater pumps with identical equipment/system alignment for EPU operation. As noted, the final feedwater temperature for a given power is also common to the three BFN units. Therefore, the plant response at EPU to a feedwater controller failure event (typically the limiting event for BFN for setting the power dependent thermal operating limits) will be very similar. In addition, the temperature change for a loss of feedwater heating event will be the same between the BFN units.

[5]Enclosure 4 (ANP-3404NP) contains the non-proprietary version of this document.

As noted in the discussion above, the plant characteristics are such that a common set of plant parameters are used for all three BFN units for both the transient and accident evaluations presented in BFN Reload Reports.

In terms of operational differences, there are no differences between the three BFN units that would have a significant effect on the results in an EPU Reload Report. As noted, the three BFN units will have the same power-flow map for EPU operation. The equipment out of service (EOOS) domains are the same for all three BFN units, and are those presented in the Unit 3 Cycle 19 Representative Reload Report.

Response to Item b):

As discussed in the response to part (a) of this RAI, the three BFN units are essentially identical in terms of plant design features and operating conditions, and therefore there would be minimal differences in the Reload Report results between the BFN units caused by plant differences. The primary driver for variations in the Reload Report results (cycle to cycle or from one unit to another) is the core design and the associated fuel reactivity characteristics. The core design and the fuel characteristics (such as fuel types in core and enrichment) can affect the void coefficient of reactivity, as well as the scram reactivity, which can affect the results of the pressurization events to some degree. The void coefficient differences can also affect the results of cold water events to some degree.

The power dependent Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) operating limits are expected to be set by pressurization events for any EPU cycle for each of the three BFN units. Flow dependent MPCR and LHGR operating limits would be set by the slow recirculation flow runup event for any EPU cycle for each of the three BFN units.

Local events such as Control Rod Withdrawal Error response and Control Rod Drop Accident results can vary somewhat due to core design differences. Some of the results in an EPU Reload Report will not vary by either cycle or by unit; these include the Loss of Coolant Accident and the Fuel Handling Accident. The Thermal Hydraulic Stability event is non limiting, as shown in the Unit 3 Cycle 19 Reload Report. Stability would not be expected to be a limiting event for setting MCPR operating limits for any future EPU cycle for any of the three BFN units.

The differences in results that arise from cycle to cycle variation in core design can be examined by comparing the transient results in the Unit 3 Cycle 19 Representative Reload Report to the transient results in the Fuel Uprate Safety Analysis Report (FUSAR). The FUSAR report was based on an equilibrium core design, while the Unit 3 Cycle 19 report used a transition core design.

All three BFN units will utilize the same reload fuel type (ATRIUM-10 XM) for EPU cycle operation. In addition, the same NRC approved methodologies listed in Technical Specification 5.6.5.b are used to perform the evaluations in the Reload Report, and to establish the limits in the Core Operating Limits Report for each BFN unit.

In summary, the Unit 3 Cycle 19 Representative Reload Report provides results that are considered to be representative of an EPU transition cycle for any of the three BFN units. The report is not intended to be bounding for future EPU cycles on unit 3, or to bound EPU cycles of the other two BFN units. As noted earlier, some modest variations in results are always expected from cycle to cycle due to core design differences. For these reasons, the Reload Report is always evaluated and updated on a cycle specific basis for each BFN unit.

SRXB-RAI 11

Chapter 14 of BFN Updated Final Safety Analysis Report addresses loss of a feedwater heater event. Clarify whether any BFN units had any loss of feedwater event during the life of the plant, and if so, what was the maximum feedwater temperature reduction during the event?

TVA Response:

The loss of feedwater heater event is an Updated Final Safety Analysis Report transient that is calculated for each reload core. A maximum feedwater temperature decrease of 100°F is used to determine the effect on fuel thermal limits. This relatively slow event would typically result in a power increase, but no vessel pressure increase or scram. Any excess steam flow (power) increase would be accommodated by bypass valves opening to maintain vessel pressure. This non-limiting event is not a transient that establishes fuel thermal limits.

There are three feedwater strings, with five heaters per string in each unit. Typical feedwater heating events involve steam side level disturbances in one or several heaters that remove or divert extraction steam from a heater causing a drop in normal feedwater temperature.

Over the last 10 years of operation there have been 30 feedwater temperature transient events recorded in operator's logs and fatigue monitoring logs for all three units. A 10 year period was selected for review because of previous extended plant shutdowns. BFN Unit 1 was restarted in 2007 after being shutdown in 1985.

The largest drop in feedwater temperature determined from analysis of the events described above was 43°F.

SRXB-RAI 14

ANP 3403P, Revision 3, "Fuel Uprate Safety Analysis Report for Browns Ferry Units 1, 2, and 3 (Proprietary)," Enclosure 1[6] to the letter dated December 15, 2015, in Section 2.8.5.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," Technical Evaluation, TVA stated, "The event was analyzed at EPU conditions and resulted in an unblocked delta CPR of 0.27." What is meant by "unblocked"?

TVA Response:

The term “unblocked” refers to whether credit was taken in the rod withdrawal error analysis for the rod block monitor reaching its trip setpoint and blocking further withdrawal of the control rod. An unblocked control rod withdrawal error analysis gives a conservative result in terms of delta critical power ratio (CPR) response, in that the control rod is allowed to stroke from its initial position to the full out position with no mitigation by the rod block monitor function.

[6]Enclosure 2 contains non-proprietary of Enclosure 1.

SRXB-RAI 15

ANP 3403P, Section 2.8.5.4.4, "Spectrum of Rod Drop Accidents," Technical Evaluation (page 95) in the Criteria Table, number of failed rods is given as less than 850. Identify the source from where the criterion for number of failed rods was obtained (less than 850 in this case).

TVA Response:

The criteria of less than 850 fuel rod failures comes from the radiological analysis of record (AOR) for the Control Rod Drop Accident (CRDA), which assumes the radiological source term is based on 850 failed fuel rods. This was based on the 8x8 fuel design that contained 62 rods per assembly. This is equivalent to 13.7 fuel assemblies or 1.79% of the entire core. Because the current fuel used at BFN has more than 62 rods per assembly, a criterion of less than 850 failed rods in the CRDA analysis ensures that the assumed core damage (850 failed rods / (764 fuel assemblies x 62 rods/fuel assembly)) for the CRDA in the radiological AOR remains bounding.

SRXB-RAI 19

TVA stated in page 2-410 of the PUSAR that the standby liquid control system (SLCS) relief valve setpoint margin is 33 pounds per square inch (psi) for EPU. Please provide the following additional information:

a. What is the allowable value (AV) for SLCS relief valve setpoint drift for BFN?

b. Whether the 33 psi AV account for relief valve setpoint drift, and justify that the available margin remains conservative at EPU conditions.

c. The relief valve setpoint margin for CLTP after accounting for the AV setpoint drift.

d. The minimum margin required for the SLCS pump discharge relief valves to remain closed during system injection.

TVA Response:

a) The maximum standby liquid control system (SLCS) relief valve setpoint tolerance (drift) is +/-75 psi.
b) The 33 psi margin takes into account the SLCS relief valve setpoint tolerance (drift) of +/-75 psi (taken from the SLCS relief valve nominal setpoint of 1425 psig) plus an additional margin of 30 psi for SLCS pump pulsation, plus elevation head and line loss differences between the SLCS pump and the relief valve inlet piping. The use of 30 psi is conservative because the Browns Ferry Nuclear Plant (BFN) SLCS employs an accumulator on the discharge piping of each SLCS pump expressly for the use of dampening the pump pressure pulsations. The 33 psi margin remains conservative for EPU conditions because the analysis takes into account the actual maximum reactor lower plenum pressure of 1201 psig predicted for the limiting Anticipated Transient Without Scram (ATWS) event plus the calculated maximum pump discharge pressure of 1295 psig during this event including line losses and elevation head between the pump and the reactor vessel lower plenum region. Line losses are conservatively calculated using an SLCS pump design flow rate of 56 gpm.
c) The relief valve setpoint margin for CLTP operation after accounting for SLCS relief valve setpoint drift is 41 psi. This value is based on a lower plenum pressure of 1174 psig (limiting ATWS event) and a maximum pump discharge pressure of 1279 psig.
d) Considering that the evaluation takes into account the relief valve drift and compensation for pump pressure pulsations in the 33 psi margin, the minimum acceptable margin would, therefore, be 0 psi. In other words, no additional margin is required when accounting for relief valve drift and pump pressure pulsations.

SRXB-RAI 21

Provide a brief description of how the stability mitigation actions (e.g., immediate water level reduction and early boron injection) are implemented in BFN. Discuss if operation at constant pressure power uprate conditions requires modification of any operator instruction, including the EOP and the EPGs and severe accident guidelines (EPG/SAG) that are applicable to BFN.

TVA Response:

Stability Mitigation Actions are implemented by entry into the Reactor Scram Abnormal Operating Instruction (AOI) and Emergency Operating Instructions (EOI-1A: ATWS RPV Control). These procedures include all immediate actions and subsequent actions to lower reactor water level and inject boron from the Standby Liquid Control System (SLCS) for an Anticipated Transient Without Scram (ATWS) event with thermal-hydraulic instabilities (ATWSI).

If reactor power is greater than 5%, then the SLCS will be initiated. Boron from the SLCS is injected in accordance with a “hard card.” This hard card is a two-page procedure which is laminated and located at the associated control panel and is part of the Reactor Scram AOI. The hard card allows actions to be carried out more quickly and has standardized the operator actions for any scram.

The action of controlling power by lowering reactor water level is implemented by the EOIs. During revision 3 of the EOIs, a separate EOI flow chart was created (ATWS RPV Control). This revision moved the action to lower water level to just after confirming Containment Isolations have been successful and inhibiting the Automatic Depressurization System (ADS), so the action occurs sooner than the previous revision. Therefore, if reactor power is above 5% or unknown, injection to the reactor vessel is stopped (except SLCS, control rod drive, and reactor core isolation cooling) and reactor water level is lowered to -50 inches. Containment and reactor vessel parameters are then evaluated to determine if reactor water level will need to be lowered further.

Operations training was performed both in the classroom and on the simulator on the mitigation strategies during an ATWS transient. This training discussed the importance of lowering reactor water level immediately to suppress instabilities during ATWS events and the new strategy for reducing reactor level rapidly. These strategies included the hard card usage and the new EOI flow chart for ATWS. The use of hard cards for injecting boron from the SLCS and lowering of reactor water level was then practiced on the simulator to ensure that each operator was able to perform the tasks.

Based on the AREVA evaluation in Fuel Uprate Safety Analysis Report, Section 2.8.5.7, Anticipated Transient without Scram, no changes to any operator instruction, including the Emergency Operating Procedures (EOPs) and the Emergency Procedures and Severe Accident Guidelines (EPGs/SAGs) that are applicable to Browns Ferry Nuclear Plant (BFN), are needed for Extended Power Uprate.

SRXB-RAI 22

Table 2.6-3 of PUSAR provides BFN peak suppression pool (SP) temperature for postulated ATWS, station blackout, and Appendix R fire events. Compare peak SP temperatures for these events with that of pre-EPU condition, and explain any significant change.

TVA Response:

The highest peak suppression pool (SP) temperature for all Anticipated Transient Without Scram (ATWS) events, using the Current Licensed Thermal Power (CLTP) inputs as shown in Table 2.8-1 of the Extended Power Uprate (EPU) License Amendment Request (LAR) Attachment 6 Power Uprate Safety Analysis Report (PUSAR), is 214.6°F. The decrease in peak SP temperature from the pre-EPU analysis to the EPU analysis is primarily due to the increased rate of neutron poison (Boron-10) injection assumed in the EPU analysis. The assumed standby liquid control system (SLCS) pump flow, SLCS storage tank boron concentration and the B-10 enrichment are increased for the EPU analysis. This results in a quicker shutdown of the ATWS unit and a subsequent reduction in the integrated heat addition to the SP. A secondary effect is the higher RHR heat exchanger K-value used in the EPU analysis, which results in a larger heat removal rate from the SP.

The peak SP temperature for Station Blackout, performed at the CLTP power level of 3458 MWt, is 194.1°F. The increase in peak SP temperature to 203.7°F at EPU is due to the higher decay heat of the EPU core operated at an initial power level of 3952 MWt.

The peak SP temperature for National Fire Protection Association (NFPA) 805 fire events, performed at the CLTP power level of 3458 MWt, is 205.7°F. The primary change between the CLTP and EPU analysis is that the CLTP analyses used an RHR heat exchanger K-value of 270 BTU/sec-°F, while the EPU analysis used a RHR heat exchanger K-value of 307 TU/sec-°F. This increased RHR heat exchanger heat removal rate is the primary reason for the mild increase in peak SP temperature for the EPU analysis (208.0°F peak SP temperature at EPU versus 205.7°F peak SP temperature at CLTP).

Justification for the RHR heat exchanger K-values used in the EPU analyses is contained in the EPU LAR Attachment 39.

SRXB-RAI 23

Provide a brief description of the plant training simulator neutronic core model. Provide the schedule as to when the BFN plant training simulator is upgraded for EPU conditions.

TVA Response:

The Browns Ferry Nuclear Plant (BFN) Unit 2 and 3 plant training simulators neutronics model is S3R, a real-time derivative of the three dimensional (3D) transient nodal code SIMULATE-3K (S3K). S3R solves the transient 3D, two-group neutron diffusion equations, including the six group equations for the delayed neutron precursors. The core model is a full 3D model based on two group methods with explicit reflector models.

The cross sections used by the model are generated for all the fuel lattices in the core, and have functional dependence on fuel burnup, moderator density history, instantaneous moderator density, fuel temperature, control rod presence, soluble boron (for standby liquid control injection scenarios), xenon, and samarium. The core model is updated for each new cycle to reflect the actual core loading and fuel resident in the core. The operating history of the fuel carried over from prior cycles is explicitly accounted for, as well as the projected operation of the new cycle.

The thermal hydraulic characteristics of the various fuel designs resident in core are explicitly accounted for in the model. All control rods are modeled explicitly. The core model accounts for fission energy deposited as thermal energy both inside the fuel pellet where the fission takes place and outside the pellet due to neutron and gamma attenuation. The heat deposition location for prompt and delayed power may be different. A decay heat model is included and is based on American National Standards Institute/American Nuclear Society (ANSI/ANS)-5.1, Decay Heat Power in Light Water Reactors.

The BFN Unit 2 training simulator already has an Extended Power Uprate (EPU) upgrade that has been developed and is available for use as needed. This upgrade to the Unit 2 simulator will be used to support operator training on EPU and for procedure validations related to EPU implementation. The Unit 3 simulator is in the process of completing an EPU upgrade. It will incorporate the changes to the unit made during the Spring 2016 refueling outage. The changes to the Unit 3 simulator are expected to be completed by late spring 2016. The EPU upgrade reflects the plant equipment changes as well as the core neutronics model changes. Note that BFN does not have a Unit 1 training simulator.

ENCLOSURE 4

AREVA Affidavit

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

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

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

3. I am familiar with the AREVA information contained in the report ANP-3473, Revision 0, "Responses to RAIs for Browns Ferry Nuclear Plant EPU Submittal – Reactor Systems Branch Questions," dated March, 2016 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 22nd

day of March, 2016.

Mary Anne Heilman
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 6/6/2016

Mary Anne Heilman
Commission Expires
Notary Public
06-06-16
State of Washington