Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

August 15, 2008

TVA-BFN-TS-418

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop OWFN, P1-35
Washington, D. C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos. 50-260
Tennessee Valley Authority	)	50-296

BROWNS FERRY NUCLEAR PLANT (BFN) – UNITS 2 AND 3 – TECHNICAL SPECIFICATIONS (TS) CHANGE TS-418 – EXTENDED POWER UPRATE (EPU) – PARTIAL RESPONSE TO ROUND 18 REQUEST FOR ADDITIONAL INFORMATION (RAI) (TAC NOS. MD5263 AND MD5264)

By letter dated June 25, 2004 (ADAMS Accession No. ML041840301), TVA submitted a license amendment application to the NRC for EPU operation of BFN Units 2 and 3. The pending EPU amendment increases the maximum authorized power level by approximately 14 percent from 3458 megawatts thermal (MWt) to 3952 MWt.

On July 17, 2008, the NRC staff issued a Round 18 RAI (ML081700102) regarding AREVA fuel methods used in support of Units 2 and 3 EPU. Round 18 consists of 32 additional RAI questions, SRXB-91 through SRXB-122. TVA agreed to provide a response to RAI Round 18 by September 15, 2008. To facilitate the processing of TS-418, TVA requested a meeting with NRC to review the draft responses to SRXB-91 through SRXB-116 and to discuss the TVA approach to responding to SRXB-117 through SRXB-122. That meeting was held on August 7, 2008, and included a presentation by AREVA on behalf of TVA of the draft responses to RAIs SRXB-91 through 116 and a general discussion of the approach to SRXB-118 through 122. Time did not permit allow the discussion of SRXB-117; however, a teleconference with NRC on SRXB-117 has been requested. This submittal provides a partial response to Round 18, namely to SRXB-92, 93, 95, 96, 97, 99, 100, and 102 through 116. The remainder of the Round 18 RAI questions will be answered by September 15, 2008.

Enclosure 1 is a proprietary response to the listed RAIs and contains information that AREVA NP, Inc. (AREVA) considers to be proprietary in nature and subsequently, pursuant to 10 CFR 9.17(a)(4), 2.390(a)(4) and 2.390(d)(1), AREVA requests that such

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information be withheld from public disclosure. Enclosure 2 is a redacted version of Enclosure 1 with the proprietary material removed and is suitable for public disclosure. Enclosure 3 contains an affidavit from AREVA supporting this request for withholding from public disclosure.

TVA has determined that the additional information provided by this letter does not affect the no significant hazards considerations associated with the proposed TS changes. The proposed TS changes still qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

No new regulatory commitments are made in this submittal. If you have any questions regarding this letter, please contact me at (256)729-7658.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 15th day of August, 2008.

Sincerely,

J. E. Emens
Licensing Supervisor

Enclosures:
1. Partial Response to Round 18 Request for Additional Information (Proprietary Information Version)
2. Partial Response to Round 18 Request for Additional Information (Non-Proprietary Information Version)
3. Affidavit

Enclosures:
cc (Enclosures):

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ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE TS-418
EXTENDED POWER UPRATE (EPU)

PARTIAL RESPONSE TO ROUND 18 REQUEST FOR ADDITIONAL INFORMATION (NON-PROPRIETARY INFORMATION VERSION)

This enclosure provides TVA's a partial response to NRC's July 17, 2008, Round 18 Request for Additional Information (RAI) (ADAMS Accession No. (ML081700102).

NRC Introduction to Round 18 RAI

Table 1.3 in Enclosure 5 to the letter dated June 25, 2004, indicates that the COTRANSA2 Version AAPR03 computer code was used to evaluate the anticipated transient without scram (ATWS) – overpressurization event. The licensee cites a May 31, 2000, letter from the Nuclear Regulatory Commission (NRC) to Framatome (now AREVA) to support the use of COTRANSA2 for the ATWS-overpressurization abnormal operating occurrence (AOO).

NRC RAI SRXB-92 (Units 2 and 3)

Subcooled boiling is a phenomenon that can have a significant impact on the efficacy of a code system to accurately predict the axial power shape and bundle flow (by impacting the two phase pressure losses). Using a subset of the KATHY data, provide a similar comparison for the XCOBRA-T Levy subcooled void model as provided for the Ohkawa-Lahey correlation in response to NRC-RAI-SRXB-A.35.

Response to SRXB-92 (Units 2 and 3)

The Levy subcooled void model was active in XCOBRA-T when performing the analyses associated with the response to NRC RAI SRXB-A.35. The Levy subcooled boiling model does not directly predict void fraction in subcooled boiling. Instead, the model predicts a critical subcooling that defines the onset of boiling. The critical subcooling is used in conjunction with a profile fit model to determine the local flow quality that accounts for the presence of subcooled boiling. The local flow quality is then used in the Ohkawa-Lahey correlation to predict the void fraction in subcooled boiling.

NRC RAI SRXB-93 (Units 2 and 3)

Provide justification for the application of the Ohkawa-Lahey void-quality correlation to pressures above 6.9 MPa.

Response to SRXB-93 (Units 2 and 3)

AREVA demonstrated the applicability of the Ohkawa-Lahey void-quality correlation up to pressures of [ ] based on FRIGG data in the response to NRC RAI SRXB-A.35.

An extensive validation of the Ohkawa-Lahey void-quality correlation is provided in Reference SRXB-93.1, which is in addition to the validation performed by AREVA. This reference demonstrates the correlation performance over a very broad range of geometries and flow conditions.

The results reported by the authors demonstrate that the correlation has good agreement for all tests except for those tests where the hydraulic diameter is in excess of 0.5 feet (ft.) (0.152 meters (m)). However, the correlation is not applied to geometries of this size in any AREVA methodologies.

A measured versus predicted plot of void fraction containing all the data analyzed by the authors is reproduced in Figure SRXB-93.1. The data with hydraulic diameter in excess of 0.5 ft. (0.152 m) are indicated as "CARRIER DATA" and "HUGHES (B&W) DATA" in this figure. The statistical results and conclusions reported by the authors for geometries similar to BWR rod bundles are consistent with the AREVA assessment that was summarized in the response to NRC RAI SRXB-A.35. One set of data in this population extended above a pressure of [ ], which was reported by Petrick (Reference SRXB-93.2). The Petrick data provided experimental results for co-current down-flow at 600, 1000, and 1500 psia (4.1, 6.9, and 10.3 MPa) in a 1.939 inch (0.049 m) diameter tube.

The mean and standard deviation of the absolute error (measured-predicted void fraction) for the Ohkawa-Lahey void-quality correlation were reported as 0.038 and 0.050, respectively for the 1500 psia (10.3 MPa) data set in Reference SRXB-93.1. The reasonable agreement between the data and Ohkawa-Lahey predictions shows the extensibility of the correlation to pressures and flow conditions outside the range experienced in BWR rod bundles under typical operating and transient conditions including its extensibility to pressures up to 1500 psia (10.3 MPa).

In summary, the Ohkawa-Lahey void-quality correlation is justified for use above 1000 psia (6.9 MPa) based on validation of the correlation to a broad range of geometries and flow conditions including pressures up to 1500 psia (10.3 MPa).

References:

SRXB-93.1 B. Chexal, et al., "An Assessment of Eight Void Fraction Models for Vertical Flows," NSAC-107, Electric Power Research Institute, December 1986.

SRXB-93.2 Michael Petrick, "A Study of Vapor Carryunder and Associated Problems," ANL-6581, Argonne National Laboratory, July 1962.

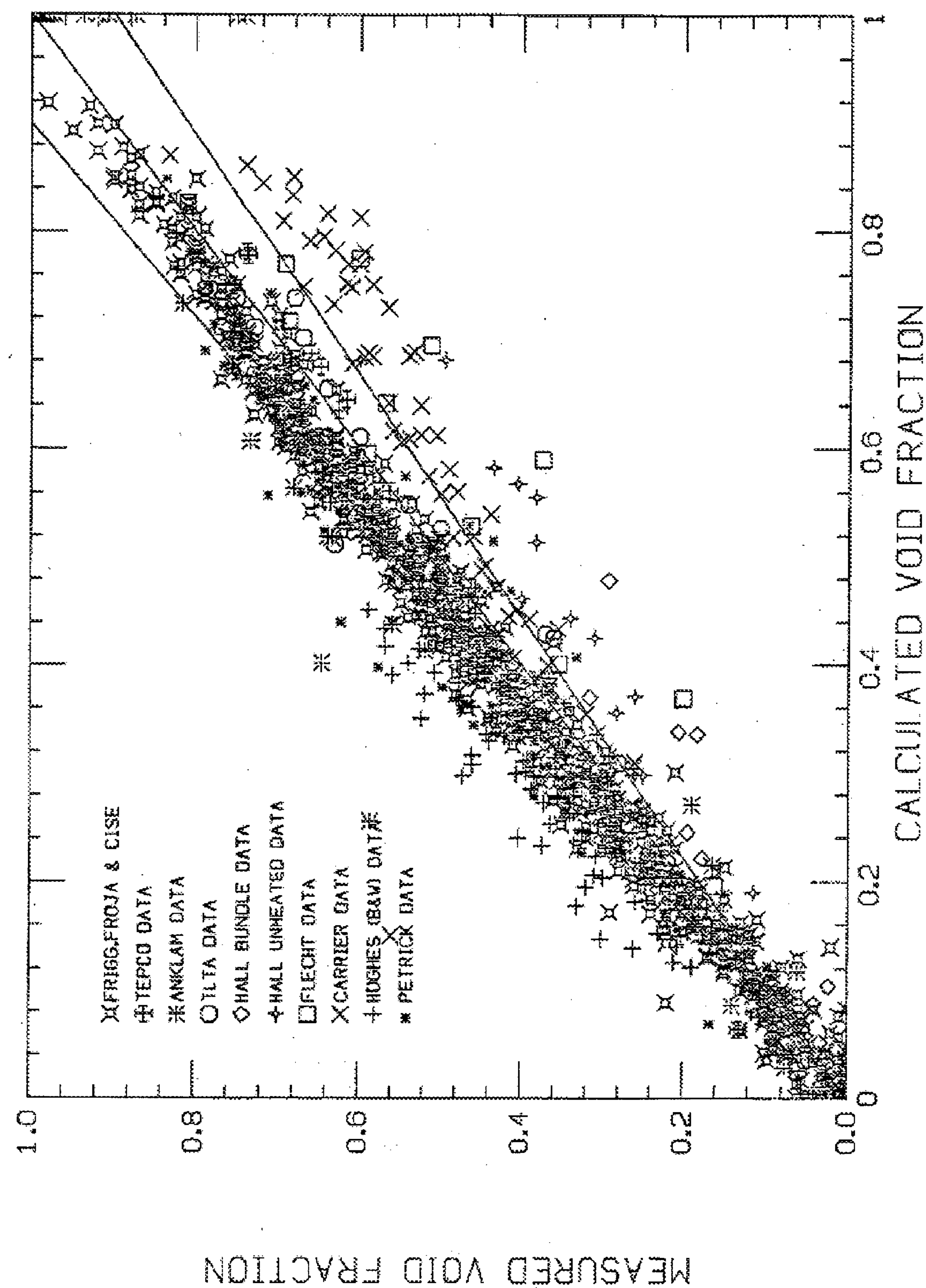

Figure SRXB-93.1 Comparison of Ohkawa-Lahey Void-Quality Correlation to Various Void Fraction Data

NRC RAI SRXB-95 (Units 2 and 3)

Provide details regarding the bypass flow during the ATWS overpressure transient as predicted by COTRANSA2. In particular determine if the bypass flow is downward during any portion of the transient. Justify the applicability of these results given any inherent constraints in the COTRANSA2 code.

Response to SRXB-95 (Units 2 and 3)

During the short-term ATWS analysis for peak pressure, the entering bypass flow remains strongly upwards due to the normal pressure gradients across the core. There are no inherent constraints on COTRANSA2 for upward bypass flow. The results from the COTRANSA2 ATWS analyses supporting the Browns Ferry EPU submittal were reviewed and it was confirmed that the bypass inlet flow remained positive throughout the analyses.

NRC RAI SRXB-96 (Units 2 and 3)

Provide a plot of the total safety relief valve (SRV) flow during ATWS overpressurization for EPU and pre-EPU conditions.

Response to SRXB-96 (Units 2 and 3)

A plot of total SRV flow during an ATWS overpressurization is provided below for EPU and non-EPU operation of a Browns Ferry Unit 3 Cycle 14 design.

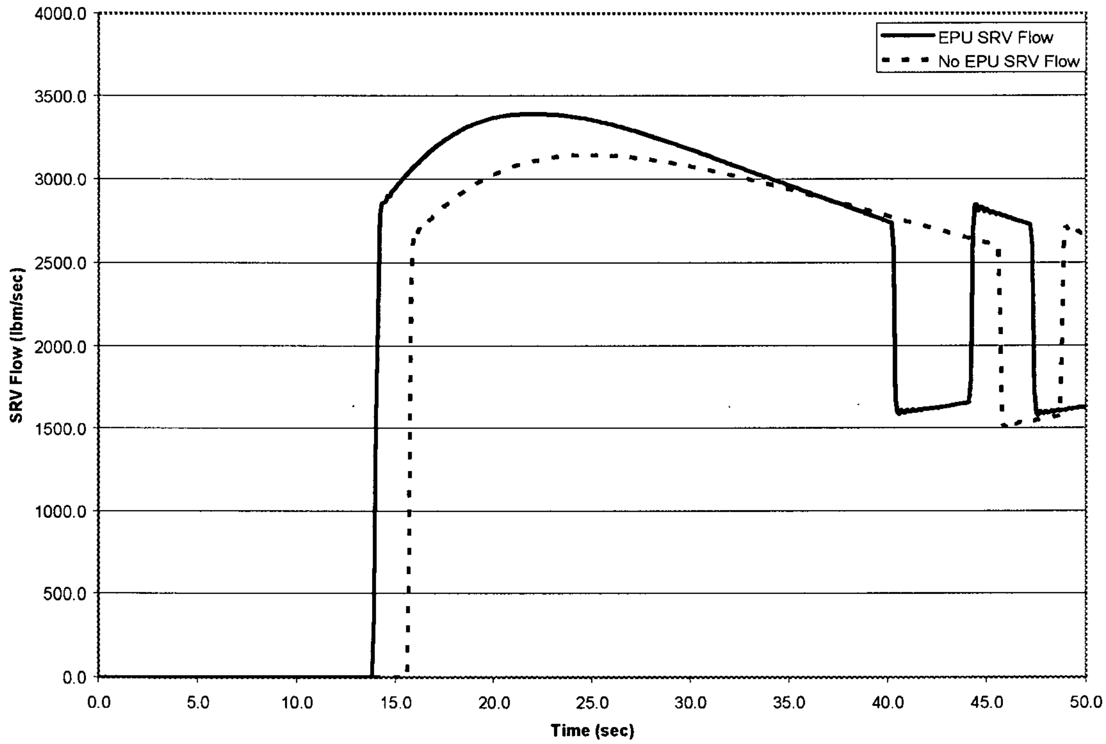

Figure SRXB-96.1 SRV Flow During ATWS Overpressurization

<u>NRC RAI SRXB-97 (Units 2 and 3)</u>

Condition 2 of the safety evaluation dated May 23, 1990, approving Advanced Nuclear Fuels (ANF) 913(P)(A), COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, August 1990 requires consideration of time step. Provide details regarding the nodalization of the steam line and time step used for ATWS overpressure analysis. Specifically provide results of sensitivity studies to verify that the node size and time steps are sufficient to preclude numerical errors in the calculation of the pressure wave propagation to the reactor core from the main steam isolation valves.

<u>Response to SRXB-97 (Units 2 and 3)</u>

Time step studies are performed for fast pressurization events to determine a converged solution. The parameter studied is [ ]. Converging on [ ] also ensures that other parameters including peak pressure are converged. The converged time step used for Browns Ferry is [ ] for the system and a steam line time step that is [ ] finer. Specifically to demonstrate the time step for the ATWS event, results are provided below for the Browns Ferry Unit 3 Cycle 14 EPU limiting ATWS analysis. The steam line nodalization is consistent with the Peach Bottom benchmark of Reference SRXB-97.1 and consists of [ ] nodes from the dome to the Main Steam Isolation Valves (MSIV). The time step study and the consistent nodalization of steam line from the original benchmark model are sufficient to preclude numerical errors in the calculation of the pressure wave propagation to the reactor core from the MSIVs.

Table SRXB-97.1 ATWS Time Step Study

Parameter	TS = [ ]	TS = [ ]	TS = [ ]	TS = [ ]
Peak vessel pressure, psig	1477	1478	1478	1477

References:

SRXB-97.1 ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,* Advanced Nuclear Fuels Corporation, August 1990.

NRC RAI SRXB-99 (Units 2 and 3)

The description of the steam line model in ANF-913(P)(A) does not provide descriptive details of the SRV model. Provide a description similar to Section 2.3.4 in ANF-913(P)(A) describing these models. In the description address the use of calculated pressures in modeling the SRV lift. When apportioning flow from a steam line node between an SRV and the downstream node, discuss how the SRV flow is calculated.

Response to SRXB-99 (Units 2 and 3)

The SRVs are modeled as a single mass sink from the steam line node where they are located. SRVs with the same setpoint are modeled as a combined valve in COTRANSA2. The net flow is the sum of the flow of each valve in parallel. Multiple SRVs are used in COTRANSA2 to represent banks of SRVs with different setpoints.

The SRVs are opened to relieve pressure when calculated pressure in the steam line node containing the valves reaches the opening setpoint pressure. The valve position is based on time since the valve setpoint was exceeded, the valve opening delay, and the valve stroke time.

Full open valve flow (*W*) is calculated as:

The COTRANSA2 input for [ ] is adjusted to account for the total pressure loss in the SRV branch from the main steam line to the SRV.

NRC RAI SRXB-100 (Units 2 and 3)

Section 2.1 of ANF-913(P)(A) states that cross sections are interpolated based on both controlled and uncontrolled states at [ ] void fraction. These void cases appear to not be consistent with the void cases used to develop cross section response surfaces for MICROBURN-B2 [ ], explain this discrepancy.

Response to SRXB-100 (Units 2 and 3)

The original COTRAN cross section tables used piece-wise linear interpolation of cross section tables at [ ] void fraction. With the introduction of MICROBURN-B the 1-D cross section table produced by the core simulator and used by COTRANSA2 was changed to be a [ ] This extension of the cross section interpolation was briefly described in the RAI Response 1 of ANF-913(P)(A) Volume 1 Supplement 4.

Specifically, the cross section table in COTRANSA2 now contains [ ] The [ ] For limiting events, the nodal void fractions typically change by 0.03 to 0.10 before the power response is dominated by the scram bank insertion. In this formulation, the cross sections used by COTRANSA2 are much more representative of the actual radial conditions in the core than if the cross sections are generated assuming all of the nodes in the plane are at [ ] percent void fraction.

The MICROBURN-B code used [ ] void fraction libraries from CASMO-3 and used [ ] to generate the nodal cross sections for the specific nodal operating history and instantaneous void fraction. MICROBURN-B2 uses [ ] void fraction libraries from CASMO-4 and uses [ ] to generate the nodal cross sections for the specific nodal operating history and instantaneous void fraction. In either case, it is the instantaneous cross sections for each node in the plane that are collapsed to planar average values to produce the cross sections used by the 1-D flux solution in COTRANSA2. In this formulation, the migration from MICROBURN-B to MICROBURN-B2 did not change the cross section representation between the core simulator and COTRANSA2.

NRC RAI SRXB-102 (Units 2 and 3)

In the response to SRXB-87 contained in the letter dated March 6, 2008, as supplemented in the letter dated May 1, 2008, TVA provided a sensitivity analysis to quantify the impact on the safety limit minimum critical power ratio (SLMCPR) of a potential increase in pin power peaking uncertainty at the harder spectrum EPU conditions.

XCOBRA-T is used to determine the transient effect on the critical power ratio (CPR). To calculate the critical heat flux ratio XCOBRA-T requires S-factors. The S-factor accounts for lattice peaking and bundle geometry effects. Describe how the additive constants and lattice calculations are used to determine the S-factors for use by XCOBRA-T. In particular, address whether the analyses are performed for local peaking using XFYRE, CASMO-4, or MICROBURN-B2.

Response to SRXB-102 (Units 2 and 3)

The CPR correlation used in XCOBRA-T is the SPCB correlation, documented in the approved Topical Report EMF-2209 Revision 2, *SPCB Critical Power Correlation,* September 2003). The SPCB CPR correlation does not use S-factors; it uses F-effective (FEFF) values to account for rod local peaking, rod location, and bundle geometry effects. The FEFF parameter is described in detail in Section 2.3 of Reference SRXB-102.1 and is composed of two parts. **[**

] The additive constants are determined from the experimental data. The FEFF values are not based on the lattice calculations from XFYRE or CASMO-4, they are determined based on the local peaking factors from **[** **]** from the MICROBURN-B2 calculations.

References:

SRXB-102.1 EMF-2209(P)(A) Revision 2, *SPCB Critical Power Correlation,* Framatome ANP, September 2003.

NRC RAI SRXB-103 (Units 2 and 3)

Provide the relationship of the term F_{eff} to the S-factor. If axial integration is required to determine the S-factors, specify how this is performed. Address whether the S-factors are sensitive to the bundle void distribution. Describe how the S-factors are determined for conditions typical (or bounding) for operation at EPU conditions.

Response to SRXB-103 (Units 2 and 3)

S-factors are not used. The SPCB CPR correlation used in XCOBRA-T depends only on the FEFF values. The FEFF values are not based on **[** **]**. The FEFF values are computed for **[**

] determined and used with SPCB.

NRC RAI SRXB-104 (Units 2 and 3)

Describe how S-factors are determined for part length rods.

Response to SRXB-104 (Units 2 and 3)

As mentioned in the responses to SRXB-102 and SRXB-103, S-factors are not used. The SPCB CPR correlation used in XCOBRA-T depends on the FEFF values. FEFF is calculated for [ ] as discussed in response SRXB-103 and in Reference SRXB-102.1. The FEFF values are computed [ ] of the part length rods. FEFF is [ ] for the vacant array locations above the top of the part length rods.

NRC RAI SRXB-105 (Units 2 and 3)

Verify that the Unit 2 transient analyses were performed using input options for closure relationships that are consistent with the NRC approval of XCOBRA-T. This includes specifying the Levy subcooled boiling model, the Martinelli-Nelson two phase friction multipliers, the two phase component loss multiplier, the wall viscosity model, and thermodynamic properties from the ASME steam tables.

Response to SRXB-105 (Units 2 and 3)

The Units 2 and 3 transient analyses used the default models consistent with the NRC approval of XCOBRA-T. The default models include Levy subcooled boiling model, the Martinelli-Nelson two phase friction multipliers, the two phase component loss multiplier, and the heated wall viscosity correction model. Thermodynamic properties from the ASME steam tables were used. The code provides a message if the default models are not used. Per AREVA's licensing analyses requirements, using models other than the default models would not be justified.

NRC RAI SRXB-106 (Units 2 and 3)

At EPU conditions, the core steam flow rate is increased. The pressure response to events such as turbine trip and load rejection is expected to be exacerbated at EPU conditions relative to pre-EPU conditions.

The NRC staff notes that the SPCB critical power correlation for ATRIUM-10 fuel is not qualified above [ ]. In the analysis of the pressurization transients, address whether the pressure exceed [ ] in the reactor core. Similarly compare the analysis conditions for those parameters listed in Condition 3 of the safety evaluation July 3, 2000, for EMF-2209(P)(A), *SPCB Critical Power Correlation*.

Response to SRXB-106 (Units 2 and 3)

Bounds checking is provided in the XCOBRA-T coding to ensure the conditions provided to the SPCB correlation are within the correlation limits as specified in Table 1.1 of EMF-2209(P)(A), *SPCB Critical Power Correlation.* Should any of the condition limits be violated, the behavior will be as specified in Section 2.6 of this document. In the specific case where the pressure limit is exceeded, XCOBRA-T will write an appropriate error message and terminate the calculation as specified in Section 2.6.3.

With respect to the remaining parameters, the behavior for transient calculations is summarized in the following table.

NRC RAI SRXB-107 (Units 2 and 3)

Address how the wall friction and component loss coefficients were determined for Unit 2. Address whether these parameters were input in the analysis to account for friction. Provide these parameters and the technical basis for their selection. Relative to pre-EPU conditions, channel flow tends to redistribute at EPU conditions as there are fewer low resistance bundles in the core. Address whether the friction parameters were selected to be consistent with this expected trend.

Response to SRXB-107 (Units 2 and 3)

Wall friction and component loss coefficients were determined for Units 2 and 3 based on single-phase testing of a prototypic ATRIUM-10 fuel assembly in the Portable Hydraulic Test Facility (PHTF). Prototypical fuel rods, spacer grids, flow channel, upper tie plate, and lower tie plate were used in the testing. A description of the PHTF facility and an overview of the process for determining the component loss coefficients are described in Reference SRXB-107.1. The assessment database for demonstrating the applicability of the single phase derived coefficients from the PHTF to two-phase flow conditions was demonstrated in the March 7, 2006, TVA response (ML060680583) to NRC RAI SRXB-A.15, and in Figure SRXB-A.15-1 in particular. This figure demonstrates that assembly conditions at EPU including Maximum Extended Load Line Limit Analysis plus (MELLLA +) are bounded by the tested two phase flow conditions.

The wall friction and component loss coefficients determined from the PHTF and utilized in Units 2 and 3 are provided in Table SRXB-107.1. The values are valid in pre-EPU and EPU conditions and account for friction. The values have been selected because they are representative of the hydraulic characteristics of actual ATRIUM-10 fuel assemblies loaded into the reactor.

Distribution of channel flow is explicitly accounted for in analyses at pre-EPU and EPU conditions by analyzing the flow distribution in the channels given the wall friction and component loss coefficients of each hydraulic channel, plus representative axial and radial power distributions for the plant conditions being analyzed. Therefore, friction parameters inherently account for the expected trend whereby channel flow redistributes at EPU conditions relative to pre-EPU conditions.

References:

SRXB-107.1 I.K. Madni, et al., "Development of Correlations For Pressure Loss/Drop Coefficients Obtained From Flow Testing of Fuel Assemblies In Framatome ANP'S PHTF," Paper Number 22428, Proceedings of ICONE10, Arlington, Va, April 14 -18, 2002.

Table SRXB-107.1 Hydraulic Characteristics of ATRIUM-10 Fuel Assemblies

1 Loss coefficients are referenced to the adjacent assembly bare rod flow area.

2 Although there are 8 spacers, the losses associated with the bottom spacer are combined with the orifice/LTP model, so that only 7 spacers are explicitly modeled for ATRIUM-10.

NRC RAI SRXB-108 (Units 2 and 3)

At EPU conditions there are a higher number of higher powered bundles. It is possible, and likely, for large axial sections of these bundles to be in an annular flow regime. Calculating pressure losses near bundle features such as fuel spacers can be important in the prediction of critical heat flux, which tends to occur below fuel spacers where the liquid film is typically thinnest.

On page 25 of Exxon Nuclear Company's XN-NF-84-105(P)(A), *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis*, it is stated that "[t]his [Martinelli-Nelson] formulation was developed for horizontal flow, but is reasonably accurate for vertical flow where both phasic flow rates are high enough to ensure turbulent co-current flow." Justify why the Martinelli-Nelson two phase friction multipliers are applicable in annular flow regimes.

Response to SRXB-108 (Units 2 and 3)

The Martinelli-Nelson two phase friction multiplier has been confirmed to be applicable in the annular flow regime by verifying the AREVA hydraulic models against two-phase full-scale heated bundle tests in the KATHY test facility in Karlstein, Germany. The range of hydraulic conditions is demonstrated in the March 7, 2006, TVA response (ML060680583) to NRC RAI SRXB-A.15, and Figure SRXB-A.15-1 in particular shows that assembly conditions at EPU (including MELLLA+) are bounded by the tested two-phase flow conditions. Many of the tested conditions are in the annular flow regime.

The AREVA hydraulic models (including the Martinelli-Nelson two phase friction multiplier) have been verified against two-phase full-scale heated bundle test data that are in the annular flow regime. Therefore, use of the Martinelli-Nelson two phase friction multiplier is justified for use in the annular flow regime within the AREVA hydraulic models.

NRC RAI SRXB-109 (Units 2 and 3)

Section 3.3 of the Technical Evaluation Report attached to the NRC's safety evaluation approving XN-NF-84-105(P)(A) states that critical power calculations may be inaccurate if the inlet flow is negative or if the inlet quality is above zero. Verify that for the transient analyses that the bundle inlet flow is positive and that the inlet qualities are less than zero.

Response to SRXB-109 (Units 2 and 3)

The critical power calculations for Browns Ferry fuel are made with the SPCB critical power correlation. Equivalent correlation bounds are specified in terms of [ ]. The range of applicability of these parameters is sufficiently broad to cover the ranges of conditions encountered during the licensing calculations. Correlation bounds checking is incorporated in the XCOBRA-T critical power calculations. The bounds checking routine does not allow a calculation outside the range of applicability of these parameters.

NRC RAI SRXB-110 (Units 2 and 3)

Transient cladding heat flux during transients will be a function of the heat hold-up in the fuel pins during AOOs. Identify all changes that have been made to the fuel rod thermal conduction models since the NRC's review and approval of RODEX2. Provide a comparison of the XCOBRA-T fuel rod models to those approved to the NRC during review of EMF-85-74(P), *RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model*, or RODEX2A. Also, provide a comparison of these models to those in BAW-10247(P), *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, or RODEX4.

Response to SRXB-110 (Units 2 and 3)

Fuel Rod Model Changes

NRC approval of the RODEX2 computer code is found in the Safety Evaluation Report associated with Reference SRXB-110.1.

NRC approval of a modification to the RODEX2 fission gas release model for application to BWRs is found in Reference SRXB-110.2. RODEX2A is acceptable for mechanical analyses. However, RODEX2 remains the approved model for Loss of Coolant Accident (LOCA) and transient analysis methodologies. As described in the Technical Evaluation Report associated with Reference SRXB-110.2, the RODEX2A code is based on the RODEX2 code with modifications only to the fission gas release model while the thermal and mechanical portions remain identical to the RODEX2 code.

The limitation on the **[**
] with NRC approval of Reference SRXB-110.3.

Use of RODEX2 for BWR applications (including LOCA and transient thermal-hydraulic analysis methodologies) to **[** **]** was acknowledged by the NRC in the Safety Evaluation Report Clarification associated with Reference SRXB-110.4.

Extension of RODEX2A burnup limit was approved in Reference SRXB-110.4. The Safety Evaluation Report for this document suggests that the RODEX2A code was "improved" relative to prior approvals. However, no changes were actually made to the code, only the benchmarking was extended to higher rod-average burnup. This point is made in Section 1.0 of the report.

With the exception of the changes associated with increasing the concentration of gadolinium that was approved in Reference SRXB-110.3, no changes (other than error corrections) have been made to the originally approved RODEX2 and RODEX2A models. Reproducibility of the RODEX2 and RODEX2A results relative to the results originally submitted to the NRC in References SRXB-110.1 and -110.2 were confirmed as recently as 2001 as described in Reference SRXB-110.5.

Since no updates have been made to RODEX2 or RODEX2A since the approval of XCOBRA-T, the fuel rod thermal conduction models approved for XCOBRA-T in Reference SRXB-110.6 are unchanged.

<u>Comparison of RODEX2 to RODEX4</u>

RODEX4 was submitted for review for use in thermal-mechanical calculations only and not for use in LOCA or transient applications. RODEX4 performs a best estimate calculation with both steady-state and transient temperature solution capability. All the RODEX4 fuel thermal-mechanical properties [ ] All the Zircaloy thermal-mechanical properties used in RODEX4, [ ]. RODEX4 properties are further described in the RODEX4 theory manual [EMF-2994(P)].

References:

SRXB-110.1 XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.

SRXB-110.2 XN-NF-85-74(P)(A) Revision 0, *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model,* Exxon Nuclear Company, August 1986.

SRXB-110.3 XN-NF-85-92(P)(A), *Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results*, Exxon Nuclear Company, November 1986.

SRXB-110.4 EMF-85-74(P) Revision 0, Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model*, Advanced Nuclear Fuels Corporation, February 1998.

SRXB-110.5 Correspondence, J.F. Mallay (Framatome) to U.S. Nuclear Regulatory Commission, "RODEX2 - Additional V&V and Consolidation of Approved Versions," NRC:01:025, September 10, 2001. (ADAMS Accession No. ML012610568)

SRXB-110.6 XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company, October 1986.

NRC RAI SRXB-111 (Units 2 and 3)

If different historical models are preserved in XCOBRA-T relative to RODEX2A, justify the use of XCOBRA-T to model transients for fuel above the previously established burnup limits for RODEX2.

Response to SRXB-111 (Units 2 and 3)

The approved rod average burnup limit is [ ] for both RODEX2 and RODEX2A. AREVA does not use RODEX2 or RODEX2A above this limit.

Approval for use of RODEX2 to [ ] was given for PWR applications in Reference SRXB-111.1. Use of RODEX2 for BWR applications (including LOCA and transient thermal-hydraulic analysis methodologies) to [ ] rod average burnup was subsequently acknowledged by the NRC in the Safety Evaluation Report clarification associated with Reference SRXB-111.2. The NRC Safety Evaluation for this topical also approves RODEX2A to [ ] rod average burnup.

References:

SRXB-111.1 ANF-81-58(P)(A), Supplements 3 and 4, Revision 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Advanced Nuclear Fuels Corporation, June 1990.

SRXB-111.2 EMF-85-74(P) Revision 0, Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model*, Advanced Nuclear Fuels Corporation, February 1998.

NRC RAI SRXB-112 (Units 2 and 3)

Some models may have been updated to conservatively bound experimental data collected subsequent to the NRC review and approval of RODEX2. The staff notes that certain assumptions may be conservative in the assessment of linear heat generation rate limits that may not be conservative when evaluating transient heat flux during AOO simulation due to the competing effects of reactivity feedback and heat flux flow mismatch. If a model is "conservatively bounding" in RODEX2, and translated to XCOBRA-T, provide a discussion of the performance of the model for thermal margin transient calculations.

Clarifications provided by the NRC following a meeting on August 7, 2008

The draft response for SRXB-112 deals with changes to the RODEX2 code in its first part, but requests additional information regarding the use of conservative assumptions in the abnormal operating occurrence (AOO) transient response. The discussion regarding the conservatism of the gap properties should be addressed in the response to the second part of RAI 112. See the second and third sentences:

> The staff notes that certain assumptions may be conservative in the assessment of linear heat generation rate limits that may not be conservative when evaluating transient heat flux during AOO simulation due to the competing effects of reactivity feedback and heat flux/flow mismatch. If a model is "conservatively bounding" in RODEX2, and

translated to XCOBRA-T, provide a discussion of the performance of the model for thermal margin transient calculations.

Summary of staff concern:

The NRC staff considered the coupling of the neutron flux and fluid conditions for AOO transient evaluations for both a reduced thermal time constant and an increased thermal time constant. When the time constant is over predicted, the fluid response to changing neutron power is lagged. A pressurization transient, therefore, would result in an increase in the reactor power that is not impeded by subsequent rapid void formation due to hold up of the heat flux in the pellet. An over prediction of the time constant will tend to increase the fission power for such a transient. However, the same effect of holding the heat up in the fuel pellet has the dual effect of reducing the cladding heat flux response; therefore, the ultimate effect on the transient critical power ratio (CPR) is a combination of the conservative prediction of peak neutron flux with the non-conservative prediction of the transient cladding heat flux.

For the case where the time constant is under predicted the inverse is true, the gross reactor power increase due to pressurization is limited due to more rapid void formation in response to the increasing neutron flux, but this is countered by a prediction of higher cladding surface heat flux relative to the pin power throughout the transient.

The input assumptions regarding the gas gap may increase or decrease the thermal resistance, and similarly, an increase or decrease in the thermal resistance does not have a clear impact on the transient predicted CPR due to competing effects in the cladding heat flux and void reactivity.

Response to SRXB-112 (Units 2 and 3)

No RODEX2 models have been updated subsequent to the NRC review and approval of RODEX2, except those associated with the NRC approved update to the **[** **]** (Reference SRXB-112.1). For instance, the NRC Safety Evaluation approving RODEX2 for **[** **]** for PWR applications in Reference SRXB-112.2 identified that no models were changed relative to original approval in Reference SRXB-112.3. Instead, only additional comparisons were made to demonstrate the applicability of the code to **[** **]**. Note, the same RODEX2 code version is used for both PWR and BWR applications.

Reproducibility of the RODEX2 results relative to the results originally submitted to the NRC in Reference SRXB-112.3 was confirmed as recently as 2001 as described in Reference SRXB-112.4.

For the pressurization transients, gap properties from RODEX2 are used to model the core average gap conductance in COTRANSA2 and the hot channel gap conductance in XCOBRA-T. The core average gap conductance in the system model is not the same as the gap conductance of the hot channel. The gap conductance of the system model is based on the average of all fuel in the core; whereas the hot channel gap conductance is based on a limiting assembly. The gap conductance is a function of exposure.

The COTRANSA2 system model includes the neutronic feedback from a change in the thermal time constant. **[**

] The XCOBRA-T hot channel model uses the boundary conditions from COTRANSA2. A higher hot channel gap conductance in XCOBRA-T is conservative for pressurization events since higher values increase the heat flux and coolant quality, and thereby decrease the margin to boiling transition. These trends have been confirmed with AREVA sensitivity calculations. Transient analysis results are more sensitive to a change in hot channel gap conductance than core average gap conductance.

For fuel rods early in life, the gap is not closed. AREVA applies the following conservatism for gap conductance for fuel rods with significant open gaps: [

]

References:

SRXB-112.1 XN-NF-85-92(P)(A), *Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results*, Exxon Nuclear Company, November 1986.

SRXB-112.2 ANF-81-58(P)(A) Revision 2, Supplements 3 and 4, *RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model,* Advanced Nuclear Fuels Corporation, June 1990.

SRXB-112.3 XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.

SRXB-112.4 Correspondence, J.F. Mallay (Framatome) to U.S. Nuclear Regulatory Commission, "RODEX2 - Additional V&V and Consolidation of Approved Versions," NRC:01:025, September 10, 2001. (ADAMS Accession No. ML012610568)

NRC RAI SRXB-113 (Units 2 and 3)

At EPU conditions a core contains a higher number of higher powered bundles. At these conditions the heat transfer may be driven by phenomena such as liquid entrainment and redeposition. The XCOBRA-T heat transfer is predicted according to Dittus-Boelter and Thom heat transfer correlations for forced convection and nucleate boiling, respectively. The NRC is aware that AREVA has temperature data from full scale critical power tests. Provide qualification of the heat transfer correlations to predict fuel rod surface temperatures for test conditions representative of higher powered bundles at flow rates similar to EPU conditions for ATRIUM-10 [i.e. ~5 – ~8 megawatts thermal]. Particularly, provide the relevant qualification data near the top of the bundle where liquid entrainment and droplet redeposition is expected to have an impact on heat transfer.

Response to SRXB-113 (Units 2 and 3)

The thermocouples used for measuring temperature data in full scale critical power tests are [

] measure heat transfer coefficients associated with pre-Critical Heat Flux heat transfer in the range of mass and heat fluxes associated with BWRs. As a result, no relevant qualification studies of the Dittus-Boelter and Thom heat transfer correlations can be performed from the test data.

As noted in Reference SRXB-113.1, fully developed nucleate boiling is relatively insensitive to flow rate and quality. However, "boiling suppression" may occur in high quality annular flow that provides very high heat transfer coefficients, resulting in decreasing wall temperature as the heat flux increases.

Extracted heat transfer information from experiments in a tube for a range of pressure, flow, and quality that is relevant to BWRs is reported in Reference SRBX-113.2. This reference shows the relative insensitivity of heat transfer coefficient to flow rate and quality and that boiling suppression does not become significant until quality reaches approximately 0.47, which is well above the range of interest to BWRs. [

] Therefore, it is concluded that liquid entrainment and droplet redeposition does not have an impact on boiling heat transfer for flow conditions that are applicable to an operating BWR at EPU and MELLLA+.

References:

SRXB-113.1 R.T. Lahey Jr. and F.J. Moody, *The Thermal-Hydraulics of a Boiling Water Nuclear Reactor,* Second Edition, American Nuclear Society, Hinsdale, IL, 1993.

SRXB-113.2 Y. Aounallah, "Boiling Suppression in Convective Flow," Proceedings of ICAPP '04, Pittsburgh, PA, June 13–17, 2004, Paper 4251.

NRC RAI SRXB-114 (Units 2 and 3)

XCOBRA-T accepts input from COTRANSA2 to capture the transient variation in power during AOOs. EPU cores are high energy cores and may have a higher peak hot excess reactivity, resulting in changes in SCRAM worth relative to pre-EPU conditions. Provide a more detailed description of how effects such as transient variation in axial power shape during SCRAM are captured in XCOBRA-T. Address whether detailed nodal power histories translated from COTRANSA2 to XCOBRA-T.

Clarifications provided by the NRC following a meeting on August 7, 2008

The transient varying axial power shape (TVAPS) phenomenon is a flow reduction effect caused by the rapid void collapse when the power is suppressed in the bottom part of the fuel bundles as the control rods insert during a scram. The channel flow stagnates as it occupies the collapsed void region and then continues to pick up energy as it traverses to the top of the fuel bundle. The fluid enthalpy at the top of the channel may lead to dryout conditions.

The response should specify if this effect is captured (or observed) in the COTRANSA2/XCOBRA-T analyses.

Additionally, information should be provided regarding the conservatism. The response should discuss how the limiting initial axial power shapes are established given operational flexibility during cycle depletion.

Response to SRXB-114 (Units 2 and 3)

During pressurization transients, the axial power shape shifts due to the void collapse in the top of core (void reactivity), the core flow increase in the bottom of core (void reactivity), and control rod insertion in the bottom of the core (scram reactivity). The coupled 1-D neutronic and thermal-hydraulic core model in COTRANSA2 determines a [ ] including the impact of void collapse and scram. This [ ] is used in XCOBRA-T. The assembly [ ] and the assembly radial peaking factor input to XCOBRA-T. The change in axial nodal power [ ] results in a change in fuel rod surface heat flux and the energy transferred to the coolant at each axial node. Both COTRANSA2 and XCOBRA-T have fuel rod heat transfer models that determine the fuel rod surface heat flux based on the nodal power history and the coolant conditions at each axial node.

Both COTRANSA2 and XCOBRA-T have thermal-hydraulic models that are used to calculate the flow at each axial node in the core and the hot channel during the pressurization transient. The energy equation captures the effect of changes in fuel rod surface heat flux on coolant conditions. The mass and momentum equations, with applicable correlations, are used to determine the local coolant flow rate during the pressurization transient. During the initial phase of the pressurization transient, these models predict a decrease in flow near the top of the core and an increase in flow near the bottom of the core. Note that although the flow decreases in the upper portion of the hot assembly, the assembly flow does not stagnate during the pressurization phase of an AOO or ATWS. Local fluid conditions (enthalpy and flow) calculated from the thermal-hydraulic model are used to determine local dryout conditions.

The initial axial power shape is determined from the COTRANSA2 calculation based on the cross section data for the core exposure considered in the analysis. The cross section data is obtained from the MICROBURN-B2 computer code. The MICROBURN-B2 calculations used to generate cross section data for COTRANSA2 licensing calculations are performed assuming that all control rods are fully withdrawn. Assuming all control rods are fully withdrawn results in a significant conservatism in calculated scram reactivity for exposure conditions with some control rods partially inserted.

NRC RAI SRXB-115 (Units 2 and 3)

At EPU conditions a larger number of bundles are operated a high power levels. It is expected, therefore, for more bundles to be near their thermal limits at the onset of a transient. Describe how the Unit 2 radial channel nodalization was evaluated. Address whether the potentially limiting bundles were grouped with non-limiting bundles. Provide the radial bundle power distribution (as predicted by MICROBURN-B2) and radial channel group assignments (in XCOBRA-T) for the initial conditions for the limiting transient analysis. Also, compare the

COTRANSA2 radial grouping with the XCOBRA-T grouping. Provide justification that the resolution is sufficient to model the transient behavior in all potentially limiting bundles.

Clarifications provided by the NRC following a meeting on August 7, 2008

The appropriateness of the XCOBRA-T nodalization for transient LHGR analysis should be explicitly addressed.

Response to SRXB-115 (Units 2 and 3)

The Units 2 and 3 radial channel nodalization was evaluated as follows:

- Each bundle is uniquely modeled hydraulically and neutronically in MICROBURN-B2 to establish input to the transient calculation at the desired state point. The limiting assemblies are identified from the MICROBURN-B2 results prior to running the transient. Table SRXB-115.1 provides the predicted radial power distribution for the limiting transient analysis of a recent EPU calculation for Unit 3 Cycle 14 (this is the same unit and cycle that is the basis for other RAI sensitivity studies).

- The fuel assemblies in the core are collapsed to 1-D hydraulically and neutronically in COTRANSA2 analyses.

- With the exception of the most limiting assembly in each group of hydraulically equivalent assemblies, hydraulically equivalent assemblies are collapsed into their respective groups in the XCOBRA analyses. Multiple calculations are then performed on the power of the most limiting assembly to determine its relationship between assembly power and the inlet flow rate. Because assembly flow is very nearly linear with assembly power, there is no sensitivity to the amount of grouping in the XCOBRA analyses when determining the relationship between assembly power and the inlet flow rate.

- Individual limiting assemblies are analyzed in the XCOBRA-T delta-CPR calculations; there is no grouping. The limiting assemblies are identified by having the [ ] for each unique fuel type. In situations where different assemblies of the same type (e.g., ATRIUM-10) are being considered as limiting, the FEFF defining the fuel type is the limiting FEFF of the different assemblies. The initial flow rate in each calculation is consistent with the relationship between assembly power and the flow rate calculated by XCOBRA. The transient hydraulic response in XCOBRA-T is driven by the lower-to-upper plenum pressure response predicted by COTRANSA2. Table SRXB-115.2 identifies the limiting assemblies for which delta-CPR was analyzed by XCOBRA-T in the Unit 3 Cycle 14 limiting transient analysis. Note that only one assembly is listed because this cycle is an all ATRIUM-10 core. Table SRXB-115.2 also provides the radial peaking factor for the limiting assembly from the MICROBURN-B2 analysis as well as the converged radial peaking factor from the XCOBRA-T analysis that was determined to result in dryout during the transient.

- In addition to delta-CPR, the XCOBRA-T hot channel analysis is used to determine the peak Linear Heat Generation Rate (LHGR) during the transient. The XCOBRA-T hot channel analysis determines the assembly radial peaking factor that results in dryout during the transient. The peak axial LHGR in the fuel assembly during the transient is also calculated in this analysis. However, because the maximum allowed steady state LHGR may be higher then the initial LHGR used in the XCOBRA-T hot channel analysis, the peak transient LHGR may be higher than calculated in the XCOBRA-T analysis. The peak LHGR for the transient is determined [

] If necessary, the steady state LHGR limit is adjusted to ensure that the peak transient LHGR remains below the transient LHGR limit established from RODEX2 analyses.

In summary, the above discussion shows that the transient analysis methodology has sufficient resolution to model the transient behavior in all potentially limiting bundles and is justified for use in analyzing the limiting transient analyses at EPU conditions.

EDIT OF POWER IN UNITS OF 10** 0

	IR: 1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
JR:	PRA																													
60									0.293	0.344	0.370	0.405	0.402	0.416	0.431	0.427	0.425	0.410	0.413	0.389	0.360	0.302								
58								0.378	0.632	0.792	0.754	0.907	0.907	0.833	0.936	0.936	0.821	0.908	0.911	0.755	0.799	0.638	0.399							
56						0.258	0.416	0.704	0.946	1.035	1.093	1.122	1.134	1.133	1.130	1.129	1.133	1.135	1.125	1.098	1.040	0.949	0.701	0.420	0.272					
54						0.418	0.810	0.971	0.944	1.098	1.014	1.155	1.040	1.159	1.005	1.005	1.159	1.033	1.158	1.027	1.103	0.947	0.972	0.812	0.423					
52					0.436	0.718	1.004	1.107	1.145	0.972	1.179	1.000	1.205	1.011	1.222	1.223	1.013	1.206	1.002	1.183	0.976	1.145	1.108	1.004	0.714	0.421				
50			0.261	0.419	0.713	0.982	0.939	1.109	0.997	1.176	0.994	1.199	1.030	1.227	1.040	1.049	1.229	1.022	1.201	0.994	1.175	0.975	1.110	0.946	0.981	0.716	0.419	0.262		
48			0.422	0.816	1.010	0.945	1.135	0.971	1.176	1.006	1.196	1.020	1.235	1.052	1.267	1.269	1.058	1.238	1.023	1.197	0.996	1.175	0.973	1.137	0.954	1.012	0.818	0.439		
46		0.403	0.712	0.981	1.119	1.122	0.978	1.180	1.002	1.199	1.014	1.225	1.047	1.260	1.069	1.077	1.264	1.056	1.227	1.014	1.201	1.006	1.182	0.977	1.124	1.121	0.982	0.711	0.382	
44	0.308	0.647	0.958	0.949	1.163	1.015	1.202	1.019	1.210	1.033	1.240	1.059	1.260	1.059	1.271	1.273	1.068	1.264	1.051	1.243	1.047	1.214	1.019	1.202	1.010	1.163	0.955	0.951	0.637	0.297
42	0.364	0.804	1.049	1.115	1.001	1.199	1.022	1.226	1.042	1.241	1.060	1.274	1.077	1.272	1.070	1.067	1.274	1.074	1.277	1.064	1.246	1.050	1.227	1.016	1.199	1.001	1.112	1.037	0.785	0.345
40	0.391	0.746	1.107	1.027	1.198	1.010	1.218	1.043	1.253	1.063	1.296	1.102	1.286	1.091	1.265	1.265	1.087	1.286	1.104	1.300	1.075	1.256	1.032	1.218	1.015	1.198	1.028	1.095	0.677	0.377
38	0.409	0.915	1.135	1.171	1.014	1.216	1.031	1.241	1.063	1.282	1.102	1.310	1.093	1.282	1.058	1.057	1.279	1.084	1.311	1.107	1.285	1.061	1.243	1.039	1.220	1.011	1.166	1.124	0.901	0.401
36	0.411	0.916	1.149	1.053	1.222	1.032	1.249	1.059	1.270	1.073	1.293	1.074	1.294	1.082	1.278	1.275	1.076	1.292	1.091	1.296	1.077	1.273	1.069	1.254	1.051	1.222	1.043	1.140	0.907	0.414
34	0.410	0.831	1.149	1.176	1.036	1.244	1.068	1.270	1.064	1.277	1.077	1.288	1.084	1.280	1.062	1.045	1.267	1.088	1.291	1.085	1.280	1.067	1.275	1.063	1.246	1.028	1.171	1.140	0.819	0.401
32	0.450	0.948	1.146	1.026	1.243	1.061	1.279	1.071	1.268	1.051	1.269	1.064	1.282	1.062	1.257	0.963	1.057	1.281	1.069	1.271	1.062	1.273	1.092	1.284	1.070	1.240	1.010	1.137	0.937	0.426
30	0.430	0.948	1.147	1.028	1.243	1.055	1.279	1.072	1.270	1.061	1.270	1.068	1.280	1.054	0.963	1.258	1.065	1.282	1.065	1.271	1.064	1.272	1.081	1.282	1.059	1.240	1.010	1.138	0.940	0.428
28	0.415	0.834	1.151	1.178	1.030	1.245	1.060	1.272	1.069	1.279	1.079	1.289	1.074	1.266	1.045	1.057	1.280	1.070	1.289	1.082	1.279	1.068	1.273	1.063	1.244	1.038	1.173	1.144	0.823	0.442
26	0.413	0.922	1.153	1.058	1.226	1.048	1.253	1.070	1.274	1.077	1.296	1.076	1.292	1.075	1.276	1.279	1.082	1.293	1.085	1.294	1.072	1.272	1.072	1.251	1.036	1.221	1.052	1.148	0.918	0.424
24	0.424	0.925	1.141	1.176	1.019	1.222	1.034	1.244	1.060	1.287	1.104	1.314	1.088	1.283	1.058	1.058	1.282	1.087	1.310	1.100	1.281	1.061	1.241	1.031	1.216	1.010	1.171	1.136	0.921	0.417
22	0.395	0.771	1.114	1.033	1.204	1.016	1.221	1.034	1.257	1.076	1.304	1.112	1.291	1.093	1.270	1.269	1.090	1.288	1.090	1.296	1.064	1.252	1.029	1.215	1.008	1.195	1.032	1.108	0.767	0.399
20	0.359	0.809	1.056	1.122	1.015	1.203	1.025	1.229	1.035	1.249	1.079	1.284	1.075	1.281	1.067	1.067	1.279	1.087	1.277	1.061	1.243	1.049	1.227	1.020	1.193	0.978	1.112	1.048	0.805	0.363
18	0.301	0.643	0.965	0.967	1.170	1.008	1.206	1.022	1.214	1.040	1.250	1.063	1.272	1.088	1.281	1.279	1.072	1.267	1.054	1.244	1.044	1.213	1.038	1.200	0.992	1.155	0.952	0.954	0.646	0.301
16		0.384	0.729	0.992	1.126	1.127	0.985	1.184	1.001	1.206	1.022	1.236	1.057	1.273	1.075	1.077	1.269	1.057	1.232	1.017	1.202	1.001	1.182	0.974	1.118	1.115	0.978	0.704	0.390	
14			0.473	0.827	1.016	0.949	1.139	0.970	1.179	1.008	1.204	1.035	1.245	1.069	1.276	1.276	1.068	1.246	1.042	1.201	1.006	1.176	0.974	1.134	0.948	1.008	0.815	0.423		
12			0.266	0.426	0.716	0.984	0.943	1.113	0.985	1.181	1.000	1.205	1.019	1.235	1.045	1.046	1.237	1.046	1.208	0.996	1.178	0.984	1.109	0.940	0.979	0.715	0.421	0.273		
10					0.440	0.716	1.006	1.111	1.149	0.975	1.185	0.997	1.210	1.028	1.228	1.228	1.030	1.216	1.011	1.185	0.976	1.146	1.108	1.002	0.713	0.415				
8						0.417	0.812	0.975	0.953	1.104	1.024	1.160	1.041	1.162	1.004	1.005	1.166	1.060	1.163	1.017	1.101	0.944	0.973	0.810	0.414					
6						0.258	0.418	0.705	0.949	1.038	1.095	1.122	1.135	1.132	1.128	1.130	1.137	1.140	1.127	1.097	1.038	0.949	0.711	0.417	0.256					
4								0.381	0.642	0.794	0.741	0.906	0.905	0.819	0.930	0.932	0.833	0.910	0.912	0.750	0.797	0.636	0.385							
2									0.299	0.351	0.369	0.398	0.423	0.407	0.413	0.413	0.402	0.416	0.409	0.385	0.351	0.313								

Table SRXB-115.1 Radial Bundle Power Distribution
As Predicted by MICROBURN-B2 for the Limiting Transient Analysis

Table SRXB-115.2 Limiting Assemblies Analyzed in the Unit 3 Cycle 14 Limiting Transient Analysis

Limiting Bundle Information	
Initial MCPR	1.5905
Location in Core I, J	23, 24
MB2 radial power	1.3138
XCT converged radial	1.5465

NRC RAI SRXB-116 (Units 2 and 3)

Address whether XCOBRA-T was used to demonstrate acceptable fuel rod thermal mechanical performance during transients. If XCOBRA-T is not used for this purpose, address how acceptable thermal mechanical performance is demonstrated during transients. If the method is not consistent with the models in RODEX2 or later NRC-approved thermal mechanical code, justify the approach.

Clarifications provided by the NRC following a meeting on August 7, 2008

Aside from describing the method for normalization of the transient LHGR to the initial LHGR, provide some additional minor clarifications:

(1) The decay heat contribution will remain essentially static during the transient, address whether the normalization capture the varying rod decay heat sources;

(2) Specify the source of the decay heat constants (i.e. ANS standard);

(3) The rod power distribution is flattened due to gamma smearing of the thermal power, address how these gamma smeared power fractions are calculated; and

(4) Address how the direct moderator heat is accounted for.

The response should also provide a detailed description of the rod heat flux calculation for bundles with part length fuel rods, and address the code change as well as items 1-4 for each region (fully rodded, plena region, above plena region).

Response to SRXB-116 (Units 2 and 3)

XCOBRA-T was used to demonstrate acceptable fuel rod thermal-mechanical performance during transients (AOOs). The fuel rod models in XCOBRA-T are consistent with RODEX2 and the fuel rod gap conductance values input to XCOBRA-T are obtained from RODEX2 analyses. The XCOBRA-T analyses are performed to ensure that fuel rod thermal-mechanical limits established with RODEX2 are not exceeded during AOOs.

An average fuel rod (local rod peaking = 1.0) is modeled in the XCOBRA-T hot channel analysis. As discussed in response SRXB-114, the assembly axial nodal power history in the XCOBRA-T analysis is based on the COTRANSA2 calculated core axial nodal power history and the assembly radial peaking factor. The power generated in each axial segment of the bundle average fuel rod is the assembly axial nodal power divided by the number of heated fuel rods in the axial node. Generally, the initial peak axial fuel rod LHGR in the XCOBRA-T hot channel analysis is less than the maximum allowed steady state LHGR. [

]

The decay heat is calculated by COTRANSA2 and is included in the total core power versus time provided as a boundary condition to XCOBRA-T. The decay heat model used in COTRANSA2 is a curve fit (11 groups) to the 1973 ANS standard decay heat model. The COTRANSA2 core power boundary condition includes the decay heat contribution based on the core average power density. The decay heat power remains essentially constant during the transient. Therefore, the decay heat during the transient is primarily a function of initial power density. Application of power peaking factors (axial, radial, local) to the COTRANSA2 average power properly accounts for local decay heat in the XCOBRA-T hot channel analysis.

Gamma smearing does not affect the XCOBRA-T hot channel calculation. The hot channel calculation models an average fuel rod (the average is not affected by flattening of the distribution). The calculation process for determining the peak transient LHGR is equivalent [
] and is
not dependent of the actual rod local peaking factor.

The total core power calculated by COTRANSA2 is distributed between the fuel rod, the active channel coolant, and the core bypass coolant. The fraction deposited in each component is based on fuel type specific calculations performed with the CASMO computer code. XCOBRA-T calculates the fuel rod surface heat flux using a fuel rod heat conduction model, the power generated in the fuel rod, and the fluid conditions at the surface of the rod. The power generated in the fuel rod is described in Reference SRXB-116.1 Section 2.5.5. The power generated in each axial section of a fuel rod is calculated using Equation 2.130 from Reference SRXB-116.1. Although Reference SRXB-116.1 states that Equation 2.130 is calculated for each axial node, the equation itself does not denote which variables are axially dependent. Because the equation is for each axial node, the variables for heat generation rate, axial peaking factor, and number of rods are axial dependent. At the time Reference SRXB-116.1 was prepared, the number of rods at each axial plane was a constant for the fuel designs being supplied. For the ATRIUM-10 fuel design with part length fuel rods (PLFRs), the number of rods became axial dependent and the code was modified to make application of Equation 2.130 correct and consistent with the NRC-approved Reference SRXB-116.1. For application to current fuel designs, a better definition of the variable N_r in Equation 2.130 would be "number of *heated* rods per assembly *at the axial plane*" (italic indicates added text).

For assembly axial nodes with PLFRs containing UO_2, the PLFRs are included in the variable N_r. For assembly axial nodes in the plena region of the PLFRs and above the PLFRs, N_r does not include the PLFRs. Equation 2.130 is correctly applied for all axial planes when the variable N_r is the number of heated rods at the axial plane.

The discussions in the preceding paragraphs are applicable for all axial regions of assemblies with PLFRs.

References:

SRXB-116.1 XN-NF-84-105(P)(A) Volume 1 Supplement 4, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis - Void Fraction Model Comparison to Experimental Data*, Advanced Nuclear Fuels Corporation, June 1988.

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE TS-418
EXTENDED POWER UPRATE (EPU)

PARTIAL RESPONSE TO ROUND 18 REQUEST FOR ADDITIONAL INFORMATION

AREVA AFFIDAVIT

This enclosure provides AREVA's affidavit for Enclosure 1.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA NP information contained in Engineering Information Record 51-9086777-000, "Responses to NRC RAI – Round 18 for Browns Ferry EPU SRXB-92, -93, -95 to -100, -102 to -116," dated August 2008, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP 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 NP 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 NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP'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 NP.

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

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

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

7. In accordance with AREVA NP'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 NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP 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 15th
day of August, 2008.

____
Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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3. I am familiar with the AREVA NP information contained in the Draft Engineering Information Record 51-9086777-000, "Responses to NRC RAI – Round 18 for Browns Ferry EPU SRXB-91 to -116," used as information in a meeting with the NRC on August 7, 2008, and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP 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 NP 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.

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SUBSCRIBED before me this 15th

day of August, 2008.

Sherry L. McFaden
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MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010