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

April 30, 2015

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 30-Day and Annual Report for Browns Ferry Nuclear Plant, Units 1, 2, and 3**

Reference:

1. Letter from TVA to NRC, "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," dated April 15, 2014 (ML14108A327)

2. Letter from TVA to NRC, "10 CFR 50.46 30-Day Report for Browns Ferry Nuclear Plant, Unit 1," dated June 20, 2014 (ML14175B390)

The purpose of this letter is to provide a 30-Day and 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 References 1 and 2 for BFN, Units 1, 2, and 3.

The Peak Cladding Temperature (PCT) changes and errors identified for BFN, Units 1, 2, and 3, described in the enclosed report, when expressed as cumulative sums of the absolute magnitudes, exceed 50 degrees Fahrenheit (°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. Notification of the PCT changes and errors was received from AREVA on April 3, 2015; therefore, the 30-Day Report is due by May 3, 2015.

Enclosure 1 to this letter contains a summary of changes to the calculated 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°F. The baseline ATRIUM™-10 fuel PCT for BFN, Unit 1, is 1944°F. Enclosure 1 also serves as the 30-day report of significant changes to the BFN, Unit 1, ECCS-LOCA AOR.

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. Enclosure 2 also serves as the 30-day report of significant changes to the BFN, Unit 2, ECCS-LOCA AOR. BFN, Unit 2, loaded the first reload batch of ATRIUM™-10XM fuel during the Spring 2015 refueling outage. This report establishes a baseline ATRIUM™-10XM fuel PCT of 1903°F for BFN, Unit 2. BFN, Unit 2, loaded eight lead-use ATRIUM™-11 fuel assemblies during the Spring 2015 refueling outage. This report establishes a baseline ATRIUM™-11 fuel PCT of 1876°F for BFN, Unit 2.

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. Enclosure 3 also serves as the 30-day report of significant changes to the BFN, Unit 3, ECCS-LOCA AOR.

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. Bono
FOR
K. Polson

K. J. Polson
Site Vice President

Enclosures:

1. 10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Unit 1
2. 10 CFR 50.46 30-Day and 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 30-DAY AND 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 June 20, 2014. 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°F.

Since the issuance of Reference 1, the Tennessee Valley Authority (TVA) has completed a review of Loss of Coolant Accident (LOCA) input parameters in preparation for new LOCA analyses to support BFN 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 also 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) - 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 degrees Fahrenheit (°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 inches 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.

## ENCLOSURE 1

## 10 CFR 50.46 30-DAY AND ANNUAL REPORT
## FOR
## BROWNS FERRY NUCLEAR PLANT, UNIT 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) | + 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 |
| Jet pump 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 | + 51°F |
| New licensing PCT | **1995°F** |
| Absolute value of accumulated changes | 73°F |

**ENCLOSURE 1**

**10 CFR 50.46 30-DAY AND 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 June 20, 2014. 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 EPU conditions. TVA 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 30-DAY AND ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 1**

## References


1. Letter from TVA to NRC, “10 CFR 50.46 30-Day Report for Browns Ferry Nuclear Plant, Unit 1,” dated June 20, 2014.

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. 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 30-DAY AND 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 April 15, 2014. 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°F.

Since the issuance of Reference 1, the Tennessee Valley Authority (TVA) has completed a review of Loss of Coolant Accident (LOCA) input parameters in preparation for new LOCA analyses to support BFN 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) - 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 degrees Fahrenheit (°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.

**ENCLOSURE 2**

**10 CFR 50.46 30-DAY AND ANNUAL REPORT**
**FOR**
**BROWNS FERRY NUCLEAR PLANT, UNIT 2**

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) | + 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 | + 51°F |
| New licensing PCT | **1995°F** |
| Absolute value of accumulated changes | 73°F |

**10 CFR 50.46 30-DAY AND ANNUAL REPORT**
**FOR**
**BROWNS FERRY NUCLEAR PLANT, UNIT 2**

## ATRIUM™-10XM Fuel Evaluation

BFN Unit 2 loaded an initial batch of ATRIUM™-10XM fuel in the Spring 2015 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 1903°F.

Several changes and/or errors have been discovered in the AREVA 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.

Since the issuance of Reference 5, TVA has completed a review of LOCA input parameters in preparation for new LOCA analyses to support 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) - 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 inches 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 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) | 1903°F |
| Implementation of ACE correlation in RELAX | + 0°F |
| Implementation of modified analysis approach | + 0°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 |
| Jet pump 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 | + 51°F |
| New licensing PCT | **1954°F** |
| Absolute value of accumulated changes | 73°F |

**ENCLOSURE 2**

**10 CFR 50.46 30-DAY AND ANNUAL REPORT
FOR
BROWNS FERRY NUCLEAR PLANT, UNIT 2**

**ATRIUM™-11 Fuel Evaluation**

BFN, Unit 2, loaded eight ATRIUM™-11 lead-use fuel assemblies in the Spring 2015 refueling outage. This report establishes Reference 6 as the AOR for ATRIUM™-11 fuel, with a baseline PCT for ATRIUM™-11 fuel of 1876°F.

The ATRIUM™-11 hot channel analyses documented in Reference 6 used the thermal-hydraulic boundary conditions from the full core ATRIUM-10XM blowdown, which is described in Reference 4. The limited number of lead-use assemblies cannot significantly impact the blowdown behavior of the core, so this assumption is appropriate for these assemblies.

For the ATRIUM™-11 lead-use fuel assemblies, the LOCA analysis is based on the ATRIUM™-10XM RELAX system analysis and input parameters. Therefore, the same changes and errors identified for the ATRIUM™-10XM analysis will also affect the ATRIUM™-11 PCT results reported in the Reference 6 AOR. These changes and errors are described below.

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.

Since the issuance of Reference 6, TVA has completed a review of LOCA input parameters in preparation for new LOCA analyses to support BFN EPU. As part of the EPU work, two changes were identified that were 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) - 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 inches 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 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 | + 0°F |
| Implementation of modified analysis approach | + 0°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 |
| Jet pump 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 | + 51°F |
| New licensing PCT | **1927°F** |
| Absolute value of accumulated changes | 73°F |

**ENCLOSURE 2**

**10 CFR 50.46 30-DAY AND ANNUAL REPORT**
**FOR**
**BROWNS FERRY NUCLEAR PLANT, UNIT 2**

## References


1. Letter from TVA to NRC, “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, Units 3,” dated April 15, 2014.

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 0, October 2012.

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 only the ATRIUM$^{TM}$-10 fuel design.

**Description of Changes and Errors Relative to the Previous Report**

The previous 10 CFR 50.46 report (Reference 1) for BFN, Unit 3, was submitted on April 15, 2014. This report cites References 2 and 3 as the analysis of record (AOR) for ATRIUM$^{TM}$-10 fuel, with a baseline Peak Cladding Temperature (PCT) for ATRIUM$^{TM}$-10 fuel of 1944°F.

Since the issuance of Reference 1, the Tennessee Valley Authority (TVA) has completed a review of Loss of Coolant Accident (LOCA) input parameters in preparation for new LOCA analyses to support BFN 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) - 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 degrees Fahrenheit (°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 inches 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.

**ENCLOSURE 3**

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

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** | |
|---|---|
| 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) | + 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 |
| Jet pump 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 | + 85°F |
| New licensing PCT | **2029°F** |
| Absolute value of accumulated changes | 107°F |

**ENCLOSURE 3**

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

**References**


1. Letter from TVA to NRC, "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, Units 3," dated April 15, 2014.

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.