ATTACHMENT 35

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

Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information

ANP-2860NP
Revision 2

Supplement 2NP
Revision 1

Extension for Use of ATRIUM 10XM Fuel for Extended Power Uprate

August 2015

AREVA Inc.

Nature of Changes

Item	Page	Description and Justification
1.	3-1, A-4 through A-6, B-1, and B-2	Adjustments are made to proprietary markings.

Contents

Tables

Figures

Nomenclature

ACE	AREVA's advanced critical power correlation
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BOL	beginning of life
BWR	boiling water reactor
CHF	critical heat flux
CPR	critical power ratio
DIVOM	delta-over-Initial CPR versus oscillation magnitude
ECCS	emergency core cooling system
EPU	extended power uprate
KATHY	Karlstein thermal hydraulic test facility
LHGR	linear heat generation rate
LOCA	loss of coolant accident
LPCS	low pressure core spray
LPRM	local power range monitor
LRNB	load reject with no bypass
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MWR	metal water reaction
NRC	Nuclear Regulatory Commission, U. S.
OLMCPR	operating limit minimum critical power ratio
PCT	peak cladding temperature
PLFR	part length fuel rod
SLMCPR	safety limit minimum critical power ratio
SER	safety evaluation report
SPCB	AREVA (formerly Siemens Power Corporation) critical power correlation
TCD	thermal conductivity degradation
TIP	traversing incore probe
WREM	water reactor evaluation model

1 Introduction

This document reviews the AREVA approved licensing methodologies to demonstrate that they are applicable to licensing and operation of the Browns Ferry Nuclear Plant with ATRIUM™* 10XM in the extended power uprate (EPU) operating domain with a representative power/flow operating map in Figure 1-1. General applicability of AREVA licensing methods to the Browns Ferry units in the EPU operating domain was discussed in the main document ANP-2680P Revision 2 and applicability of these licensing methods to the ATRIUM 10XM fuel design in the current operating domain was discussed in ANP-2680P Revision 2, Supplement 1P, Revision 0. Application of the three methods added for ATRIUM 10XM (SAFLIM3D, ACE and RODEX4) for EPU applications are addressed in this document. EPU methods application issues with no fuel type sensitivity are addressed in the base report and are not repeated in this supplement.

This document applies to all 3 Browns Ferry units since all 3 Browns Ferry BWR/4s are either identical or have only minor differences. The most significant difference between the units is the core loadings and corresponding core designs. The impact of the differences in core designs between units and cycles is addressed in the cycle specific reload report for each unit. Minor differences between the plants and units do not impact the application of AREVA's methodology as presented in this document.

A review of the RAI's received from license applications for the introduction of ATRIUM 10XM at EPU conditions did not identify anything that has not already been addressed.

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* ATRIUM is a trademark of AREVA Inc.

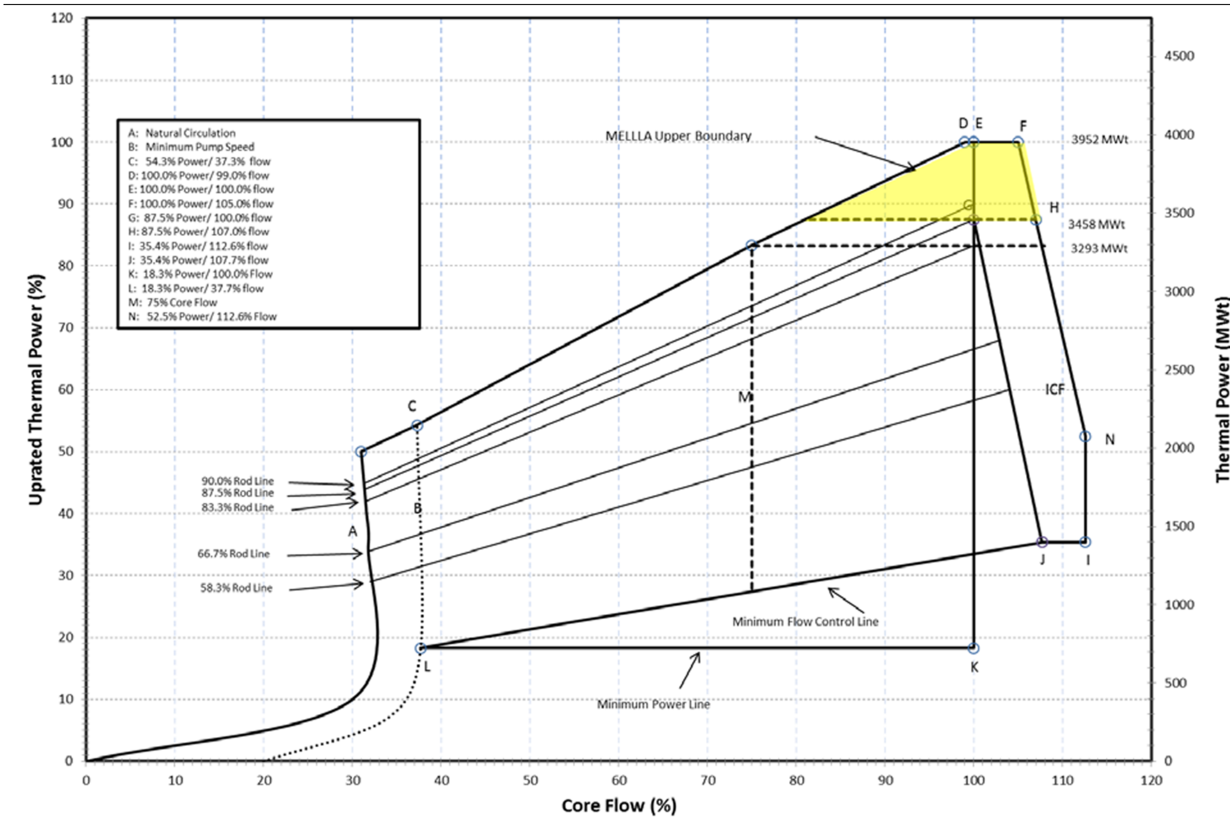

Figure 1-1 Browns Ferry Power Flow Operating Map

2 Overview

ANP-2860P Revision 2 included many sensitivity analyses for ATRIUM-10 at EPU power levels. The results and conclusions of the ATRIUM-10 remain applicable to AREVA methods with ATRIUM 10XM at EPU due to the similarity of design characteristics. These design characteristics are explicitly accounted for in all of the models. The differences in fuel design characteristics between the ATRIUM-10 and ATRIUM 10XM are discussed in ANP-2860P Rev. 2, Supplement 1P.

The first step in determining the applicability of current licensing methods to Browns Ferry operating conditions was a review of AREVA BWR topical reports listed in Table 2-1 to identify SER restrictions. This review identified that there are no SER restrictions on core power level or core flow for the AREVA topical reports. The review also indicated that there are no SER restrictions on the parameters most impacted by operation at EPU power level at any core flow rate: steam flow, feedwater flow, jet pump M-ratio, and core average void fraction.

The second step consisted of an evaluation of the core and reactor conditions experienced under Browns Ferry operating conditions to determine any challenges to the validity of the models. Operating margin for variations in the reactor power within the constraints of the power/flow map is mitigated to a large extent by variations in the limiting assembly radial power factor. A decrease in the limiting assembly radial power factor is necessary since the thermal operating limits (MCPR, MAPLHGR and LHGR) that restrict assembly power are insensitive to the core thermal power.

Based on these fundamental characteristics each of the major analysis domains (thermal-hydraulics, mechanics, core neutronics, transient analysis, LOCA and stability) are assessed to determine any challenges to application.

Operating experience with other plants that are operating at EPU conditions similar to the EPU that is proposed for Browns Ferry have demonstrated that the reactor models behave as expected.

Table 2-1 AREVA Licensing Topical Reports

Document Number	Document Title
XN-NF-79-56(P)(A) Revision 1 and Supplement 1	"Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981
XN-75-32(P)(A) Supplements 1 through 4	"Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, October 1983. (Base document not approved.)
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
XN-NF-81-51(P)(A)	"LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, May 1986
XN-NF-85-67(P)(A) Revision 1	“Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel,” Exxon Nuclear Company, September 1986
XN-NF-85-74(P)(A)	"RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model" Exxon Nuclear Company, August 1986
XN-NF-85-92(P)(A)	"Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986
ANF-89-98(P)(A) Revision 1 and Supplement 1	"Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995
ANF-90-82(P)(A) Revision 1	"Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A)	"RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998
EMF-93-177(P)(A) Revision 1	"Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005
BAW-10247PA Revision 0	"Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2	"Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983

Table 2-1 AREVA Licensing Topical Reports *(Continued)*

Document Number	Document Title
XN-NF-80-19(P)(A) Volume 4 Revision 1	"Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986
EMF-CC-074(P)(A) Volume 1	"STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994
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
EMF-CC-074(P)(A) Volume 4, Revision 0	"BWR Stability Analysis Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000
BAW-10255PA Revision 2	"Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP, May 2008
EMF-3028P-A Volume 2 Revision 4	“RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2: Theory Manual,” AREVA NP, May, 2013
XN-NF-79-59(P)(A)	"Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983
XN-NF-80-19(P)(A) Volume 3 Revision 2	"Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987
EMF-2245(P)(A) Revision 0	"Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000
EMF-2209(P)(A) Revision 3	"SPCB Critical Power Correlation," AREVA NP, September 2009
ANP-10298PA Revision 0	"ACE/ATRIUM 10XM Critical Power Correlation," AREVA, March 2010
ANP-3140(P) Revision 0	“Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation,” AREVA NP, August 2012

Table 2-1 AREVA Licensing Topical Reports *(Continued)*

Document Number	Document Title
ANP-10307PA Revision 0	"AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011
XN-CC-33(A) Revision 1	"HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual," Exxon Nuclear Company, November 1975
XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C	"Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982
XN-NF-82-07(P)(A) Revision 1	"Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, November 1982
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, February 1987
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
ANF-91-048(P)(A)	"Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," Advanced Nuclear Fuels Corporation, January 1993
ANF-91-048(P)(A) Supplements 1 and 2	"BWR Jet Pump Model Revision for RELAX," Siemens Power Corporation, October 1997
EMF-2292(P)(A) Revision 0	"ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000
EMF-2361(P)(A) Revision 0	"EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, May 2001
ANF-1358(P)(A) Revision 3	"The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005

3 Thermal Hydraulic Analysis

AREVA assembly thermal-hydraulic methods are qualified and validated against full-scale heated bundle tests in the KATHY test facility in Karlstein, Germany. The KATHY tests are used to characterize the assembly two-phase pressure drop and CHF performance. This allows the hydraulic models to be verified for AREVA fuel designs over a wide range of hydraulic conditions prototypic of reactor conditions.

The matrix of test conditions for the ATRIUM 10XM fuel design in KATHY is compared to reactor conditions in Figure 3-1. This figure illustrates that the test conditions bound typical assembly conditions as well as anticipated EPU operation for Browns Ferry. The data is based upon the projected EPU operating conditions for the Browns Ferry reactors. Figure 3-1 also shows that the key physical phenomena (e.g. fluid quality and assembly flows) for Browns Ferry EPU operating conditions are consistent with current reactor experience.

This similarity of assembly conditions is further enforced in AREVA analysis methodologies by the imposition of SPCB and ACE critical power correlation limits and, therefore, core designs must remain within the same parameter space. Since the bundle operating conditions for Browns Ferry are within the envelope of hydraulic test data used for model qualification and operating experience, the hydraulic models used to compute the core flow distribution and local void content remain valid for Browns Ferry at EPU operating conditions.

3.1 *Void Quality Correlation*

Qualification of the void-quality correlation used in AREVA methodologies was presented in Section 5 of ANP-2860P Revision 2 using ATRIUM-10 measurements. **[**

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Qualification of the void-quality correlation used in AREVA methodologies using ATRIUM 10XM measurements was presented in Section 2.2 of ANP-2860P Revision 2 Supplement 1P. Figure

2-2 of ANP-2860P Revision 2 Supplement 1P is repeated here as Figure 3-4. As noted in Section 2.2.4 good agreement is achieved between predicted and measured void fraction with measured ATRIUM 10XM void fractions up to **[ ]** .

Figure 3-1 Comparison of KATHY Two-Phase Pressure Drop and Void Fraction Test Matrices and Typical Browns Ferry Reactor Conditions with EPU

Figure 3-2 Void Fraction Correlation
Comparison to FRIGG and ATRIUM-10 Test Data

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Figure 3-3 [ ] Void-Quality Compared to Correlation Behavior

Figure 3-4 Validation of Dix-Findlay using ATRIUM-10 and ATRIUM 10XM Void Data

4 AREVA CHF/CPR Correlations

All AREVA CHF and CPR correlations are approved by the NRC staff to be applicable over specified ranges of assembly operating conditions. The NRC staff also approved specific corrective actions when the computed conditions fall outside of the approved range to assure that conservative calculations are obtained. For Browns Ferry operating conditions, some analyses can predict assembly conditions to be outside the approved range of specified conditions for the CHF correlations. Consequently, the AREVA licensing methods are programmed to determine whether the computed assembly conditions fall outside of the approved range of applicability for the CHF correlation and impose approved corrective actions as appropriate to conservatively assess the critical power margin for the assembly. The CPR correlation used for the ATRIUM 10XM fuel is the ACE/ATRIUM 10XM critical power correlation (References 1 and 2). Reference 2 was approved in the Safety Evaluation provided in Reference 3. The corrective actions for when the computed conditions fall outside the approved range are provided in Reference 1.

Applicability of SPCB at EPU conditions for ATRIUM-10 fuel was previously addressed in ANP-2860P Rev 2.

5 Safety Limit MCPR

The safety limit MCPR (SLMCPR) methodology is used to determine the Technical Specification SLMCPR value that ensures that 99.9% of the fuel rods are expected to avoid boiling transition during normal reactor operation and anticipated operation occurrences. The SLMCPR methodology for Browns Ferry is described in Reference 4. The SLMCPR is determined by statistically combining calculation uncertainties and plant measurement uncertainties that are associated with the calculation of MCPR. The thermal hydraulic, neutronic, and critical power correlation methodologies are used in the calculation of MCPR. The applicability of these methodologies for Browns Ferry operating conditions is discussed in other sections of this report.

AREVA calculates the SLMCPR on a cycle-specific basis to protect all allowed reactor operating conditions. The analysis incorporates the cycle-specific fuel and core designs. The initial MCPR distribution of the core is a major factor affecting how many rods are predicted to be in boiling transition. The MCPR distribution of the core depends on the neutronic design of the reload fuel and the fuel assembly power distributions in the core. AREVA SLMCPR methodology specifies that analyses be performed with a design basis power distribution that "… conservatively represents expected reactor operating states which could both exist at the MCPR operating limit and produce a MCPR equal to the MCPR safety limit during an anticipated operational occurrence." (Reference 4, Section 3.3.2).

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The impact that a flatter core power distribution may have on the SLMCPR is explicitly accounted for by the methodology. EPU operation tends to lead to a flatter core power distribution; [

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A plant specific application of the SAFLIM3D method (Reference 4) with ATRIUM 10XM fuel at EPU power is provided in Reference 5.

Figure 5-1 Assembly Power Distribution for Limiting Case in Safety Limit MCPR Analysis

6 Mechanical Limits Methodology

The LHGR limit is established to support plant operation while satisfying the fuel mechanical design criteria. The methodology for performing the fuel rod evaluation is described in Reference 6. Fuel rod design criteria evaluated by the methodology are contained in References 6 and 7.

Fuel rod power histories are generated as part of the methodology for equilibrium cycle conditions as well as cycle-specific operation. These power histories include the impact of channel bow using the same model and limitations as previously described in Section 5 (Safety Limit MCPR). A comprehensive number of uncertainties are taken into account in the categories of operating power uncertainties, code model parameter uncertainties, and fuel manufacturing tolerances. In addition, adjustments are made to the power history inputs for possible differences in planned versus actual operation. Upper limits on the analysis results are obtained for comparison to the design limits for fuel melt, cladding strain, rod internal pressure and other topics as described by the design criteria.

Since the power history inputs, which include LHGR, fast neutron flux, reactor coolant pressure and reactor coolant temperature, are used as input to the analysis, the results explicitly account for conditions at EPU such as higher coolant voiding and offsets in axial power and neutron fast flux. The resulting LHGR limit is used to monitor the fuel so it is maintained within the same maximum allowable steady-state power envelope as analyzed.

The methodology was applied to both the transition fuel cycle and equilibrium fuel cycle for the Browns Ferry units at EPU. These design analyses were performed for the ATRIUM 10XM fuel design and demonstrated compliance to the design criteria found in References 6 and 7. A summary of these design analyses and the compliance to each of the design criteria is found in Reference 8.

7 Core Neutronics

The AREVA neutronic methodologies (Reference 9) are characterized by technically rigorous treatment of phenomena and are very well benchmarked (>100 cycles of operation plus gamma scan data for ATRIUM-10). Recent operating experience is tabulated in Table 7-1. These tables present the reactor operating conditions and in particular the average and hot assembly powers for both US and European applications. As can be seen from this information, the average and peak bundle powers in this experience base exceed that associated with the Browns Ferry application of ATRIUM 10XM at EPU conditions.

For Browns Ferry operation the high powered assemblies in uprated cores will be subject to the same LHGR, MAPLHGR, MCPR, and cold shutdown margin limits and restrictions as high powered assemblies in pre-EPU designs at Brown Ferry.

7.1 *Shutdown Margin*

The design process for developing a target cold critical eigenvalue for the Browns Ferry fuel/core design was discussed in Section 7.1 of ANP-2860P Revision 2. Benchmarking of the Browns Ferry cycles resulted in the cold critical data presented in Figure 7-1 of ANP-2860P Revision 2. Figure 7-1 of this supplement updates this figure with the most recent cycles of Browns Ferry. The target cold critical eigenvalues used for the latest reference cycle design concentrated more heavily on the later operating cycles which is typical of the selection process. The resulting target cold critical eigenvalues used for the Browns Ferry reference design is shown in Table 7-2 and Figure 7-2. This determination of the target, together with a conservatively chosen design goal, ensures conservative determination of shutdown margin for the design.

Fuel manufacturing parameters, core modeling of critical conditions, and control blades are not affected by EPU therefore the shutdown margin acceptance criteria (0.38% $\Delta k/k$) remains applicable.

Experience with other plants that have transitioned to ATRIUM 10XM and operating at EPU have not shown any significant change in the cold target eigenvalue as demonstrated in Figure 7-3 and Figure 7-4.

7.2 *LHGR Monitoring of Advanced Fuel Designs*

Through various interactions between AREVA and the NRC, the NRC has requested verification that certain detailed models available with MICROBURN-B2 are utilized in the modeling of advanced fuel designs. These models include the impact of LPRM detectors (instrument tube) on the surrounding fuel rods and the impact of modeling the plenum region above the end of the heated portion of the part-length rods. The explicit LPRM model is used in the core monitoring to account for perturbations to the local peaking factors of rods surrounding the LPRM, hence rod power biases due to the presence of LPRM detectors are accounted for in the monitoring of LHGR limits. Monitoring for conformance with the operating limit LHGR for all fuel types including the ATRIUM 10XM will include explicit modeling of the fission gas plena in the node directly above the top of PLFR active fuel length. This provides the confirmation that the NRC has requested.

7.3 *Bypass Boiling*

The conclusion of ANP-2860P Revision 2 concerning bypass boiling remains applicable to the ATRIUM 10XM fuel design operating at EPU since the pressure drop characteristics of this fuel design are essentially the same as the ATRIUM-10.

The level of bypass boiling for a given state-point is a direct result of the hydraulic solution. The potential for boiling increases as the power/flow ratio increases or the inlet subcooling decreases. While the licensing methodology utilizes a **[** **]** to estimate the potential for localized bypass boiling. This **[** **]** to specifically determine a bounding local void distribution in the bypass. The model is conservative in that it **[** **]**. Review of the edit of bypass channel exit void for a Browns Ferry equilibrium cycle case identified the worst exposure statepoint with a few assemblies with insignificant (< 0.005 void fraction) bypass channel exit void at a cycle exposure of 14,752.8 MWd/MTU. To force more boiling in the bypass the inlet subcooling was set to a value of 20

BTU/lbm (compared to the typical value of 27 for this statepoint) at the 14,752.8 MWd/MTU statepoint to demonstrate the capability of this model to predict localized bypass boiling. The results are presented in Figure 7-5 through Figure 7-7.

Figure 7-5 presents the average void fraction for the channel bypass and Figure 7-6 presents the core exit bypass channel void fraction. One of the more significant impacts of voiding in the bypass is the impact on the LPRM reading. The average void fraction of the four channels surrounding any LPRM location is presented in Figure 7-7. Since insignificant boiling is observed at any LPRM location for normal operating conditions, there is no impact on LPRM readings due to EPU.

7.4 *Normal Operation*

The conclusion of ANP-2860P Revision 2 concerning normal operation continue to remain applicable to the ATRIUM 10XM fuel design operating at EPU since the thermal hydraulic conditions are independent of fuel design.

From a neutronic perspective, moderator density (void fraction) and exposure cause the greatest variation in cross sections. Reactor conditions for Browns Ferry at EPU are not significantly different from that of current experience and are bounded by the experience for the important parameters. Browns Ferry operating conditions (Figure 7-8, Figure 7-9, and Figure 7-10) can be compared to the equivalent data of the topical report EMF-2158(P)(A). Comparison of Figure 7-8 vs. Figure C-19 of ANP-2638P Revision 2 (Reference 10) and Figure 7-9 vs. Figure C-20 of ANP-2638P Revision 2 shows that Browns Ferry operation is within the range of the original methodology approval for assembly power and exit void fraction.

The axial profile of the power and void fraction of the limiting assembly and core average values at the exposure statepoint that experiences the highest exit void fraction during the cycle are presented in Figure 7-10 for a Browns Ferry design. These profiles demonstrate that the core average void fraction and the maximum assembly power void fractions are bounded by the topical report data and are consistent with recent experience on other reactors.

Figure 7-11 presents a histogram of the void fraction for the same statepoint presented in Figure 7-10 which is typical for Browns Ferry EPU conditions. This histogram was taken at the point of maximum exit void fraction expected during the cycle. The distribution of voids is shifted slightly toward the 70 -80 % void fraction levels. The population of nodes experiencing 85 -90% voids is still small.

The neutronic and thermal hydraulic conditions predicted for the Browns Ferry EPU operation are bounded by the data provided in the topical report EMF-2158(P)(A). Concerns about Pu production with high voids are not relevant as the isotopic validation presented in the topical report continues to be applicable to Browns Ferry EPU operation.

7.5 *Power Distribution Uncertainty*

A Browns Ferry specific assessment of the power distribution uncertainties was presented in Section 7.3 of ANP-2860P Revision 2. The database has been updated since that report was released incorporating additional data for the most recent cycles.

The Browns Ferry specific $\delta T'_{ij}$ database is shown versus cycle number in Figure 7-12, versus power to flow ratio in Figure 7-13, versus core void in Figure 7-14, and versus core power in Figure 7-15. These figures represent the same data with different independent variables. The database includes 112 full core gamma TIP measurements: 14 for Unit 1 Cycles 7 and 8, 46 for Unit 2 Cycles 13, through 15 (through February 2008), and 52 for Unit 3 Cycles 11 through 13 (to September 2007). Figure 7-14 represents the database consisting of Unit 1 Cycles 7 and 8, Unit 2 Cycles 14 and 15, and Unit 3, Cycles 12 and 13. Void fraction data for Unit 2 Cycle 13 and Unit 3 Cycle 11 was not readily available.

Figure 7-12 through Figure 7-15 clearly demonstrate that the D-lattice radial TIP uncertainty reported on page 9-8 for “TIP Distribution Calculation” in the Reference 9 topical report is very conservative for Browns Ferry. Figure 7-12 through Figure 7-15 also clearly demonstrate there is no correlation in the Browns Ferry specific uncertainty component due to the core power to flow ratio, core power or core average void fraction. Operation at the maximum core power and minimum core flow conditions allowed for EPU operations corresponds to a power to flow ratio of 38.95 MW-th/Mlb/hr which is within the range of the data already taken.

Operating experience with two reloads of ATRIUM 10XM fuel in another reactor operating at EPU has demonstrated no significant change in the uncertainty of the predicted detector response relative to the measurements as demonstrated in Figure 7-16 and Figure 7-17.

The primary concern about EPU has been higher void fractions. This section demonstrates that there does not exist a trend that would indicate that the power distribution uncertainties increases with higher void fraction.

Table 7-1 CASMO-4/MICROBURN-B2 Operating Experience

Reactor	Reactor Size, #FA	Power, MWt (% Uprated)*	Ave. Bundle Power, MWt/FA	Approximate Peak Bundle Power, MWt/FA
A	592	2575 (0.0)	4.4	7.2
B	592	2575 (0.0)	4.4	7.4
C	532	2292 (0.0)	4.3	7.3
D	840	3690 (0.0)	4.4	7.5
E	500	2500 (15.7)	5.0	8.0
F	444	1800 (5.9)	4.1	7.3
G	676	2928 (8.0)	4.3	7.6
H	700	3300 (9.3)	4.7	8.0
I	784	3840 (0.0)	4.9	8.1
J	624	3237 (11.9)	5.2	7.8
K	648	3600 (14.7)	5.6	8.6
L	648	2500 (10.1)	3.9	6.9
M	624	3091 (6.7)	5.0	7.7
N	800	3898 (1.7)	4.9	7.7
O	764	3952 (20.0)	5.2	7.7
P	560	2923 (20.0)	5.2	8.0
Q	724	2957 (17.8)	4.1	6.6
R‡	764	3952 (20.0)	5.2	7.8

* Latest power uprates.
‡ Browns Ferry with ATRIUM 10XM

Table 7-2 Browns Ferry Target Cold Critical Eigenvalue

Cycle Exposure (MWd/MTU)	k-eff
0.0	0.9940
20,000.0	0.9940

Figure 7-1 Browns Ferry Cold Critical Data

Figure 7-2 Browns Ferry Cold Critical Data Used For Cold Target Determination

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Figure 7-3 Cold Critical Data for an EPU Plant Transition to ATRIUM 10XM

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Figure 7-4 Cold Critical Data for an EPU Plant Transition to ATRIUM 10XM

Figure 7-5 MICROBURN-B2 Multi-Channel Average Bypass Void Distribution from a Browns Ferry Equilibrium Cycle Design

Figure 7-6 MICROBURN-B2 Multi-Channel Exit Bypass Void Distribution from a Browns Ferry Equilibrium Cycle Design

Figure 7-7 MICROBURN-B2 Multi-Channel Bypass Void at an LPRM Location from a Browns Ferry Equilibrium Cycle Design

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Figure 7-8 Maximum Assembly Power in a Browns Ferry EPU Design with ATRIUM 10XM

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Figure 7-9 Maximum Exit Void Fraction in a Browns Ferry EPU Design with ATRIUM 10XM

Figure 7-10 Browns Ferry Design Axial Profile of Power and Void Fraction

Figure 7-11 Browns Ferry Design Nodal Void Fraction Histogram

Figure 7-12 Browns Ferry δT'ij Gamma TIP Response vs. Cycle Number

Figure 7-13 FN δT'ij Gamma TIP Response vs. Power/Flow Ratio

Figure 7-14 Browns Ferry δT'ij Gamma TIP Response vs. Core Average Void Fraction

Figure 7-15 Browns Ferry δT'ij Gamma TIP Response vs. Core Power

Figure 7-16 TIP Statistics for Operating Cycles from Another BWR/4 During the Transition from ATRIUM-10 to ATRIUM 10XM Fuel

Figure 7-17 TIP Statistics for Operating Cycles from Another BWR/4 During the Transition from ATRIUM-10 to ATRIUM 10XM Fuel

8 Transient Analysis

The core phenomena of primary interest for limiting transients in BWRs are void fraction/quality relationships, determination of CHF, pressure drop, reactivity feedbacks and heat transfer correlations. One fundamental validation of the core hydraulic solution is separate effects testing against KATHY transient CHF measurements. The transient benchmark to time of dryout for prototypic Load Reject with no Bypass (LRNB) and pump trip transients encompass the transient integration of the continuity equations (including the void-quality closure relation), heat transfer, and determination of CHF. Typical benchmarks to KATHY (Figure 8-1) illustrate that the transient hydraulic solution and application of ACE (AREVA Critical Power Correlation) result in conservative predictions of the time of dryout. The measured data is taken from ATRIUM 10XM tests.

Outside of the core, the system simulation relies primarily on solutions of the basic conservation equations and equations of state. The models associated with predicting the pressure wave are general and have no limitation within the range of variation associated with Browns Ferry EPU operation.

The reactivity feedbacks are validated by a variety of means including initial qualification of advanced fuel design lattice calculations to Monte Carlo results as required by SER restrictions, steady-state monitoring of reactor operation (power distributions and eigenvalue), and the Peach Bottom 2 turbine trip benchmarks that exhibited a minimum of 2% conservatism in the calculation of integral power.

From these qualifications and the observation that the nodal hydraulic conditions during EPU operation are expected to be within the current operating experience, the transient analysis methods remain valid.

Appendix A, Section A.3.1, provides a summary of the impact of thermal conductivity degradation on transient analysis and corrective actions taken in the Browns Ferry analyses.

Figure 8-1 Typical Hydraulic Benchmarks to KATHY Transient Simulations (time to dryout)

9 LOCA Analysis

The conclusion of ANP-2860P Revision 2 concerning LOCA continue to remain applicable to the ATRIUM 10XM fuel design operating at EPU since the stored energy and decay heat characteristics are essentially the same as the ATRIUM-10.

LOCA results are strongly dependent on local power and are weakly dependent on core average power. As discussed in previous sections, maximum local power is not significantly changed due to EPU because the core is still constrained by the same thermal limits. The parameters associated with EPU that may impact LOCA results at each of the core flow rates in the operating domain are: increased core average initial stored energy, decreased initial coolant inventory, relative flow distribution between highest power and average power assemblies, and increased core decay heat.

BWR LOCA analyses are not sensitive to initial stored energy. During the blowdown phase the heat transfer remains high and the stored energy is removed prior to the start of the heatup phase. Initial inventory differences may impact LOCA event timing and the minimum inventory during blowdown prior to refill of the reactor vessel. However, any impact on event timing or minimum inventory would be smaller than the impact associated with the different size breaks that are already considered in the break spectrum analyses. At the elevated powers associated with EPU conditions, the difference in flow between the highest power assembly and the average power assembly is reduced. Therefore, these parameters do not change the range of conditions encountered or the capability of the codes to model LOCA at EPU conditions.

The potential impact of the EPU on LOCA analyses is thus primarily associated with the increase in decay heat levels in the core. For the EXEM BWR-2000 LOCA methodology the decay heat is conservatively modeled. The 11 group decay heat equation curve fit to the 1971 draft ANS standard for fission product decay heat from the WREM model is used to calculate fission product decay heat during blowdown. The draft ANS standard values are used for spray cooling and reflood. The required multiplier of 1.2 is applied to the fission product decay heat throughout the LOCA scenario. The models used for decay heat calculations are valid for EPU.

From the above discussion and the observation that nodal thermal-hydraulic conditions during EPU are expected to be within the current operating experience, the LOCA methods remain applicable for EPU conditions.

Independent of EPU, additional modifications have been made to the approved EXEM BWR-2000 LOCA methodology to more accurately model advanced fuel designs and to address regulatory concerns with the approved methodology. These modifications are described in Appendix B. Appendix A summarizes the assessment of thermal conductivity degradation in the Browns Ferry LOCA analyses.

10 Stability Analysis

10.1 *Linear Stability*

The conclusion of ANP-2860P Revision 2 concerning stability continue to remain applicable to the ATRIUM 10XM fuel design operating at EPU since the time constant of this fuel design is essentially the same as the ATRIUM-10.

The flatter radial power profile characteristic of EPU core designs will tend to decrease the first azimuthal eigenvalue separation and result in slightly higher regional decay ratios. These effects are computed by STAIF as it directly computes the channel, global, and regional decay ratio and does not rely on a correlation to protect the regional mode.

STAIF has been benchmarked against full assembly tests (in KATHY facility) to validate the channel hydraulics from a decay ratio of approximately 0.4 to limit cycles. These tests or benchmarks exceed the bounds of allowed operation. These benchmarks include prototypical ATRIUM-10 assemblies. From a reactor perspective, STAIF is benchmarked to both global and regional reactor data, and includes current reactor cycle and fuel design elements. This strong benchmarking qualification and the direct computation of the regional mode assure that the impact of the EPU core designs are reflected in the stability analysis.

The prototypical ATRIUM-10 benchmark analyses presented above apply to ATRIUM 10XM as well since the differences between ATRIUM-10 and ATRIUM 10XM are below the level of detail included in those benchmarks.

10.2 *DIVOM*

RAMONA5-FA has been generically approved to calculate DIVOM for EPU operation (Reference 11 and 12).

11 **ATWS**

The conclusion of ANP-2860P Revision 2 concerning ATWS continue to remain applicable to the ATRIUM 10XM fuel design operating at EPU since the void reactivity and doppler characteristics of this fuel design are essentially the same as the ATRIUM-10.

The COTRANSA2 computer code is the primary code used for the ATWS overpressurization analysis. The ATWS overpressurization event is not used to establish operating limits for critical power; therefore, the critical power correlation(s) pressure limit is not a factor in the analysis.

Dryout conditions are not expected to occur for the core average channel that is modeled in COTRANSA2 for the ATWS overpressurization analysis. Dryout might occur in the limiting (high power) channels of the core during the ATWS event; however, these channels are not modeled in COTRANSA2 analyses. For the ATWS overpressurization analysis, ignoring dryout for the hot channels is conservative in that it maximizes the heat transferred to the coolant and results in a higher calculated pressure.

The ATWS event is not limiting relative to acceptance criteria identified in 10 CFR 50.46. The core remains covered and adequately cooled during the event. Following the initial power increase during the pressurization phase, the core returns to natural circulation conditions after the recirculation pumps trip and fuel cladding temperatures are maintained at acceptably low levels. The ATWS event is significantly less limiting than the loss of coolant accident relative to 10 CFR 50.46 acceptance criteria.

Appendix A (Section A.3.3) summarizes biases developed to address NRC concerns regarding Void-Quality Correlation, Thermal Conductivity Degradation (TCD) and Doppler mismatch for AREVA overpressurization analyses.

11.1 ***ATWS Instability***

The ATWS instability event analysis was presented in Section 2.2 of ANP-2860P Revision 2. This analysis applies equally to ATRIUM 10XM due to the similarity of the fuel rod design. The phenomena that limit the oscillations in the ATRIUM-10 are essentially the same for the ATRIUM 10XM.

12 Summary

Similar to ANP-2860P Revision 2, this review concluded that there are no SER restrictions on AREVA methodology that are impacted by EPU at Browns Ferry. All of the arguments presented in ANP-2860P Revision 2 apply equally to cores loaded with ATRIUM 10XM. Since the EPU core and assembly conditions for the Browns Ferry units are equivalent to core and assembly conditions of other plants for which the methodology was benchmarked, the AREVA methodology (including uncertainties) remains applicable for EPU conditions at the Browns Ferry Units. The introduction of ATRIUM 10XM fuel does not change the conclusions in ANP-2860P Rev 2 since the methodologies fully account for all of the fuel design specific characteristics.

More specifically:

a) The steady state and transient neutronics and thermal-hydraulic analytical methods and code systems supporting EPU at Browns Ferry are within NRC approved applicability ranges because the conditions for EPU application at Browns Ferry are equivalent to existing core and assembly conditions in other plants for which the AREVA methodology was benchmarked.

b) The calculational and measurement uncertainties applied in Browns Ferry applications are valid at EPU because the conditions for Browns Ferry application are bounded by existing core and assembly conditions for which the AREVA methodology was benchmarked.

c) The assessment database and uncertainty of models used to simulate the plant response at EPU conditions for Browns Ferry are bounded by core and assembly conditions for which the AREVA methodology was benchmarked.

Section 5, Appendix A, and Appendix B summarize methodology or application enhancements specifically for Browns Ferry:

a) **[ ]**

b) LOCA radiation view factors and **[ ]**

c) Thermal conductivity degradation

d) Void quality correlation biases

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17. ANP-10300P, Revision 0, *AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios*, (ML100040158) December 2009.

18. EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model*, Framatome ANP, May 2001.

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Appendix A Fuel Conductivity Degradation

A.1 *Introduction*

The Nuclear Regulatory Commission, U. S. (NRC) issued Information Notice (IN) 2009-23 (No. ML091550527), dated October 8, 2009, for concerns regarding the use of historical fuel thermal conductivity models in the safety analysis of operating reactor plants. IN 2009-23 discusses how historical fuel thermal mechanical codes may overpredict fuel rod thermal conductivity at higher burn-ups based on new experimental data. This new experimental data showed significant degradation of fuel pellet thermal conductivity with exposure. The NRC staff concluded that the use of the older legacy fuel models will result in predicted fuel pellet conductivities that are higher than the expected values.

This appendix summarizes the impact and treatment of fuel conductivity degradation for licensing safety analyses supporting operation at Browns Ferry.

A.2 *Disposition of Licensing Safety Analysis for Browns Ferry ATRIUM 10XM Fuel*

RODEX2 and RODEX2A codes were approved by the NRC in the early and mid-1980's, respectively. At that time, thermal conductivity degradation (TCD) with exposure was not well characterized by irradiation tests or post-irradiation specific-effects tests at high burnups. The fuel codes developed at that time did not accurately account for this phenomenon. Analyses performed with RODEX2/2A are impacted by the lack of an accurate thermal conductivity degradation model. Likewise, conductivity models in the transient codes COTRANSA2 and XCOBRA-T do not account for thermal conductivity degradation.

RODEX4 (Reference 6) is a best-estimate, state-of-the-art fuel code that fully accounts for burnup degradation of fuel thermal conductivity. RODEX4, therefore, can be used to quantify the impact of burnup-dependent fuel thermal conductivity degradation and its effect on key analysis parameters.

Thermal-mechanical licensing safety analyses for Browns Ferry are performed with RODEX4 and therefore explicitly account for thermal conductivity degradation. No additional assessment is needed for those analyses. For thermal-hydraulic and safety analyses an evaluation is needed. The following analysis methodologies use RODEX2 and/or include a separate UO_2 thermal conductivity correlation:

- Anticipated Operational Occurrence (AOO) analysis based on COTRANSA2/RODEX2/XCOBRA-T codes;
- Loss of Coolant Accidents (LOCA) analyses based on RELAX/RODEX2/HUXY codes;
- Overpressurization analyses based on COTRANSA2/RODEX2 codes;
- Stability analyses based on STAIF/RAMONA5-FA codes.

A.3 *Assessment of Analyses for Browns Ferry Operations*

The issues identified in IN 2009-23 were entered into the AREVA corrective action program in 2009. A summary of the investigation was provided to the NRC in a white paper (Reference 13). The white paper presented results of an extensive evaluation; for BWRs the assessments consisted primarily of ATRIUM-10 fuel.

The NRC reviewed Reference 13 and provided requests for information in Reference 14. AREVA provided responses in Reference 15. Items relevant from References 14 and 15 are also discussed in the following subsections.

An assessment of the impact of thermal conductivity degradation on fuel licensing analyses was previously provided in Reference 16 for the introduction of ATRIUM 10XM fuel at Browns Ferry for current licensed thermal power conditions.

A.3.1 Anticipated Operational Occurrence Analyses

The computer codes COTRANSA2 and XCOBRA-T are used in AOO analyses. Both codes use UO_2 thermal conductivity correlations that do not address TCD. In addition, the core average gap conductance used in the COTRANSA2 system calculations and the hot channel gap conductance used in XCOBRA-T hot channel calculations are obtained from RODEX2 calculations. In general, the sensitivity to conductivity and gap conductance for AOO analyses is in the opposite direction for the core and hot channel, i.e., putting more energy into the coolant (higher thermal conductivity/higher gap conductance) is non-conservative for the system calculation but conservative for the hot channel calculation. The competing effects between the core and hot channel calculation minimize the overall impact of thermal conductivity degradation.

The assessment of Reference 13 demonstrated that COTRANSA2 uses several conservative assumptions, which results in conservatism relative to the Peach Bottom turbine trip qualification database. The COTRANSA2 methodology results in predicted integral power increases that are bounding relative to the Peach Bottom benchmark tests. With the 110% integral power multiplier used in the methodology, the COTRANSA2 predicted to measured mean integral power is **[ ]** for the Peach Bottom turbine trip tests. The COTRANSA2 benchmark testing was performed using the same UO_2 conductivity model as used in the current licensing analyses. Therefore, the benchmarking comparisons inherently include any impact of UO_2 conductivity degradation with exposure exhibited in the Peach Bottom tests.

The prior assessment was based on fuel designs current at the time of the Peach Bottom tests. To supplement the assessment with modern fuel, calculations were performed using the as-submitted AURORA-B code (Reference 17). AURORA-B is built from previous NRC approved methods. These methods include codes RODEX4, MICROBURN-B2, and S-RELAP5; UO_2 thermal conductivity degradation is correctly modeled. It is noted that the AURORA-B methodology and application have not yet been reviewed by the NRC; however, the staff accepted its use for sensitivity calculations for this assessment (Reference 14). The AURORA-B sensitivity studies show that the impact of fuel thermal conductivity degradation with exposure results in a decrease in the ΔCPR of **[ ]** increase in the transient LHGR excursion.

Based on the inherent conservatisms associated with the transient analysis codes and the small impact of thermal conductivity degradation with exposure for the AOO analysis, it is concluded that MCPR and LHGR operating limits based on the AOO methodology are not impacted.

The LHGR limits of the fuel do not change for EPU operation. Since thermal conductivity degradation is primarily a function of LHGR, TCD at EPU conditions does not change the conservatisms nor invalidate the sensitivity discussed above. Therefore, the AOO methodology remains applicable at EPU conditions. It should be noted that transient LHGR analyses are performed with the RODEX4 code for Browns Ferry ATRIUM 10XM fuel, which correctly accounts for thermal conductivity degradation.

A.3.2 Loss of Coolant Accident Analyses

LOCA analyses are performed using the EXEM BWR-2000 methodology and include the use of the RODEX2, RELAX and HUXY computer codes. In addition to the initial stored energy, the RODEX2 code is used to calculate fuel mechanical parameters for use in the HUXY computer code that potentially impact the clad ballooning and rupture models. Clad ballooning has a small impact on Peak Cladding Temperature (PCT) and metal water reaction (MWR), but clad rupture can have a significant impact on PCT, depending on event timing.

The LOCA event is divided into two phases: the blowdown and refill/reflood phases. During the initial or blowdown portion of a LOCA, good cooling conditions exist, and the initial stored energy in the fuel is removed. While a decrease in the thermal conductivity increases the overall thermal resistance, heat transfer conditions remain sufficient to remove the initial stored energy. Numerous sensitivity studies have been performed to demonstrate that BWR LOCA analyses are insensitive to initial stored energy. After the initial phase of a LOCA, the heat transfer coefficient at the cladding surface is degraded due to the loss of coolant (low flow and high quality). As a result, the heat transfer from the fuel is primarily controlled by the surface heat flux, and the temperature profile across the pellet is very flat. When compared to the rod surface thermal resistance, the pellet thermal conductivity is not a significant portion of the fuel rod total thermal resistance. Therefore, LOCA calculations are not sensitive to the UO_2 thermal conductivity used in RELAX and HUXY.

To demonstrate that limiting LOCA calculations are not sensitive to UO_2 thermal conductivity, assessments were performed for multiple BWRs. **[**

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Assessments of the potential impact of exposure-dependent degradation of UO_2 thermal conductivity on the fuel mechanical parameters were made using the RODEX4 computer code.
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] The results of these evaluations were summarized to the NRC in References 13 and 15.

The impact of TCD was incorporated in the Browns Ferry ATRIUM 10XM and ATRIUM-10 HUXY analyses. [

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After the NRC approval of RODEX2, more Halden tests were performed with fuel centerline temperature monitoring. As with the RODEX4 submittal, [

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The ATRIUM 10XM and ATRIUM-10 PCT results with the impact of TCD will be presented in the MAPLHGR reports that will be included in the Browns Ferry Licensing Amendment Request for operation at EPU. Each cycle the MAPLHGR limit will be analyzed for any new neutronic lattice designs. The impact of TCD will be analyzed if warranted by the exposure dependent PCT results for the new lattice.

A.3.2.1 **Responses to NRC Requests**

From the NRC's review of Reference 13, additional information was requested in Reference 14. The information requests and responses are provided as follows:

A detailed explanation of the source of the heat transfer coefficients utilized in the HUXY calculation

This request is answered in Reference 15 and this answer is applicable to Browns Ferry.

A description of how LOCA analyses are initialized in terms of power distribution; specifically, how thermal limits (such as MLHGR or OLMCPR) are considered in the initialization

This request is answered in Reference 15. **[**

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A characterization of the PCT sensitivity to fuel conductivity for plants where early boiling transition is predicted to occur during the early stages of LOCA

The limiting two-loop break in the LOCA break spectrum analyses for ATRIUM 10XM fuel at EPU (a small break) does not show early boiling transition. A review of the entire EPU break spectrum identified that large break two-loop and single-loop cases have boiling transition occurring at **[**

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A.3.3 Overpressurization Analyses

The COTRANSA2 code is used to perform analyses to demonstrate that the reactor vessel pressure will not exceed the ASME vessel pressure limit during specified events. COTRANSA2 is also used to demonstrate that the vessel pressure does not exceed the overpressure acceptance criterion for an anticipated transient without scram (ATWS) event.

Analyses using COTRANSA2 are potentially affected by UO_2 thermal conductivity degradation with exposure, as described in Section A.3.1 for AOO analyses. As discussed in Reference 13, the impact on overpressurization analysis was assessed in two ways: using AURORA-B to assess the relative impact of using UO_2 thermal conductivity degradation with exposure; and decreasing the core average thermal conductivity input into COTRANSA2 to account for the effects of exposure. Reference 13 summarized the increase in pressure as less than a **[ ]** pressure rise (peak pressure – initial pressure) for the AURORA-B assessment and a pressure rise of **[ ]** for COTRANSA2 when the core average thermal conductivity assumed a 30% reduction. The Reference 13 evaluations concluded that the impact of UO_2 thermal conductivity degradation with exposure on the peak vessel pressure in overpressurization analyses was a small increase, the increase is less than the existing margins to the acceptance criteria.

The impact of TCD will be accounted for in ASME and ATWS overpressurization analyses performed for Browns Ferry by reducing the core average thermal conductivity in COTRANSA2 to account for the effects of exposure. The reduction will be calculated based on the exposure of the fuel in the core.

A.3.3.1 **Responses to NRC Requests**

From the NRC's review of Reference 13, additional information was requested in Reference 14. The requests and responses to the requests are provided as follows:

> *A comprehensive list of the identified nonconservative biases in the AREVA overpressure analysis methods*
>
> The comprehensive list of items was provided in Reference 15. The biases applicable for Browns Ferry are summarized as follows. These biases are addressed for each cycle to ensure that the pressure limits are not exceeded.

Void-Quality Correlation: The bias is **[ ]** for ASME and **[ ]** for ATWS calculations.

Thermal Conductivity Degradation: In Reference 13 AREVA evaluated the impact of TCD for ATRIUM-10 fuel in two ways: using the AURORA-B code (Reference 17) to assess the relative impact of using UO_2 thermal conductivity with exposure degradation; and decreasing the core average thermal conductivity input into COTRANSA2 to account for the effects of exposure degradation. It was noted that changing the UO_2 thermal conductivity model provides a conservative estimate of the impact of exposure degradation on calculated peak vessel pressure. The limiting results obtained for the plants assessed in support of Reference 13 were reported as follows. For ASME, the increase in peak reactor pressure is expected to be less than **[ ]** of the pressure rise (peak pressure – initial pressure). For ATWS, the increase in pressure rise was **[ ]**.

Doppler Model Mismatch Between MICROBURN-B2 and COTRANSA2: The bias is **[ ]** of the calculated pressure rise from steady-state conditions for the ASME calculation and **[ ]** for the ATWS calculation.

Verification that the nonconservative biases are considered in an integral sense in the safety analyses.

Reference 15 demonstrated that it is conservative to add the biases together from separate effect assessments. The integral study demonstrated a decrease in total bias pressure.

These three non-conservative biases remain applicable at EPU conditions and applicable to ATRIUM 10XM fuel. The peak pressures of the ASME and ATWS overpressurization events presented in the Reload Licensing report include these penalties. The Reload Licensing report will be included in the Browns Ferry Licensing Amendment Request for operation at EPU.

A.3.4 Stability Analyses

As summarized in Reference 13, the computer codes STAIF and RAMONA5-FA are used in stability analyses. Both of these codes have fuel models that include UO_2 thermal conductivity degradation with exposure. Therefore, there is no impact on AREVA stability analyses.

A.3.5 Fire Event Analyses

The analyses to demonstrate compliance with Appendix R criteria and the newer NFPA 805 criteria are performed using the LOCA analysis codes. For these analyses, the calculated PCT is much lower than for LOCA analyses. As detailed in Section A.3.2 for the LOCA analyses, the impact of UO_2 thermal conductivity degradation with exposure has only a small impact on calculated PCT. Like the Browns Ferry LOCA analyses, the fire protection analyses are limiting at beginning of life. Therefore, the conclusions from these analyses would not be affected by UO_2 thermal conductivity degradation with exposure.

Appendix B LOCA Modifications

B.1 ***LOCA Analysis***

The AREVA LOCA methodology applied at Browns Ferry differs from the approved methodology in three aspects:

B.1.1 Radiation View Factors

In the Safety Evaluation for Reference 18 the NRC approved the AREVA EXEM BWR-2000 ECCS evaluation model. The HUXY code (Reference 19) is the part of this model that performs the heatup calculations and provides PCT and local clad oxidation at the axial plane of interest. The code evaluates the radiation heat transfer between the fuel rod of interest and other fuels rods, the internal water canisters, and the fuel channel. AREVA has implemented an automated approach for calculating radiation view factors within the HUXY computer program.

The original approach was based on the method of cross-strings as described in Section 2.3 of Reference 19. This resulted in the derivation and programming of analytical expressions as a function of fuel rod diameters for the radiation view factors between each fuel rod and its predominant neighbors. The view factors were then internally computed throughout the HUXY heatup analyses based on these analytical expressions and the time dependent evolution of the fuel rod dimensions.

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B.1.3 Thermal Conductivity Degradation

The EXEM BWR-2000 ECCS evaluation model uses the RODEX2 fuel rod models and therefore, underpredicts the impact of thermal conductivity degradation with exposure. The evaluation of thermal conductivity degradation and impact on PCT for Browns Ferry are presented in A.3.2.

Figure B-1 [ ]