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February 23, 2011

10 CFR 50.90

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

Browns Ferry Nuclear Plant, Unit 1
Facility Operating License No. DPR-33
NRC Docket No. 50-259

Subject: **Response to NRC Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel (TAC No. ME3775)**

References:

1. Letter from TVA to NRC, "Technical Specification Change TS-473, AREVA Fuel Transition," dated April 16, 2010
2. NRC Letter to TVA, "Browns Ferry Nuclear Plant, Unit 1 - Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel (TAC No. ME3775)," dated January 24, 2011

On April 16, 2010, the Tennessee Valley Authority (TVA) submitted "Technical Specification Change TS-473, AREVA Fuel Transition," (Reference 1) to the NRC requesting approval of a license amendment to support using AREVA Fuel in Unit 1 at Browns Ferry Nuclear Plant. On January 24, 2011, TVA received a Request for Additional Information (RAI) letter from the NRC (Reference 2) containing 16 questions related to Technical Specifications Change TS-473. The NRC requested the responses within 30 days, i.e., no later than February 23, 2011.

Enclosure 1 to this letter provides the TVA responses to the 16 NRC RAI questions. Incorporated by reference to Enclosure 2 are responses from AREVA NP to 14 of the questions.

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Enclosure 2 to this letter contains information that AREVA NP considers to be proprietary in nature and subsequently, pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4), it is requested that such information be withheld from public disclosure.

Enclosure 3 contains the redacted version of Enclosure 2 with the proprietary material removed, suitable for public disclosure.

Enclosure 4 provides the affidavit, supporting this request.

This letter does not include any new regulatory commitments. Please direct any questions concerning this matter to Tom Matthews at (423) 751-2687.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on the 23rd day of February, 2011.

Respectfully,

R. M. Krich

Enclosures:

1. TVA Response to Request for Additional Information
2. Browns Ferry Unit 1 AREVA Fuel Transition Input to TVA for RAIs, *Proprietary*
3. Browns Ferry Unit 1 AREVA Fuel Transition Input to TVA for RAIs, *Non-Proprietary*
4. Affidavit

cc (Enclosures):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Browns Ferry Nuclear Plant
Alabama State Department of Public Health

ENCLOSURE 1

Tennessee Valley Authority

Browns Ferry Nuclear Plant, Unit 1

Technical Specifications Change TS-473 - AREVA Fuel Transition

TVA Response to Request for Additional Information

ENCLOSURE 1

Tennessee Valley Authority

Browns Ferry Nuclear Plant, Unit 1

Technical Specifications Change TS-473 - AREVA Fuel Transition

TVA Response to Request for Additional Information

NRC Question 1

ANP-2877P, Section 1.0:

Explain what "chamfered pellet design" is and describe how this design reduces the occurrence of pellet chipping during manufacturing, and then reducing the pellet-clad-interaction failure due to missing pellet surface.

TVA Response

AREVA NP has provided this response on page 1 of Enclosure 2.

NRC Question 2

ANP-2877P, Section 2.1.4:

Explain the "Harmonized Advanced Load Chain" modifications that improved the upper tie plate (UTP) connection by making it simpler and more robust.

TVA Response

AREVA NP has provided this response on page 3 of Enclosure 2.

NRC Question 3

ANP-2877P, Section 2.1.5, ANP-2859(P) Appendix B:

(a) *Provide details of the distribution of Gadolinia (UO_2+Gd_2O_3) rods in the BFN Unit 1 core for the upcoming cycle, with respect to the number of Gadolinia rods and respective Gadolium (Gd) enrichment.*

(b) *With degraded thermal conductivity, and lower melting point of the UO_2+Gd_2O_3 mixture, describe what adjustments are made in the Gadolinia rods to prevent failure of the Gadolinia rod melting. Is there any restriction on linear heat generation rate (LHGR) limit for the Gadolinia rods during normal operation and anticipated operational occurrences (ADOs)?*

(c) *Section 3.2.4 of ANP-2877 indicates that "for ADOs, the fuel temperatures are calculated using the same power history (as for normal operating temperatures), except that additional calculations are performed at elevated power levels as a function of exposure corresponding with the Protection Against Power Transients (PAPT) LHGR limit." Describe this process with example calculations.*

(d) *Describe the adjustments, and how the adjustments are applied to the fuel melt temperature, for exposure and Gadolinia content, as stated in Section 3.2.4 of ANP-2877. Show a typical calculation.*

TVA Response

(a) AREVA NP has provided this response on page 6 of Enclosure 2.

(b) AREVA NP has provided this response on page 11 of Enclosure 2.

(c) AREVA NP has provided this response on page 12 of Enclosure 2.

(d) AREVA NP has provided this response on page 14 of Enclosure 2.

NRC Question 4

ANP-2877P, Section 3.2:

Section 3.2, under bullet "Cladding Collapse," states that "The pellet/clad gap is evaluated [up to a proprietary rod exposure] to ensure the cladding does not [proprietary end state].

Section 3.2.2 "Cladding Collapse" states that gap conditions are evaluated after the first [proprietary rod exposure stating a different end state than Section 3.2 above].

After reviewing the proprietary information contained in ANP-2877, Section 3.2, please explain why there is discrepancy between the above two statements and correct the error, if needed.

TVA Response

AREVA NP has provided this response on page 15 of Enclosure 2.

NRC Question 5

ANP-2877P, Section 3.2.6:

Section 3.2.6 of ANP-2877 states that "the evaluation (for cladding rupture) is covered separate from this report." Identify the location of this report if it is part of the license amendment request or is contained in other docketed material. Otherwise, please provide a copy of this report.

TVA Response

AREVA NP has provided this response on page 16 of Enclosure 2. The AREVA NP response refers to "EMF-2361(P)(A) Revision 0, 'EXEM BWR-2000 ECCS Evaluation Model,' Framatome ANP, May 2001." This document and its associated safety evaluation report are available in Agencywide Document Access and Management System (ADAMS) Accession Nos. ML003772936 and ML003772909, respectively.

NRC Question 6

ANP-2877P, Sections 3.2.8, and 3.3.7:

Discuss the impact of Gd content in Gadolinia rods (UO_2+Gd_2O_3) on fuel densification, swelling and fission gas release in fuel rods.

TVA Response

AREVA NP has provided this response on page 16 of Enclosure 2.

NRC Question 7

ANP-2877P, Section 3.3.8:

Section 3.3.8 states that "Mixed core conditions for liftoff are considered on a specific basis as determined by the plant and other fuel types. Analyses to date indicate a large margin to liftoff under normal operating conditions." Justify this claim by providing a summary of the analysis and calculations.

TVA Response

AREVA NP has provided this response on page 20 of Enclosure 2.

NRC Question 8

ANP-2877P, Sections 3.4.1 through 3.4.3, Table 3.3 Item 3.4.2:

(a) [deleted]

(b) *Item 3.4.2 of Table 3.3 indicates that violent expulsion of fuel criteria for fuel is less than 280 calories per gram (cal/g) for coolability, and is less than 170 cal/g for rod failure. Standard Review Plan (SRP, NUREG-0800) Section 4.2, Appendix B, Section C (Core coolability criteria) stipulates that for fuel rod thermal-mechanical calculations, employed to demonstrate compliance with Criteria 1 (peak radial average fuel enthalpy must remain below 230 cal/g) and Criteria 2 (peal fuel temperature must remain below incipient fuel melting conditions), must be based upon design-specific information accounting for manufacturing tolerances and modeling using the NRC-approved methods, including burnup-enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature. Provide justification for the items in 3.4.2 of Table 3.3 in support of the Standard*

Review Plan acceptance criteria, specifically, the difference between the SRP value for coolability (less than 230 cal/g) and the value in Table 3.3, item 3.4.2.

TVA Response

(a) No response needed

(b) The values shown in ANP-2877P (violent expulsion of fuel of 280 cal/g; excessive fuel enthalpy of 170 cal/g) were used as the acceptance criteria in the reload analysis provided in document ANP-2863(P). These criteria were specifically approved for use via the safety evaluation report for ANF-89-98-(P)(A), Revision 1. Results shown in ANP-2863(P) were developed using the methodology described in XN-NF-80-19(P)(A). The use of ATRIUM-10 fuel was previously approved for use in Browns Ferry Nuclear Plant, Units 2 and 3, using the methodology described in XN-NF-80-19(P)(A).

NRC Question 9

ANP-2821 (P) Thermal-Hydraulic (T-H) Design Report, Section 3.1:

Provide a summary of detailed calculations for thermal-hydraulic characterization for the ATRIUM 10 reload fuel for BFN Unit 1. These calculations should show how the licensee obtained the loss coefficients and friction factors listed in Table 3.3 of ANP-2821; for example, the upper tie plate (UTP) loss coefficient, spacer loss coefficients, lower tie plate (LTP) grid loss coefficients, orifice, and LTP loss coefficients.

TVA Response

AREVA NP has provided this response on page 21 of Enclosure 2.

NRC Question 10

ANP-2821 (P), Section 3.2:

Provide detailed calculations that demonstrate thermal-hydraulic compatibility of ATRIUM 10 with the co-resident GE14 fuel in BFN Unit 1. These calculations should show the following:

(a) Calculations should demonstrate that during the entire transition from a full core GE14 to a full core ATRIUM 10 fuel, there will be no major impacts on thermal-hydraulic operation of BFN Unit 1 and should demonstrate compliance over the entire licensing range of the power/flow map.

(b) Justify your selection of bottom-peaked axial power distribution as a basis for the hydraulic compatibility results, compared to the results for top-and middlepeaked axial power distributions.

TVA Response

(a) AREVA NP has provided this response on page 23 of Enclosure 2.

(b) AREVA NP has provided this response on page 24 of Enclosure 2.

NRC Question 11

Thermal margin performance:

(a) Discuss the impact of part length rods and Gadolinia (UO_2+Gd_2O_3) on the application of SPCB critical power correlation.

(b) Will any of these part length rods undergo boiling transition/dryout during normal operating conditions or during transients and accident conditions?

TVA Response

(a) AREVA NP has provided this response on page 24 of Enclosure 2.

(b) AREVA NP has provided this response on page 25 of Enclosure 2.

NRC Question 12

Mixed core and Critical Power Ratio (CPR) calculations:

The proposed BFN Unit 1 core with AREVA Atrium 10 and GE14 fuel designs will constitute a "mixed core." Provide details of the impact of the mixed core on the CPR calculations, accounting for the differences in mechanical, thermal and hydraulic characteristics of the two fuel designs in the transition core at BFN Unit 1.

TVA Response

AREVA NP has provided this response on page 25 of Enclosure 2.

NRC Question 13

ANP-2821 P, Section 3.6 Stability:

General Design Criterion 12 of Title 10 of the Code of Federal Regulations, Part 50 Appendix A requires suppression of reactor power oscillations so that the Specified Acceptable Fuel Design Limits are not exceeded. Demonstrate, with supporting analyses and calculations, how thermal hydraulic and neutronic stability of the mixed core will be maintained at the BFN Unit 1 throughout the upcoming and following cycles of operation.

TVA Response

AREVA NP has provided this response on page 26 of Enclosure 2.

NRC Question 14

ANP-2859P Section 2.0:

Provide a reference or summarize the methodology by which BFN Unit 1 is designed to achieve 71 Gigawatt-days of additional energy via final feedwater temperature reduction operation, beyond the full power capability.

TVA Response

The reduction in final feedwater temperature at the Browns Ferry Nuclear Plant is achieved through the removal from service of the last stage of feedwater heaters. The removal of these heaters results in a reduction in the final feedwater temperature of just over 50 °F at the current licensed power level of 3458 MWt. The resulting increase in core inlet subcooling increases the core reactivity, and allows the core to remain at full power for an additional length of time.

The use of final feedwater temperature reduction at the Browns Ferry Nuclear Plant was initially evaluated by General Electric Nuclear Energy in 1994. This evaluation assumed a power level of 3293 MWt (original licensed power), and evaluated a final feedwater temperature reduction of 47°F at rated power. A 10 CFR 50.59 evaluation was performed to add this mode of cycle extension into the design basis. The 10 CFR 50.59 evaluation concluded that the use of final feedwater temperature reduction did not constitute an unreviewed safety question, and therefore prior NRC approval was not required for implementation. The Updated Final Safety Analysis Report (UFSAR) was revised to reflect the use of final feedwater temperature reduction for all three units.

For Browns Ferry Nuclear Plant, Unit 1, TVA docketed License Amendment Requests (LARs) for both a power uprate (to 3458 MWt) and ARTS/MELLLA, prior to the restart of the unit in 2007. The ARTS/MELLLA LAR (TS-430, TAC No. MC1330) was approved by the NRC on September 26, 2006. The power uprate LAR (TS-431, TAC No. MD3048) was approved by the NRC on March 6, 2007. The various safety analysis reports docketed as part of these two LARs specifically considered the impacts of using feedwater temperature reduction. The evaluated temperature reduction in these reports was increased from the original 47°F value to 55°F, to reflect the impact of the higher power level. The UFSAR has been modified to reflect the 55°F feedwater temperature reduction value. The various AREVA NP reports docketed as part of TS-473 consider the use of a final feedwater temperature reduction of up to 55°F at rated power.

NRC Question 15

Shutdown Margin:

Describe the analysis procedure used to ensure that the shutdown margin is within the TS limit throughout the transition cycles. Specifically, address how the eigenvalue biases and uncertainties are determined and accounted for during the transition cycles.

TVA Response

AREVA NP has provided this response on page 27 of Enclosure 2.

NRC Question 16

EMF-2158:

Licensee has used EMF-2158 methodology to perform fuel cycle design and fuel management calculations for the Cycle 10 operation of BFN Unit 1 to generate nuclear data including cross sections, local power peaking factors, and associated uncertainties.

Section 5 of XN-NF-80-19(P)(A), Volume 1, Supplement 3, and Section 9 of EMF-2158(P)(A) together provide very detailed description of the analyses and calculations to determine the traversing in-core probe detector (TIP) uncertainty components for boiling-water reactors. Sections 9.4 and 9.5 provide combined uncertainties for TIP distribution calculations, TIP distribution measurement, net calculated TIP distribution and synthesized TIP distribution uncertainty. Provide details of the calculations and uncertainties listed in Chapter 9 of EMF-2158 applicable to the D-lattice BFN Unit 1 plant. Show that the BFN Unit 1 uncertainties documented in EMF-2158 for D-lattice plants remain conservative.

TVA Response

AREVA NP has provided this response on page 29 of Enclosure 2.