ATTACHMENT 16

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 467

Revision of Technical Specifications to allow utilization of AREVA NP fuel and associated analysis methodologies

Response to NRC Comments Regarding Browns Ferry Unit 1 Proposed Fuel Transition Amendment

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Attached is the non-proprietary version of responses to the Reference 2 request for miscellaneous technical information.

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20004-016 (07/23/2009)

AREVA

AREVA NP Inc.,

an AREVA and Siemens company

Engineering Information Record

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Responses to NRC Comments Regarding Browns Ferry Unit 1 Proposed Fuel Transition Amendment

AREVA NP INC.

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Proposed Fuel Transition Amendment

Safety Related? ☒ YES ☐ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title/Discipline	Signature	P/LP, R/LR, A/A-CRF, A/A-CRI	Date	Pages/Sections Prepared/Reviewed/ Approved or Comments
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Record of Revision

Revision No.	Date	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	9/11/2009	All	New document
001	9/30/09	2.3	Revised last paragraph of response to provide additional clarification.
		2.4	Provided additional detail in response to Item 4.
		2.6, Appendix A	Added second paragraph to response and Appendix A to address core monitoring uncertainty comparison.
		2.7	Provided additional detail in response to Item 7.
		2.8 and 3.0	Revised response based on NRC issuance of SE on additive constants and added Reference 6.
		2.10	Provided additional detail in response to Item 10.

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1.0 INTRODUCTION

The Tennessee Valley Authority (TVA) is preparing a license amendment request (LAR) to submit to the U. S. Nuclear Regulatory Commission (NRC) requesting approval to transition from GE13 and GE14 fuel to AREVA's ATRIUM-10 design. Two meetings were held between the NRC, TVA and AREVA at the NRC's Rockville, MD headquarters on January 28, 2009 and March 16, 2009 in preparation for this LAR submittal. Reference 1 provides the NRC meeting summary from the January 28, 2009 meeting. Enclosure 2 of Reference 1 lists eleven items the NRC requested that TVA include in the LAR submittal. AREVA's responses to the Reference 1 request are presented below.

2.0 AREVA RESPONSES

The following subsections include the specific items identified and present AREVA's discussion about how the item is to be addressed.

2.1 Item 1

The Nuclear Regulatory Commission (NRC) staff mentioned the need for the licensee to address NRC staff concerns raised in previous request for additional information (RAI) for recent boiling-water reactor (BWR) fuel transition license amendment request and in previous RAIs regarding Browns Ferry Units 2 and 3.

Response: *Addressed in Reference 2.*

2.2 Item 2

By letter dated January 30, 2009, AREVA confirmed an NRC staff identified concern regarding the Ohkawa-Lahey void quality correlation. This nonconservatism of the void-quality correlation bias for the Unit 1 American Society of Mechanical Engineers and anticipated transient without scram (ATWS) overpressure calculations should be addressed.

Response: *Addressed in Section 8.2 of Reference 2.*

2.3 Item 3

As a result of recent reviews of Browns Ferry licensing requests, the NRC staff has had the opportunity to review some plant data from Unit 2 during its transition from GE to ATRIUM fuel. During its review, it was noted that operating experience could be used to justify the methods applied to ensure acceptably low traversing incore probes (TIP) root mean square differences. Additionally concerns exist regarding the approaches proposed to ensure that the core monitoring system is sufficiently accurate to be consistent with the conditions on MICROBURN-B2. Helpful information that should be provided include relevant operating data, such as TIP measurements for transition cores representative of Unit 1. These may include data collected at similar plants at extended power uprate (EPU) conditions with mixed cores of ATRIUM-10/GE14 fuel as well as data collected at the stretch power uprate conditions with mixed cores of ATRIUM-10/GE14 fuel at Units 2 and 3.

A discussion of the analysis options and core monitoring system inputs or refinements that were implemented to garner the degree of accuracy would be useful, as a description of those options, inputs, and refinements that will be implemented for Unit 1.

Additionally, a discussion of the impact on core simulation or core monitoring accuracy from operating strategy would be useful. A proposed discussion could include whether the methods challenged for maximum extended load line limit analysis operation or EPU operation.

Response: *All of the TIP statistics calculations performed for Browns Ferry are performed in accordance with the description in Reference 5. Data for Browns Ferry Unit 1 are provided in Reference 2 along with transition cycles for Browns Ferry Units 2 and 3. The Browns Ferry Unit 1 core design is very similar to the core design for both Units 2 and 3 so the TIP statistics are expected to be very similar.*

The process of generating the MICROBURN-B2 inputs for Browns Ferry Unit 1 is identical to that used for Units 2 and 3. The MICROBURN-B2 calculation options used for Browns Ferry Unit 1 are also identical to those used for Units 2 and 3.

Core monitoring accuracy has been evaluated by TIP comparisons consistent with the approved methodology provided in Reference 5. These comparisons have been assessed for a wide variety of core operating strategies from annual cycles to two-year cycles, conventional loading and single rod sequence loading, heavy spectral shift operation as well as minimal flow window operation, and also MELLLA operation, without any observable degradation in accuracy, i.e., the requirements as detailed in Reference 5 have been met under all of these operating conditions.

2.4 Item 4

The NRC staff noted that a quantitative justification of the usage of reference analyses would be essential. If the Units 2 and 3 analyses are going to be heavily referenced, a justification would be needed supporting the applicability and a discussion of the relevant plant design differences should be included. Therefore, if the break spectrum for Units 2 and 3 will not be revisited for its applicability to Unit 1, then the break spectrum analysis should already account for transition core effects, and its applicability should be appropriately justified in concert with Title 10, *Code of Federal Regulations* (10 CFR) Section 50.46 and the appropriate loss-of-coolant accident (LOCA) methodology topical reports.

Additionally, the fact that AREVA does not have generically approved methods for the evaluation of ATWS was discussed. It was recommended that the submittal address how compliance with the applicable requirements (containment temperature and core coolability) will be demonstrated.

It was stressed that should LOCA analyses from the previous fuel type be referenced in the application, a quantitative justification for the continued applicability of the maximum average planar linear heat generation rate limit would be necessary. It will be necessary for differences in the fuel design in terms of stored energy and geometry to be addressed.

Response: *The Disposition of Events Section of the BFE1-9 reload report provides the basis for Units 2 and 3 analyses referenced for Unit 1. The LOCA analyses are for Units 1, 2 and 3 (refer to Section 9.0 of Reference 3). The basis for the LOCA analyses being applicable for transition cores is provided in Appendix A of Reference 3. Justification for continued applicability of the maximum average planar linear heat generation rate limit is based on thermal-hydraulic characteristics of the GE14 (legacy fuel) and ATRIUM-10 fuel designs being similar. Therefore, the core response during a LOCA will not be significantly different for a full core of GE14 fuel or a mixed core of GE14 and ATRIUM-10 fuel. In addition, since about 95% of the reactor system volume is outside the core region, slight changes in core volume and fluid energy due to fuel design differences will produce an insignificant change in total system volume and energy (i.e, the differences in fuel design in terms of stored energy and geometry do not have a significant impact on the core response). Therefore, the ATRIUM-10 and GE14 LOCA analysis and resulting licensing PCT and MAPLHGR limits remain applicable for the mixed core for BFE1-9. In the reload report for each cycle, the new (fresh) ATRIUM-10 fuel type is evaluated to confirm continued applicability of the MAPLHGR limits and to determine if the new fuel type PCT remains bounded by the previously reported licensing basis PCT. Compliance with applicable requirements of ATWS (containment temperature and core coolability) are addressed in Reference 2 Sections 8.1 and 8.3.*

2.5 Item 5

The inputs for the fuel used in the nuclear core simulation as it relates to CASMO4/MICROBURN-B2 should be described along with the source of the isotopic and mass data.

Response: *The U-234 and U-236 concentration levels for each lot of BLEU material are measured prior to the material being put into inventory. These concentrations are directly input to the CASMO-4 calculation for BLEU fuel. The primary effect on BLEU fuel neutronics is decreased reactivity due to U-236 capture. As-built weights of total uranium, U-235 and U-236 are entered in the POWERPLEX-III CMSS data files for each fresh reload of fuel. The effects of U-234 are included in the lumped macroscopic cross sections for structural material and other isotopes not explicitly modeled.*

2.6 Item 6

A description regarding how the thermal mechanical limits are evaluated for legacy fuel should be provided. A discussion will be necessary explaining how compliance with these limits will be demonstrated and monitored for the legacy fuel. If legacy vendor analyses are used to establish the limits, confirmation should be provided that these limits are consistent with the uncertainties in the core monitoring system in terms of assessing operational thermal margin.

Response: *GNF provided steady state LHGR limits for the legacy coresident fuel and method of establishing bounding transient LHGR limits as described in Appendix A of Reference 3. The core monitoring system monitors LHGR consistently with how GNF monitors LHGR.*

TVA has performed an assessment of the LHGR uncertainty and Nodal Pin Power uncertainty and determined that there is no negative impact of monitoring GNF LHGR limits with the AREVA methodology (see Appendix A).

2.7 Item 7

Information will be necessary regarding the legacy fuel similar to the type of information provided for other EPU fuel transitions. It will be useful if this information includes any nuclear and hydraulic data used in the safety analysis.

Response: *Hydraulic aspects are addressed in Reference 4. Nuclear data is generated based on detailed lattice and bundle design information for the legacy fuel, provided to AREVA by TVA, allowing explicit modeling of the legacy fuel for all licensing calculations.*

2.8 Item 8

Disposition of the 10 CFR Part 21 that indicated minimum critical power ratio (MCPR) nonconservatism when using erroneous additive constants in SPCB will need to be addressed, including how MCPR is evaluated for legacy fuel.

Response: *The NRC has provided the SER for the addendum to the SPCB correlation in Reference 6, and therefore no disposition of the Part 21 is required.*

2.9 Item 9

Typically cold eigenvalue targets are established based on AREVA simulation of previous cycle operation between three and five cycles. Given the extended time Unit 1 was shut down and defueled, this degree of data does not exist for Unit 1. A description of the following would be useful: (1) how the cold eigenvalue target is established, and (2) any conservatism in the cold target eigenvalue that compensates for the absence of operational data.

Response: *Addressed in Section 7.1 of Reference 2.*

2.10 Item 10

Analyses performed using the AREVA NP suite of analysis codes require that the core be nodalized. It is possible that the legacy fuel and ATRIUM-10 fuel will include differences in the axial geometric variation. Describe how geometry changes within a node are treated in the steady state and transient analyses. Justify the thermal margins based on any errors or biases observed when standard production analysis assumptions are applied to such instances.

Response: *Most of AREVA's analysis codes nodalize each fuel type explicitly (grouped by hydraulic channels, where a channel consists of one or more assemblies of the same fuel type in XCOBRA and XCOBRA-T and individual assemblies in MICROBURN-B2). The most significant axial geometric variation difference between the ATRIUM-10 and the legacy fuel is the axial location of the top of the part length rods. The part length rod fuel column lengths are explicitly modeled in the nodalization of the CASMO-4/MICROBURN-B2 methodology where the top and bottom of part length fuel rods in the GE14 and ATRIUM-10 fuel coincide with standard node boundaries. For hydraulic applications the modeling method for part length rods is treated the same for both the ATRIUM-10 and legacy fuel, refer to Section 4.0 of Reference 2. For geometry changes within a calculational node, the impact on hydraulics is appropriately modeled for both ATRIUM-10 and legacy fuel. COTRANSA2 uses a 1-D model for the core and therefore is based on the average geometry of the ATRIUM-10 and legacy fuel in the core. Because of the simplification of the core, the COTRANSA2 hydraulic solution is rebalanced to match the pressure drop from MICROBURN-B2; likewise the core neutronic response is renomalized to match the response from MICROBURN-B2. The addition of legacy fuel does not increase error or biases relative to thermal margin.*

When standard production analyses assumptions are applied, the impact of a geometry change in a node occurs in the hydraulic calculations. For the ATRIUM-10 and legacy fuel, the geometric variations occur near node boundaries, the impact is estimated as a fraction of an inch of surface friction. This impact is small relative to the pressure drop methodology uncertainty and therefore has an insignificant impact on thermal margins; error (uncertainty) and biases are not increased. The addition of legacy fuel does not increase error or biases relative to thermal margin.

2.11 Item 11

A description of any aspects of ATRIUM-10 BLEU fuel that would differentiate it from ATRIUM-10 fuel in terms of steady state or transient: neutronic, thermal hydraulic, or mechanical performance should be provided.

Response: *AREVA has assessed the effects of BLEU fuel on all neutronic, thermal-hydraulic, and mechanical fuel performance parameters. The only item that differentiates BLEU fuel from non-BLEU fuel is the isotopic differences associated with BLEU. The BLEU material meets the CGU (Commercial Grade Uranium) specification (ASTM C996-96) with the exception of the uranium isotopes U-232, U-234, and U-236. The behavior of these uranium isotopes under irradiation is well understood. The lattice depletion (CASMO-4) and 3-D core simulator (MICROBURN-B2) codes track these isotopes to account for the off-spec concentrations. The primary effect of these isotopes on BLEU fuel is decreased reactivity due to the higher concentration of U-236. The CASMO-4/MICROBURN-B2 methodology explicitly models the U-236 with cross section data for a range of temperatures and voids and U-236 is included in the depletion chains for burnup calculations.*

3.0 REFERENCES

1. Letter, E. Brown (USNRC) to Distribution, "Summary of January 28, 2009, meeting with the Tennessee Valley Authority Regarding Proposed Fuel Transition Amendment (TAC No. ME0438)," ADAMS Accession Number ML090780181, Dated March 23, 2009

2. ANP-2860P Revision 1, *Browns Ferry Unit 1—Summary of Responses to Requests for Additional Information*, September 2009.

3. ANP-2638P Revision 1, *Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions*, September 2009.

4. ANP-2807(P) Revision 0, *Browns Ferry Unit 1 Thermal-Hydraulic Design Report for ATRIUM™-10 Fuel Assemblies*, June 2009.

5. EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, October 1999.

6. Letter, T. B. Blount (US NRC) to R. L. Gardner (AREVA), "Final Safety Evaluation for AREVA NP, Inc. (AREVA) Topical Reports (TR) EMF-2209(P), Revision 2, Addendum 1, "SPCB Additive Constants for ATRIUM-10 Fuel" and ANP-10249(P), Revision 0, Supplement 1, "ACE Additive Constants for ATRIUM-10 Fuel" (TAC Nos. MD8754 and ME0162)," September 23, 2009.

APPENDIX A:

TVA Nuclear, Nuclear Fuel Engineering - BWRFE

Memorandum dated september 28, 2009

TVA Nuclear

1101 Market Street, Chattanooga, TN 37402

Nuclear Fuel Engineering - BWRFE

Date: September 28, 2009
EDMS: L32 090923 801

To: G. C. Storey, Manager, Nuclear Fuel Engineering - BWRFE, COO, LP 4G-C

From: T. W. Eichenberg, Senior Specialist, Nuclear Fuel Engineering - BWRFE, COO, LP 4G-C TWE

Subject: Browns Ferry Unit 1 Cycle 9: Comparison of GNF LHGR Limit Uncertainty with AREVA Core Monitoring Uncertainty

Ref: 1: NEDC-33173P, **Applicability of GE Methods to Expanded Operating Domain**, GE Energy Nuclear, February, 2006.

Ref: 2: Letter from Ho K. Nieh (USNRC) to Robert E. Brown (GEH), Final Safety Evaluation for General Electric (GE)-Hitachi Nuclear Energy Americas, LLC (GHNE) Licensing Topical Report (LTR) NEDC-33173P, "*Applicability of GE Methods to Expanded Operating Domains*", (TAC NO. MD0277), January 17, 2008.

Ref: 3: 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.

Introduction

As part of the BFN Unit 1 Technical Specification (TS) License Amendment Request (LAR) supporting transition to AREVA ATRIUM-10 fuel, the NRC has requested unique information as part of the submittal package. One aspect of the information involves the applicability of GNF LHGR limits within the context of the AREVA core monitoring system.

Review

GEH submitted Reference 1 as part of the NRC evaluation of methods applicability for EPU. TVA identified the GNF LHGR uncertainty for mechanical and thermal limits, found in Reference 1, (page 2-29). The NRC issued the final safety evaluation for Reference 1, per Reference 2. Reference 2 identifies the identical LHGR uncertainty (page 44), as reflected in Reference 1. A direct comparison can be made between the Reference 1 LHGR uncertainty and the Nodal Pin Power uncertainty from the AREVA core monitoring methodology basis, found in Reference 3, (page 2-6 Table 2.3).

Conclusion

TVA has reviewed the LHGR uncertainty, and Nodal Pin Power uncertainty identified in References 1, 2, & 3. Based upon the review, TVA has concluded that the reported and accepted uncertainty in the GNF LHGR methodology bounds the uncertainty of the AREVA monitoring methodology. Consequently, there is no negative impact of monitoring GNF LHGR limits with the AREVA methodology.

Reviewer Concurrence: William B. Bird 9/29/2009

W. B. Bird, Engineer, Nuclear Fuels Engineering, BWRFE Date

cc:

EDMS [L32 090923 801]
BFE Files (BFE-2870)