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

April 22, 2016

10 CFR 50.4
10 CFR 50.46

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **10 CFR 50.46 Annual Report for Browns Ferry Nuclear Plant, Units 1 and 2, and 10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Unit 3**

Reference:

1. Letter from TVA to NRC, "10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3," dated April 30, 2015 (ML15120A295)
2. Letter from TVA to NRC, "10 CFR 50.46 30-Day Report for Browns Ferry Nuclear Plant, Unit 2," dated December 22, 2015 (ML15356A654)

The purpose of this letter is to provide the Annual Report, as required by Title 10 of the Code of Federal Regulations (10 CFR) 50.46, of changes or errors discovered in the Emergency Core Cooling System (ECCS) evaluation model for Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3. In accordance with 10 CFR 50.46, "Acceptance Criteria for ECCS for Light-Water Nuclear Power Reactors," paragraph (a)(3)(ii), Enclosures 1, 2, and 3 describe the nature and the estimated effect on the limiting ECCS analysis, of changes or errors discovered since submittal of Reference 1 for BFN, Units 1 and 3, and Reference 2 for BFN, Unit 2.

BFN, Units 1 and 2, have no new changes or errors since the issuance of References 1 and 2, respectively.

Enclosure 1 to this letter contains a summary of changes to the calculated Peak Cladding Temperature (PCT) made to the BFN, Unit 1, ECCS-Loss of Coolant Accident (LOCA) analysis of record (AOR). The baseline GE14 fuel PCT for BFN Unit 1, is 1760 degrees Fahrenheit (°F). The baseline ATRIUM™-10 fuel PCT for BFN, Unit 1 is 1944°F.

Enclosure 2 to this letter contains a summary of changes to the calculated PCT made to the BFN, Unit 2, ECCS-LOCA AOR. The baseline ATRIUM™-10 fuel PCT for BFN, Unit 2 is 1944°F. The baseline ATRIUM™-10XM fuel PCT for BFN, Unit 2 is 1906°F. The baseline ATRIUM™-11 lead use fuel assembly PCT for BFN, Unit 2 is 1876°F.

Enclosure 3 to this letter contains a summary of changes to the calculated PCT made to the BFN, Unit 3, ECCS-LOCA AOR. The baseline ATRIUM™-10 fuel PCT for BFN, Unit 3, is 1944°F. BFN, Unit 3, loaded the first reload batch of ATRIUM™-10XM fuel during the Spring 2016 refueling outage. This report establishes a baseline ATRIUM™-10XM fuel PCT of 1906°F for BFN, Unit 3.

Enclosure 3 also serves as the 30-day report of significant changes to the BFN, Unit 3, ECCS-LOCA AOR for ATRIUM™-10XM fuel. The PCT changes and errors identified for BFN, Unit 3, described in the enclosed report, when expressed as cumulative sums of the absolute magnitudes, exceed 50 °F. In accordance with 10 CFR 50.46(a)(3)(ii), a holder of an operating license or construction permit is required to report changes and errors affecting an ECCS evaluation model to the NRC within 30 days when the cumulative sum of the absolute magnitudes of resulting PCT changes exceeds 50°F. The licensee is also required to include with the report, a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with the 10 CFR 50.46 requirements. The affected LOCA analysis, for ATRIUM™-10XM fuel assemblies, became effective upon startup from the Spring 2016 refueling outage on March 27, 2016; therefore, the 30-day report is due on April 26, 2016.

As presented in this report, compliance with 10 CFR 50.46 requirements is demonstrated by the calculated PCT for all three BFN units remaining below the 2200°F limit. Therefore, the Tennessee Valley Authority has concluded that no proposed schedule for providing a reanalysis or other action is required.

There are no new regulatory commitments in this letter. Please direct questions concerning this issue to Jamie L. Paul at (256) 729-2636.

Respectfully,

S. M. Bono
Site Vice President

Enclosures:

1. 10 CFR 50.46 Annual Report for Browns Ferry Nuclear Plant, Unit 1
2. 10 CFR 50.46 Annual Report for Browns Ferry Nuclear Plant, Unit 2
3. 10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Unit 3

cc (w/Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Browns Ferry Nuclear Plant
NRC Project Manager - Browns Ferry Nuclear Plant

ENCLOSURE 1

10 CFR 50.46 ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 1

The Browns Ferry Nuclear Plant (BFN), Unit 1, core contains both the ATRIUM™-10 and GE14 fuel designs.

ATRIUM™-10 Fuel Evaluation

The previous 10 CFR 50.46 report (Reference 1) for BFN, Unit 1, was submitted on April 30, 2015. This report cites References 2 and 3 as the analysis of record (AOR) for ATRIUM™-10 fuel, with a baseline Peak Cladding Temperature (PCT) for ATRIUM™-10 fuel of 1944 degrees Fahrenheit (°F).

No new changes or errors have been discovered in the ATRIUM™-10 loss of coolant accident (LOCA) analyses since the issuance of Reference 1.

Table 1 details the accumulated PCT impact due to errors and changes in the LOCA analyses since the AOR in Reference 3 of this enclosure.

Table 1: Cumulative Effect of PCT Changes - BFN, Unit 1 (ATRIUM™-10)	
Baseline PCT (Reference 3)	1944°F
Thermal Conductivity Degradation (previously reported)	+ 0°F
Recirculation pump characteristics (flow, pump speed, and pump power) (previously reported)	+ 9°F
Rated recirculation pump torque (previously reported)	+ 28°F
Elevation of bottom vessel flange (previously reported)	- 5°F
In-core detector housing diameter (previously reported)	+ 17°F
Jet pump riser inlet thermal sleeve diameter (previously reported)	+ 4°F
Rams head diameter (previously reported)	+ 1°F
Dryer water seal skirt diameter (previously reported)	- 6°F
Core plate girder support width (previously reported)	+ 1°F
Main steam line diameter (previously reported)	+ 2°F
Accumulated changes since baseline analysis	+ 51°F
New licensing PCT	**1995°F**
Absolute value of accumulated changes	73°F

ENCLOSURE 1

10 CFR 50.46 ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 1

GE14 Fuel Evaluation

The previous 10 CFR 50.46 report (Reference 1) for BFN, Unit 1, was submitted on April 30, 2015. This report cites Reference 4 as the AOR for GE14 fuel. The applicability of this analysis to the current plant configuration was confirmed by GE-Hitachi Nuclear Energy (GEH) in Reference 5. Reference 4 provides PCT results for both Current Licensed Thermal Power (CLTP) and extended power uprate (EPU) conditions. The Tennessee Valley Authority has elected to use the CLTP results for 10 CFR 50.46 reporting, because EPU has not been approved for BFN, Unit 1, and all GE14 fuel is scheduled to be discharged prior to the planned EPU implementation date. The baseline PCT for GE14 fuel at CLTP conditions is 1760°F.

No new changes or errors have been discovered in the GE14 LOCA analysis since the issuance of Reference 1.

Table 2 details the accumulated PCT impact due to errors and changes in the GE14 LOCA analyses since the AOR in Reference 4 of this enclosure.

Table 2: Cumulative Effect of PCT Changes - BFN, Unit 1 (GE14)	
Baseline PCT	1760°F
Input coefficient database error (previously reported)	+25°F
Revised gamma heat deposition formulation (previously reported)	+15°F
Pellet thermal conductivity degradation (previously reported)	+0°F
SAFER04A Maintenance Update Changes (previously reported)	+ 0 °F
SAFER04A E4-Mass Non-conservatism (previously reported)	+ 10 °F
SAFER04A E4 Minimum Core DP model (previously reported)	+ 20 °F
SAFER04A Bundle/Lower Plenum CCFL Head (previously reported)	- 20 °F
Accumulated changes since baseline analysis	+50°F
New licensing PCT	**1810°F**
Absolute value of accumulated changes	90°F

ENCLOSURE 1

10 CFR 50.46 ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 1

References

1. Letter from TVA to NRC, "10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3," dated April 30, 2015. (ML15120A295)

2. AREVA NP Inc., "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis," ANP-3015(P) Revision 0, September 2011.

3. AREVA NP Inc., "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM^{TM}-10 Fuel," ANP-3016(P) Revision 1, November 2013.

4. GE Nuclear Energy, "Browns Ferry Nuclear Plant Units 1, 2, and 3: SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-32484P Revision 6, February 2005.

5. GE-Hitachi Nuclear Energy, "Browns Ferry Nuclear Plant Unit 1: Supplementary Report Regarding ECCS-LOCA Evaluation Additional Single Failure Evaluation at Current Licensed Thermal Power," NEDC-32484P Rev. 6, Supplement 2, September 2012.

ENCLOSURE 2

10 CFR 50.46 ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 2

The Browns Ferry Nuclear Plant (BFN), Unit 2, core contains both the ATRIUM™-10 and ATRIUM™-10XM fuel designs, as well as ATRIUM™-11 lead-use assemblies.

ATRIUM™-10 Fuel Evaluation

The previous 10 CFR 50.46 report (Reference 1) for BFN, Unit 2, was submitted on December 22, 2015. This report cites References 2 and 3 as the analysis of record (AOR) for ATRIUM™-10 fuel, with a baseline Peak Cladding Temperature (PCT) of 1944 degrees Fahrenheit (°F).

No new changes or errors have been discovered in the ATRIUM™-10 loss of coolant accident (LOCA) analyses since the issuance of Reference 1.

Table 1 details the accumulated PCT impact due to errors and changes in the LOCA analyses since the AOR in Reference 3 of this enclosure.

Table 1: Cumulative Effect of PCT Changes - BFN, Unit 2 (ATRIUM™-10)	
Baseline PCT (Reference 3)	1944°F
Thermal Conductivity Degradation (previously reported)	+ 0°F
Recirculation pump characteristics (flow, pump speed, and pump power) (previously reported)	+ 9°F
Rated recirculation pump torque (previously reported)	+ 28°F
Elevation of bottom vessel flange (previously reported)	- 5°F
In-core detector housing diameter (previously reported)	+ 17°F
Jet pump riser inlet thermal sleeve diameter (previously reported)	+ 4°F
Rams head diameter (previously reported)	+ 1°F
Dryer water seal skirt diameter (previously reported)	- 6°F
Core plate girder support width (previously reported)	+ 1°F
Main steam line diameter (previously reported)	+ 2°F
Accumulated changes since baseline analysis	+ 51°F
New licensing PCT	**1995°F**
Absolute value of accumulated changes	73°F

ENCLOSURE 2

10 CFR 50.46 ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 2

ATRIUM™-10XM Fuel Evaluation

The previous 10 CFR 50.46 report (Reference 1) for BFN, Unit 2, was submitted on December 22, 2015. This report cites References 4 and 5 as the AOR for ATRIUM™-10XM fuel, with a baseline PCT of 1906°F.

No new changes or errors have been discovered in the ATRIUM™-10XM LOCA analyses since the issuance of Reference 1.

Table 2 details the accumulated PCT impact due to errors and changes in the LOCA analyses since the AOR in Reference 5 of this enclosure.

Table 2: Cumulative Effect of PCT Changes - BFN, Unit 2 (ATRIUM™-10XM)	
Baseline PCT (Reference 5)	1906°F
Implementation of ACE correlation in RELAX (previously reported)	+ 0°F
Implementation of modified analysis approach (previously reported)	+ 0°F
Thermal Conductivity Degradation (previously reported)	+ 0°F
Recirculation pump characteristics (flow, pump speed, and pump power) (previously reported)	+ 9°F
Rated recirculation pump torque (previously reported)	+ 28°F
Elevation of bottom vessel flange (previously reported)	- 5°F
In-core detector housing diameter (previously reported)	+ 17°F
Jet pump riser inlet thermal sleeve diameter (previously reported)	+ 4°F
Rams head diameter (previously reported)	+ 1°F
Dryer water seal skirt diameter (previously reported)	- 6°F
Core plate girder support width (previously reported)	+ 1°F
Main steam line diameter (previously reported)	+ 2°F
Accumulated changes since baseline analysis	+ 51°F
New licensing PCT	**1957°F**
Absolute value of accumulated changes	73°F

ENCLOSURE 2

**10 CFR 50.46 ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 2**

ATRIUM™-11 Fuel Evaluation

The previous 10 CFR 50.46 report (Reference 1) for BFN, Unit 2, was submitted on December 22, 2015. This report cites Reference 6 as the AOR for ATRIUM™-11 lead use fuel assemblies, with a baseline PCT of 1876°F.

No new changes or errors affecting the ATRIUM™-11 LOCA analyses have been discovered since the issuance of Reference 1.

Table 3 details the accumulated PCT impact due to errors and changes in the LOCA analyses since the AOR in Reference 6 of this enclosure.

Table 3: Cumulative Effect of PCT Changes - BFN, Unit 2 (ATRIUM™-11)	
Baseline PCT (Reference 6)	1876°F
Implementation of ACE correlation in RELAX (previously reported)	+ 0°F
Implementation of modified analysis approach (previously reported)	+ 0°F
Thermal Conductivity Degradation (previously reported)	+ 0°F
Recirculation pump characteristics (flow, pump speed, and pump power) (previously reported)	+ 9°F
Rated recirculation pump torque (previously reported)	+ 28°F
Elevation of bottom vessel flange (previously reported)	- 5°F
In-core detector housing diameter (previously reported)	+ 17°F
Jet pump riser inlet thermal sleeve diameter (previously reported)	+ 4°F
Rams head diameter (previously reported)	+ 1°F
Dryer water seal skirt diameter (previously reported)	- 6°F
Core plate girder support width (previously reported)	+ 1°F
Main steam line diameter (previously reported)	+ 2°F
Accumulated changes since baseline analysis	+ 51°F
New licensing PCT	**1927°F**
Absolute value of accumulated changes	73°F

ENCLOSURE 2

**10 CFR 50.46 ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 2**

References

1. Letter from TVA to NRC, "10 CFR 50.46 30-Day Report for Browns Ferry Nuclear Plant, Unit 2," dated December 22, 2015. (ML15356A654)

2. AREVA NP Inc., "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis," ANP-3015(P) Revision 0, September 2011.

3. AREVA NP Inc., "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM™-10 Fuel," ANP-3016(P) Revision 1, November 2013.

4. AREVA NP Inc., "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM™ 10XM Fuel," ANP-3152(P) Revision 0, October 2012.

5. AREVA NP Inc., "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM™ 10XM Fuel," ANP-3153(P) Revision 1, December 2015.

6. AREVA NP Inc., "Browns Ferry Unit 2 Cycle 19 Reload Analysis," ANP-3354 Revision 0, November 2014.

ENCLOSURE 3

**10 CFR 50.46 30-DAY AND ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 3**

The Browns Ferry Nuclear Plant (BFN), Unit 3, core contains both the ATRIUM™-10 and ATRIUM™-10XM fuel designs.

ATRIUM™-10 Fuel Evaluation

The previous 10 CFR 50.46 report (Reference 1) for BFN, Unit 3, was submitted on April 30, 2015. This report cites References 2 and 3 as the analysis of record (AOR) for ATRIUM™-10 fuel, with a baseline Peak Cladding Temperature (PCT) for ATRIUM™-10 fuel of 1944 degrees Fahrenheit (°F).

No new changes or errors have been discovered in the ATRIUM™-10 loss of coolant accident (LOCA) analyses since the issuance of Reference 1.

Table 1 details the accumulated PCT impact due to errors and changes in the LOCA analyses since the AOR in Reference 3 of this enclosure.

Table 1: Cumulative Effect of PCT Changes - BFN, Unit 3 (ATRIUM™-10)	
Baseline PCT (Reference 3)	1944°F
Increased core spray leakage from lower sectional replacement hardware modification analysis (previously reported)	+ 34°F
Thermal Conductivity Degradation (previously reported)	+ 0°F
Recirculation pump characteristics (flow, pump speed, and pump power) (previously reported)	+ 9°F
Rated recirculation pump torque (previously reported)	+ 28°F
Elevation of bottom vessel flange (previously reported)	- 5°F
In-core detector housing diameter (previously reported)	+ 17°F
Jet pump riser inlet thermal sleeve diameter (previously reported)	+ 4°F
Rams head diameter (previously reported)	+ 1°F
Dryer water seal skirt diameter (previously reported)	- 6°F
Core plate girder support width (previously reported)	+ 1°F
Main steam line diameter (previously reported)	+ 2°F
Accumulated changes since baseline analysis	+ 85°F
New licensing PCT	**2029°F**
Absolute value of accumulated changes	107°F

10 CFR 50.46 30-DAY AND ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 3

ATRIUM™-10XM Fuel Evaluation

BFN, Unit 3, loaded an initial batch of ATRIUM™-10XM fuel in the Spring 2016 refueling outage. This report establishes References 4 and 5 as the AOR for ATRIUM™-10XM fuel, with a baseline PCT for ATRIUM™-10XM fuel of 1906°F.

Several changes and/or errors have been discovered in the ATRIUM™-10XM LOCA analyses since the issuance of the Reference 5 AOR.

The first error is in the implementation of the ACE correlation within the RELAX code. The ACE implementation within RELAX did not include an interpolation method to smooth the results at the edge of the correlation's range of applicability, as required by the licensing topical report for the AREVA LOCA methodology. This error has an estimated PCT impact of 0°F.

The second error is in the scripts used to extract specific event times from RELAX system calculations for the previously-approved modified analysis approach for BFN LOCA break spectrum calculations. AREVA determined that this error did not affect the limiting PCT results for ATRIUM™-10XM fuel at BFN, so the estimated PCT impact is 0°F.

The third change is based on a 10 CFR 21 communication issued by GE Hitachi (Reference 6). This communication described the potential for higher than expected leakage in the core spray line associated with the core spray lower sectional replacement hardware. Reference 6 indicates that the core spray flow delivered to the vessel could be non-conservative by 136 gallons per minute. In 2010, the Tennessee Valley Authority (TVA) determined that the issue only physically applies to BFN, Unit 3. AREVA performed calculations to evaluate the impact of this reduced core spray flow on the PCT value. The evaluation showed an increase of 28°F. A similar change has been reported for the ATRIUM™-10 LOCA analyses since 2010.

The fourth change is related to fuel thermal conductivity degradation. Reference 7 indicates that burnup degradation of fuel thermal conductivity over the approved burnup range was not supported by experimental data when older generation codes, like RODEX2 were approved. Hence, it is not explicitly modeled. In recent evaluations into this phenomenon, it appears that the use of the RODEX2 code (used to provide inputs to RELAX and HUXY) results in conservatively high temperatures at low burnup (less than 15 gigawatt days per metric ton uranium), but under-predicts pellet temperatures at higher exposures.

In Reference 7, AREVA provided a response to an NRC question on the effect of this non-conservatism on LOCA analysis. For BFN, the current method shows that the limiting PCT is at beginning of life. The effects of degrading thermal conductivity at higher burnups were not sufficient to result in a new limiting PCT at higher exposure. Therefore, there is no change in the reported PCT due to thermal conductivity degradation for BFN.

Since the issuance of Reference 4, TVA has completed a review of LOCA input parameters in preparation for new LOCA analyses to support extended power uprate (EPU). As part of the EPU work, two changes were identified related to previously-installed recirculation pump upgrades. Several additional input changes were made for minor corrections and refinements to the RELAX geometry inputs, based on a review of plant drawings and reference documents.

On April 3, 2015, AREVA notified TVA of the PCT impact of each change. AREVA revised the PCT notification on April 10, 2015.

- Recirculation pump characteristics (flow, pump speed, and pump power) - The performance curves for recirculation flow, pump capacity vs. pump speed, and pump power vs. pump speed were updated based on actual plant data from recent cycles, reflecting current plant configuration. Estimated PCT impact = +9°F.
- Rated recirculation pump torque - The input value was updated based on previously installed pump upgrades. Estimated PCT impact = +28°F.
- Elevation of bottom vessel flange - The input value for the elevation of the bottom of the vessel flange was decreased by 4.62 inches based on a review of plant drawings and reference documents. Estimated PCT impact = -5°F.
- In-core detector housing diameter - The input value for the detector housing inner diameter was increased by 0.11 inches based on a review of plant drawings and reference documents. Estimated PCT impact = +17°F.
- Jet pump riser inlet thermal sleeve diameter - The input value for the riser inlet thermal sleeve inner diameter was decreased by 0.152 inches based on a review of plant drawings and reference documents. Estimated PCT impact = +4°F.
- Jet pump rams head diameter - The input value for the rams head outer diameter was decreased by 0.19 inches based on a review of plant drawings and reference documents. Estimated PCT impact = +1°F.
- Dryer water seal skirt diameter - The input values for the dryer water seal skirt inner and outer diameters were adjusted by 0.50 and 0.75 inches, respectively, based on a review of plant drawings and reference documents. Estimated PCT impact = -6°F.
- Core plate girder support width - The input value for the width of the core plate girder support was reduced by 2 inches based on a review of plant input drawings and reference documents. Estimated PCT impact = +1°F.
- Main steam line diameter - The input value for the inner diameter of some sections of the main steam line was increased by 0.412 inches based on a review of plant input drawings and reference documents. Estimated PCT impact = +2°F.

Table 2: Cumulative Effect of PCT Changes - BFN, Unit 3 (ATRIUM™-10XM)	
Baseline PCT (Reference 5)	1906°F
Implementation of ACE correlation in RELAX	+ 0°F
Implementation of modified analysis approach	+ 0°F
Increased core spray leakage from lower sectional replacement hardware modification analysis	+ 28°F
Thermal Conductivity Degradation	+ 0°F
Recirculation pump characteristics (flow, pump speed, and pump power)	+ 9°F
Rated recirculation pump torque	+ 28°F
Elevation of bottom vessel flange	- 5°F
In-core detector housing diameter	+ 17°F
Jet pump riser inlet thermal sleeve diameter	+ 4°F
Rams head diameter	+ 1°F
Dryer water seal skirt diameter	- 6°F
Core plate girder support width	+ 1°F
Main steam line diameter	+ 2°F
Accumulated changes since baseline analysis	+ 79°F
New licensing PCT	**1985°F**
Absolute value of accumulated changes	101°F

References

1. Letter from TVA to NRC, “10 CFR 50.46 Annual Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3,” dated April 30, 2015. (ML15120A295)

2. AREVA NP Inc., “Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis,” ANP-3015(P) Revision 0, September 2011.

3. AREVA NP Inc., “Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM™-10 Fuel,” ANP-3016(P) Revision 1, November 2013.

4. AREVA NP Inc., “Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM™ 10XM Fuel,” ANP-3152(P) Revision 0, October 2012.

5. AREVA NP Inc., "Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM™ 10XM Fuel," ANP-3153(P) Revision 1, December 2015.

6. GE-Hitachi Nuclear Energy, "Potential to Exceed Allowable Core Spray Leakage," 10 CFR 21 Communication SC 10-05, dated March 15, 2010.

7. AREVA Letter to NRC, P. Salas (AREVA) to USNRC Document Control Desk, "Response to NRC Letter Regarding Nuclear Fuel Thermal Conductivity Degradation Evaluation for Light Water Reactors Using AREVA Codes and Methods," NRC 12:023, dated April 27, 2012. (ML121220377)