TVA

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

May 30, 2012

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, Unit 3
Facility Operating License No DPR-68
NRC Docket No 50-296

Subject: **10 CFR 50.46 30-Day Report for Browns Ferry Nuclear Plant, Unit 3**

Reference: TVA Letter to NRC, "10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Units 2 and 3," dated April 18, 2012

The purpose of this letter is to provide a 10 CFR 50.46 30-day report of significant changes discovered in the Emergency Core Cooling System (ECCS) evaluation model for Browns Ferry Nuclear Plant (BFN), Unit 3. In accordance with 10 CFR 50.46, "Acceptance Criteria for ECCS for Light-Water Nuclear Power Reactors," paragraph (a)(3)(ii), the enclosure describes the nature and the estimated effect on the limiting ECCS analysis of changes or errors discovered since submittal of the April 18, 2012 letter for Browns Ferry Nuclear Plant, Unit 3.

As described in the April 18, 2012 letter, TVA reported a significant increase in Peak Cladding Temperature (PCT) due to ECCS analysis changes created to address issues with the applicability of the EXEM BWR-2000 LOCA methodology identified during the NRC acceptance review of the BFN, Unit 1, ATRIUM™-10 fuel transition License Amendment Request.

During the spring 2012 refueling outage, BFN completed modifications to the Automatic Depressurization System (ADS), which has restored the automatic initiation capability of the ADS in the event of a single failure. A new analysis has been created, which credits the automatic initiation capability of ADS and provides a more typical 10 CFR 50 Appendix K event progression. In accordance with the commitment in the April 18, 2012 letter, TVA will submit the revised Loss of Coolant Accident (LOCA) Analysis for Browns

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Ferry Nuclear Plant, Unit 3, to the NRC for review and approval by June 30, 2012.

There are no new regulatory commitments in this letter. Please direct questions concerning this issue to Tom Hess at (423) 751-3487.

Respectfully,

D. E. Cribbs

for J. W. Shea
Manager, Corporate Nuclear Licensing

Enclosure:

10 CFR 50.46 30-Day Report for Browns Ferry Nuclear Plant, Unit 3

cc (w/Enclosure):

NRC Regional Administrator – Region II

NRC Senior Resident Inspector – Browns Ferry Nuclear Plant

ENCLOSURE

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

The Browns Ferry Nuclear Plant (BFN) Unit 3 core currently contains only the ATRIUM™-10 fuel design.

Changes and Errors Reported Prior to Automatic Depressurization System Modification

The previous 10 CFR 50.46 report for BFN Unit 3 was submitted on April 18, 2012 (Reference 4). This report referenced ANP-2910(P) Revision 1 (Reference 2) as the baseline analysis, with a baseline Peak Cladding Temperature (PCT) of 1992 °F.

As part of the acceptance review of the BFN Unit 1 ATRIUM™-10 fuel transition License Amendment Request (LAR), the NRC questioned the applicability of the EXEM BWR-2000 LOCA methodology to analyze the unique BFN configuration that resulted in the limiting Loss of Coolant Accident (LOCA) event. The NRC position is that the limiting event in Reference 1 (i.e., a single failure resulting in failure of automatic initiation of the Automatic Depressurization System (ADS)) results in an event outside of the typical 10 CFR 50 Appendix K assumed event progression. Consequently the NRC concluded that the EXEM BWR-2000 methodology was being applied outside its Safety Evaluation Report basis for the as-analyzed limiting event in ANP-2908(P). Technical details regarding the NRC concerns and the modeling changes made by AREVA to address those concerns are available in the October 7, 2011, TVA response to "NRC Request for Additional Information" (Reference 3). These modeling changes were previously reported to the NRC in Reference 4.

The Reference 4 letter discussed the modified analysis methodology and thermal limit (Maximum Average Planar Heat Generation Rate (MAPLHGR) and Minimum Critical Power Ratio (MCPR)) penalties that were applied to ensure operability of BFN Unit 3. Reference 4 also detailed the PCT changes due to the revised methodology and the thermal limit penalties. A summary of those changes is provided below:

- Reference 9 describes additional Core Spray line leakage for Browns Ferry Unit 3. The leakage analysis indicated the core spray flow delivered inside the shroud is reduced by 136 gpm. The AREVA assessment of Browns Ferry break spectrum calculations assumes a 150 gpm reduction in core spray. For the manual operation of the ADS system configuration, Reference 4 reported that the impact of the 150 gpm additional leakage is 19 °F.

- AREVA has developed a new computational approach for calculating radiation view factors and implemented it within the HUXY computer code. The numerical computation, achieved through a ray-tracing algorithm, provides a straight forward accounting of the geometry to compute the view factor from each fuel rod to all other fuel rods, internal water structures, and the external fuel channel. Reference 4 reported that the impact of this method is 1 °F.

- Thermal limit (MAPLHGR and MCPR) penalties were assessed, as described in Reference 4, to ensure that a PCT less than 2200 °F can be attained. The previously reported PCT impact of this change is (-241 °F).

- Methodology changes were a result of NRC concern regarding a specific phenomena and how it is addressed with AREVA's EXEM BWR-2000 LOCA methodology. Section 4.4 of Reference 5 provides a detailed description of the calculation approach. The PCT impact due to the methodology changes was previously reported in Reference 4 (+322 °F). This PCT change includes the impact of the exposure dependent heatup evaluation of the limiting cycle-specific lattice PCT for all Browns Ferry cycles through the BFE3-15 design, which are typically reported in the MAPLHGR heatup report. Reference 8 determined this "adder" to be 12 °F (2073 °F – 2061 °F).

Table 1 – Previously Reported PCT Changes – BFN Unit 3

Baseline PCT (Reference 2)	1992 °F
Revised HUXY numerical view factor treatment analysis (reported in the June 3, 2011, 10 CFR 50.46 Annual Report)	+1 °F
Increased core spray leakage from lower sectional replacement hardware modification analysis (reported in the April 30, 2010, 10 CFR 50.46 Annual Report)	+19 °F
Modification to methodology to address NRC concerns regarding 10 CFR 50 Appendix K event progression (Reported in the April 18, 2012 10 CFR 50.46 30-Day and Annual Report)	+322 °F
Change in initial conditions for analysis (MAPLHGR reduction and MCPR increase) (Reported in the April 18, 2012 10 CFR 50.46 30-Day and Annual Report)	-241 °F
Accumulated changes since baseline analysis	+101 °F
Absolute value of accumulated changes	583 °F
Previously Reported PCT (Reference 4)	2093 °F

Description of Changes Relative to the Automatic Depressurization System Modification

A modification to the ADS has been completed during the spring 2012 refueling outage, which restored the automatic initiation capability of the ADS in the event of a single failure and returned BFN Unit 3 to a more typical 10 CFR 50 Appendix K event progression. A new LOCA analysis (References 5 and 6), which assumes the ADS modification is implemented, has been submitted to the NRC as part of the BFN Unit 1 ATRIUM™-10 fuel transition LAR. This new analysis, and the modified methodology described therein, has been approved by the NRC on a Unit 1 specific basis. Reference 4 contained a commitment to submit the revised LOCA analysis for application to BFN Unit 3 by June 30, 2012.

The revised LOCA analysis (References 5 and 6) provide a basis for the operability of BFN Unit 3 without the thermal limit penalties that were described in Reference 4. A description of PCT changes subsequent to those reported in Reference 4 was provided by AREVA in Reference 7. These PCT impacts are summarized below:

- Crediting the automatic ADS initiation results in a PCT reduction of 412 °F.

- The change to the ADS system allows the MAPLHGR and MCPR penalties reported in Reference 4 to be removed. The MAPLHGR will be increased to the values shown in Figure 2.1 of Reference 6, and the assumed MCPR will be decreased to the value presented in Table 4.1 of Reference 5. The PCT impact of removing these penalties is an increase of 241 °F (effectively removing the thermal limit change previously reported in Reference 4).

The change in ADS also changes the sensitivity of PCT for two previously reported changes:

- Reference 6 presents the exposure dependent MAPLHGR evaluation for cycle-specific Browns Ferry lattices. The evaluation includes all previous cycles for Units 2 and 3 including BFE3-16. This analysis also includes the computation change in determining numerical view factors previously discussed. The cycle specific lattice impact on PCT is +35 °F due to the limiting break characteristics determined in Reference 5. Therefore, compared to the impact described in Reference 8, the net change in cycle specific lattice PCT is +23 °F (35 °F – 12°F).

- Reference 7 presents a PCT estimate for the additional core spray leakage at Unit 3 using the limiting break conditions presented in Reference 5. The impact is +34 °F. As discussed previously, the additional core spray leakage estimate in Reference 4 was reported to be +19 °F. Therefore, when compared to the Reference 4 impact, the net PCT change due to additional core spray leakage is +15 °F (34 °F – 19 °F).

Reference 10 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 (<15 GWd/MTU), but underpredicts pellet temperatures at higher exposures.

In Reference 10, AREVA provided a response to an NRC question on the impact of this non conservatism on LOCA analysis. For Browns Ferry, the current method had shown that the limiting PCT is at beginning of life (BOL). 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 Browns Ferry.

Table 2 – PCT Changes Relative to the Previous Report – BFN Unit 3

Previously Reported PCT (Reference 4)	2093 °F
Automatic Initiation of ADS	-412 °F
Change in Initial Conditions (MAPLHGR Increase and MCPR decrease)	+241 °F
Net change to Limiting Browns Ferry Cycle Specific Lattice (Through BFE3-16)	+23 °F
Net change to additional Unit 3 Core Spray Leakage	+15 °F
Thermal Conductivity Degradation	+ 0 °F
Accumulated changes since previous report	-133 °F
Absolute value of accumulated changes since previous report	691 °F
Current PCT	1960 °F

References

1. ANP-2908(P) Revision 0, "Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Break Spectrum Analysis," AREVA NP Inc., March 2010.

2. ANP-2910(P) Revision 1, "Browns Ferry Units 1, 2, and 3 105% OLTP LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM™-10 Fuel," AREVA NP Inc., November 2010.

3. TVA Letter to NRC, "Response to NRC Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel (TAC NO. ME3775)," October 7, 2011.

4. TVA Letter to NRC, "10 CFR 50.46 30-Day and Annual Report for Browns Ferry Nuclear Plant, Units 2 and 3," April 18, 2012.

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

6. ANP-3016(P) Revision 0, "Browns Ferry Units 1, 2, and 3 LOCA-ECCS MAPLHGR Limit for ATRIUM™-10 Fuel, Areva NP Inc.," dated December 2011.

7. FAB12-2176, "Transmittal of TVA Requested 10 CFR 50.46 PCT Error Reporting Supporting Operation of Browns Ferry Unit 3 Cycle 16," AREVA NP Inc., May 10, 2012.

8. FAB11-2353, "Transmittal of BFE2-17 and BFE3-15 Operability Assessment Revision 3," AREVA NP Inc., July 13, 2011.

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

10. 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," NRC12:023, April 27, 2012 (ML121220377).