UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 3, 2012

Mr. Joseph W. Shea
Corporate Manager - Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENTS REGARDING THE TRANSITION TO AREVA FUEL (TAC NO. ME3775) (TS-473)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant (BFN), Unit 1. This amendment is in response to your application dated April 16, 2010, as supplemented by letters dated February 23, May 12, October 7, 2011, and April 18, 2012. The scope of the request focused on the Unit 1 transition to AREVA fuel, addition of AREVA NP analyses methodologies to the list of approved methods to be used in determining the core operating limits in the core operating limits report, and additional technical specification changes to reflect the AREVA NP specific methods for monitoring and enforcing the thermal limits. The requested approval for transition to AREVA fuel is in support of Unit 1 operation at 105 percent original licensed thermal power level.

It should be noted that the licensee acknowledged the existence of a potential issue identified under Part 21 to Title 10 to the *Code of Federal Regulations* (Part 21) concerning increased leakage in the core spray system that could affect the submitted analyses. The licensee stated that this issue applies only to the core spray of Unit 3. Provided that this remains true, this approval is unaffected by the Part 21 issue with increased core spray leakage.

The NRC staff review of this fuel transition request and another Unit 3 outage-related request was delayed by two identified issues with the proposed emergency core cooling system evaluation model. First, the originally proposed analysis was not reflective of the as-built plant configuration, due to a potential single failure vulnerability of the automatic depressurization system. Second, the emergency core cooling system model originally proposed for implementation for Unit 1 did not represent the reactor system in sufficient detail to model the complex two-phase conditions present following a small break loss-of-coolant accident.

It should be noted that this is a unit-specific review and the conclusions are not applicable to other units or facilities, or at extended operating conditions. A copy of the Safety Evaluation is also enclosed. A proprietary version of this letter and safety evaluation was issued in a letter to the TVA dated April 27, 2012. This letter transmits a non-proprietary version. **Proprietary information was removed, as indicated by empty bold-face double-brackets, such as [[ ]].** A Notice of Issuance was included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Eva A. Brown, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosures:
1. Amendment No. 281 to DPR-33
2. Safety Evaluation

cc w/ encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 281
Renewed License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated April 16, 2010, as supplemented by letters dated February 23, May 12, October 7, 2011, and April 18, 2012, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-33 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 281, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to the beginning of Unit 1 Cycle 10.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: April 27, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 281

RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace Page 3 of Renewed Operating License DPR-33 with the attached Page 3.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.3-7	3.3-7
3.3-29	3.3-29
3.3-30	3.3-30
3.7-16	3.7-16
5.0-24	5.0-24
5.0-24a	5.0-24a
5.0-24b	5.0-24b
5.0-24c	5.0-24c

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(e) Refer to COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

<u>OR</u>

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable EOC-RPT, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER $\geq$ 30% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Restore channel to OPERABLE status.	72 hours
	<u>OR</u>		
	A.2	-----NOTE----- Not applicable if inoperable channel is the result of an inoperable breaker. ----- Place channel in trip.	72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. <u>AND</u> MCPR and LHGR limits for inoperable EOC-RPT not made applicable.	B.1	Restore EOC-RPT trip capability.	2 hours
	<u>OR</u>		
	B.2	Apply MCPR and LHGR limits for inoperable EOC-RPT as specified in the COLR.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Reduce THERMAL POWER to < 30% RTP.	4 hours

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

5.6 Reporting Requirements (continued)

5.6.4 (Deleted).

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

(1) The APLHGRs for Specification 3.2.1;

(2) The LHGR for Specification 3.2.3;

(3) The MCPR Operating Limits for Specification 3.2.2;

(4) The period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and

(5) The RBM setpoints and applicable reactor thermal power ranges for each of the setpoints for Specification 3.3.2.1, Table 3.3.2.1-1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, Revision 16, General Electric Standard Application for Reactor Fuel, October 2007.

2. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, March 1984.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. XN-NF-85-67(P)(A) Revision 1, Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, Exxon Nuclear Company, September 1986.

4. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

5. ANF-89-98(P)(A) Revision 1 and Supplement 1, Generic Mechanical Design Criteria for BWR Fuel Designs, Advanced Nuclear Fuels Corporation, May 1995.

6. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Exxon Nuclear Company, March 1983.

7. XN-NF-80-19(P)(A) Volume 4 Revision 1, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, Exxon Nuclear Company, June 1986.

8. EMF-2158(P)(A) Revision 0, Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURNB2, Siemens Power Corporation, October 1999.

9. XN-NF-80-19(P)(A) Volume 3 Revision 2, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, Exxon Nuclear Company, January 1987.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

10. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Exxon Nuclear Company, February 1987.

11. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, ANF Critical Power Methodology for Boiling Water Reactors, Advanced Nuclear Fuels Corporation, November 1990.

12. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, Advanced Nuclear Fuels Corporation, August 1990.

13. ANF-1358(P)(A) Revision 3, The Loss of Feedwater Heating Transient in Boiling Water Reactors, Framatome ANP, September 2005.

14. EMF-2209(P)(A) Revision 3, SPCB Critical Power Correlation, AREVA NP, September 2009.

15. EMF-2245(P)(A) Revision 0, Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, Siemens Power Corporation, August 2000.

16. EMF-2361(P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP Inc., May 2001 as supplemented by the site-specific approval in NRC safety evaluation dated April 27, 2012

17. EMF-2292(P)(A) Revision 0, ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients, Siemens Power Corporation, September 2000.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. EMF-CC-074(P)(A), Volume 4, Revision 0, BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2, Siemens Power Corporation, August 2000.

19. BAW-10255(P)(A), Revision 2, Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code, AREVA NP, May 2008.

(continued)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 281

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 INTRODUCTION

By letter dated April 16, 2010 (the submittal), as supplemented by letters dated February 23, May 12, and October 7, 2011, and April 18, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML101160153, ML110590054, ML11137A199, ML11286A107, and ML12114A004), the Tennessee Valley Authority (TVA, the licensee) submitted a request for changes to the Browns Ferry Nuclear Plant, Unit 1 (BFN) Technical Specifications (TSs). The requested changes would add the AREVA NP analysis methodologies to the list of approved methods to be used in determining the core operating limits in the core operating limits report (COLR). This change is needed to support the transition to AREVA fuel in BFN. Additional TS changes are requested to reflect the AREVA NP specific methods for monitoring and enforcing of the thermal limits.

These changes support the transition from Global Nuclear Fuel (GNF) GE14 design to the AREVA ATRIUM-10 fuel design commencing with the reload batch in the fall of 2012 in Unit 1. The initial Unit 1 reload and at least one follow-on reload will utilize Blended Low Enriched Uranium (BLEU) provided to TVA under a joint project with the Department of Energy. The main difference between BLEU and commercial grade uranium is in the isotopic content of the uranium (typically around 0.09 percent U-234 and 1.6 percent U-236 is contained in the BLEU fuel).

The supplements dated February 23, May 12, October 7, 2011, and April 18, 2012 (References 2, 23, 24, and 25), provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 10, 2011 (76 FR 1467). The NRC staff notes that the October 23 and November 17, 2009, letters referenced in the January 10, 2011, *Federal Register* Notice were subsequently withdrawn by the licensee in a letter dated February 2, 2010 (ADAMS Accession No. ML100341331).

2.0 REGULATORY EVALUATION

2.1 Requirements and Guidance

In Section 50.36 to Title 10 to the *Code of Federal Regulations* (10 CFR), the Commission established its regulatory requirements related to the content of the TSs. Consistent with 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs [Limiting Conditions for Operation]; (3) surveillance requirements; (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TSs.

Section 50.46 to 10 CFR, establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.

Section 50.62 to 10 CFR, as it relates to the acceptable reduction of risk from anticipated transient without scram (ATWS) events via (a) inclusion of prescribed design features and (b) demonstration of their adequacy, requires that light-water-cooled plants be equipped with the systems and equipment that are designed to reduce risks attributable to ATWS events to an acceptable level. The rule also requires a demonstration of the adequacy of the features as specified in the NRC requirements and the discussion contained in the statement of considerations of the final rule.

Appendix K to 10 CFR, establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a loss-of-coolant accident (LOCA).

The licensee's submittal referenced the following Appendix A to 10 CFR Part 50, General Design Criteria:

- (GDC)-10, *Reactor design*, requires the reactor design (reactor core, reactor coolant system (RCS), control and protection systems) to assure that specified acceptable fuel design limits (FDLs) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).

- GDC-27, *Combined reactivity control system capability*, requires the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the ECCS, to reliably controlling reactivity changes under postulated accident conditions.

- GDC-35, *Emergency core cooling*, requires a system to provide abundant emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

The following explains the applicability of GDC for Unit 1. The construction permit was issued by the Atomic Energy Commission (AEC) on May 10, 1967, and the operating license was issued on December 20, 1973. The plant GDC are listed in Appendix A to the Final Safety Analysis Report, *Conformance to AEC Proposed General Design Criteria*, with more details given in the applicable Updated Final Safety Analysis Report (UFSAR) sections.

In accordance with an U. S. Nuclear Regulatory Commission (NRC) Staff Requirements memorandum from S. J. Chilk to J. M. Taylor, *SECY-92-223 - Resolution of Deviations Identified during the Systematic Evaluation Program*, dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply Part 50 Appendix A to 10 CFR, *General Design Criteria for Nuclear Power Plants,* to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC that constitute the licensing bases for Unit 1 are those in the UFSAR, except as required by 10 CFR. The NRC staff identified the following AEC GDC as being applicable to the proposed amendment:

- AEC GDC-7, *Suppression of Power Oscillations*, requires that the core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

- AEC GDC-13, *Fission Process Monitors and Controls*, requires that means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

- AEC GDC-14, *Core Protection Systems*, requires core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

- AEC GDC-15, *Engineered Safety Features Protection Systems*, requires that protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

- AEC GDC-16, *Monitoring Reactor Coolant Pressure Boundary*, requires that means shall be provided for monitoring the reactor coolant pressures boundary to detect leakage.

- AEC GDC-17, *Redundancy of Reactivity Control*, requires at least two independent reactivity control systems, preferably of different principles, shall be provided.

- AEC GDC-30, *Reactivity Holddown Capability*, requires at least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

- AEC GDC-29, *Reactivity Shutdown Capability*, requires at least one of the reactivity control systems provided shall be capable of making the core subcritical under any conditions (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

- AEC GDC-31, *Reactivity Control Systems Malfunction*, requires that the reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

- AEC GDC-32, *Maximum Reactivity Worth of Control Rods*, requires that limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

- AEC GDC-33, *Reactor Coolant Pressure Boundary Capability*, requires that the reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

- AEC GDC-40, *Missile Protection*, requires protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

- AEC GDC-42, *Engineered Safety Features Components Capability*, requires that engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

In an effort to avoid TS changes for every fuel reload cycle that results in changing cycle-specific parameter limits, licensees have relocated the cycle-specific core operating parameters from the TSs to the COLR. This is done in accordance with the guidelines of Generic Letter (GL) 88-16, *Removal of Cycle-Specific Parameter Limits from Technical Specifications*. It also provides that licensees identify, in the TS section titled, *Reporting Requirements*, the previously approved analytical methods used to determine the core operating limits by identifying the topical report number, title, and date (or identify by NRC letter and date the safety evaluation (SE) for a plant-specific methodology). The NRC staff approved the request to remove the cycle-specific parameter limits to the COLR for BFN in a SE dated May 20, 1993 (ADAMS Accession No. ML020030217).

The *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (SRP, NUREG-0800), Section 4.2, *Fuel System Design*, Section 4.3, *Nuclear Design*, and Section 4.4, *Thermal and Hydraulic Design*, provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and thermal and hydraulic design of the core. In addition, the SRP provides guidance for compliance with the applicable GDC in Appendix A to 10 CFR Part 50. According to SRP Section 4.2, the fuel system safety review provides assurance that:

- The fuel system is not damaged as a result of normal operation and AOOs,

- Fuel system damage is never so severe as to prevent control rod insertion when it is required,

- The number of fuel rod failures is not underestimated for postulated accidents, and
- Coolability is always maintained.

2.2 System and Accident Background

2.2.1 Background

Unit 1 is a General Electric (GE) designed boiling-water reactor (BWR) of the BWR/4 class. It is currently licensed to operate at 3458 megawatts thermal (MWt), with a calorimetric power measurement accuracy of 2 percent.

Its ECCS includes a single, turbine-driven, high pressure coolant injection (HPCI) system that injects into the reactor core via the feedwater piping, which is located in the downcomer region at an elevation above the core. The HPCI can be aligned to take its suction from the condensate supply header of the suppression pool. The HPCI provides cooling in the event of a small break LOCA, which is roughly characterized by a break size insufficient to lower reactor pressure to the cut-in pressure of the low pressure ECCS equipment. Motive force for the HPCI pump is provided by a steam turbine that is supplied by the nuclear system. Its control system is battery-powered, making it independent of the offsite and diesel-powered electrical supply systems.

The HPCI is bolstered by an automatic depressurization system (ADS). The ADS is a control system that, after a time delay, opens up to six main steam relief valves (MSRVs) in order to depressurize the reactor to the cut-in pressure of the low pressure ECCS. The ADS is also designed to mitigate small break LOCAs. Its control system is powered by the Board B station battery, and the MSRVs are air-operated valves. In addition to a reliable instrument air supply system, the ADS valves have a nitrogen accumulator with sufficient nitrogen to permit ADS valve operation in the event that instrument air is lost.

The low pressure ECCS includes the core spray (CS) and low pressure coolant injection (LPCI) systems. The low pressure ECCS is designed to provide makeup cooling in the event of a large break LOCA that results in a rapid depressurization and emptying of the reactor vessel, or in the event that the ADS successfully operates to reduce the reactor coolant system pressure following a small break LOCA.

The CS system is designed to provide top-down spray cooling in the core to mitigate the effects of large break LOCAs, and for small break LOCAs following ADS operation. The CS consists of two separate loops. Each loop consists of two 50-percent capacity pumps, a spray sparger in the reactor vessel above the core, piping and valves to convey water from the pressure suppression pool to the sparger, and the associated controls and instrumentation.

The LPCI system is an operational mode of the residual heat removal (RHR) system, and it accomplishes similar functions to CS by providing low-pressure, bottom-up reflood cooling to the core. The LPCI system at Browns Ferry is comprised of two separate divisions, each providing emergency core coolant injection to a single recirculation loop. Cross-tie equipment between the two divisions, at Unit 1, has been eliminated, although the system is cross-connected to Unit 2. The basic components of the RHR system – for each division – that are essential for LPCI operation include suction piping from the pressure suppression pool, two pumps in parallel, two

heat exchangers, associated valves and piping to form an injection alignment from the suppression pool to the recirculation loop, and the associated controls and instrumentation.

2.2.2 Accident Sequence Overview

2.2.2.1 Large-Break LOCA

The large break LOCA effectively presents two challenges to the fuel clad surface temperature as the sequence progresses. The opening of a large break in a recirculation loop causes core flow degradation as water, reversing through the jet pumps of the broken loop, exits the break. The depressurization resulting from the loss of coolant accelerates as the downcomer liquid level falls below the recirculation line and the discharge flow transitions from a two-phase mixture to predominantly steam. As the RCS depressurizes, liquid coolant in the lower plenum flashes to steam, expelling a two-phase mixture into the fuel and bypass regions of the core. Once the lower plenum flashing reduces, liquid from the downcomer and possibly the core region then begins to drain into the lower plenum. The two possible times of heatup are the initial phase during the core flow degradation, and the second as the liquid returns from the core region to the lower plenum.

The core response during a large break LOCA (LBLOCA) is dominated by void reactivity effects as the liquid moderator is evacuated from the core. Control rods also provide negative reactivity. The fuel temperature cools as energy is transferred from the fuel to the cladding, and into the coolant.

The effects of the LBLOCA are mitigated primarily by the low pressure ECCS. The CS system provides spray cooling above the core, although its effectiveness may be initially compromised by counter current flow limitations as steam rises through the heated region of the core. The LPCI also provides cooling flow from the intact recirculation loop, into the downcomer, through the lower plenum, and into the core.

2.2.2.2 Small Break LOCA

The ADS generally aids in depressurizing the reactor vessel during a small break LOCA, driving the accident to follow a sequence roughly similar to the large break LOCA discussed above.

Prior to ADS actuation, however, a slow reduction in inventory can be expected. If the break size is small enough, the pressure control system will maintain a constant reactor pressure. A main steamline isolation would then cause the system pressure to increase as the high pressure ECCS inject coolant at a rate faster than it is being expelled through the break.

The small break LOCA (SBLOCA) thus progresses more slowly, and is therefore sensitive to both the decay heat load, and the ADS blowdown capacity.

2.2.3 ATRIUM-10 Fuel Design

This AREVA ATRIUM-10 design utilizes a 10x10 array of fuel rods, with 83 full length fuel rods and eight partial length fuel rods. The partial length fuel rods are approximately two-thirds the length of the full length fuel rods. The use of partial length rods improves fuel utilization in the high void upper region of the bundle and also enhances cold shutdown margin, stability, and pressure

drop performance.

Topical Report ANP-2877P, *Mechanical Design Report for Browns Ferry, Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105 [percent] OLTP [original license thermal power])*, AREVA NP, Inc., Revision 0), November 2009, provides a design description, mechanical design criteria, fuel mechanical analysis results, and test results for the ATRIUM-10 fuel assembly design evaluated according to the AREVA BWR generic mechanical design criteria. The generic design criteria have been approved by the NRC and the criteria are applicable to the subject design. The fuel channel design was evaluated to the criteria given in EMF-93-177(P)(A) Revision 1, *Mechanical Design for BWR Fuel Channels*, Framatome ANP, Inc., August 2005 (Reference 5). Mechanical analyses of the ATRIUM-10 fuel assembly design are performed using NRC-approved design analyses methodology (References 6 through 10). The analyses have evaluated the proposed fuel for Unit 1 for maximum assembly average exposure of **[[ ]]** and a maximum rod average exposure for full-length fuel rod of **[[ ]]**. The AREVA ATRIUM-10 design uses an interior water channel rather than water rods to increase neutron moderation.

3.0 TECHNICAL EVALUATION

3.1 Mechanical Design of Unit 1 Reload ATRIUM-10 Fuel

3.1.1 Fuel Assembly and Fuel Channel

The ATRIUM-10 fuel assembly consists of a lower tie plate (LTP) and upper tie plate (UTP), 91 fuel rods, 8 spacer grids, a central water channel, and miscellaneous assembly hardware. Of the 91 fuel rods, 8 are part-length fuel rods (PLFRs). The structural connection between the LTP and UTP is provided by the water channel. Seven spacers occupy the axial locations and the eighth spacer is located just above the LTP to restrain the lower ends of the fuel rods. Table 2.1 of Reference 3 lists the main fuel assembly and components with materials specification and values for dimensions.

The spacer grid of ATRIUM-10 assembly is the ULTRAFLOW design with square grid of intersecting Zircaloy-4 strips with Nickel alloy 718 springs. Each cell has two springs and two opposing supports. A larger cell for the water channel is located one cell spacing off-center in a diagonal direction away from the control blade corner. Small swirl vanes are situated on the top edges of each cell and along the peripheral side strips of the grid.

The water channel which is a structural tie between the LTP and UTP is in the shape of a square duct with rounded corners made from Zircaloy-4 sheet.

The lower tie plate assembly consists of the diffuser box of the FUELGUARD grid that is cast of low carbon steel and is designed to increase the debris filtering capability. The LTP does not provide lateral restraint capability for the full-length fuel rods, instead, the full length rods rest on top of the grid rods and the lateral restraint is provided by the lowermost spacer grid. The UTP is cast of low carbon stainless steel and machined.

The UTP connection has been made simpler and more robust through the Harmonized Advanced Load Chain (HALC) modification process as discussed in Reference 3. The HALC improvement consists of installing a compression nut on the end of the connecting bolt and is used to retain the locking lug, locking ring, and locking spring. To remove the UTP, the compression spring only needs to be depressed enough to unseat the locking lug. The locking ring and locking spring must then be compressed to rotate the locking lug to align and engage with the locking ring. The UTP can then be removed. Installation is accomplished in the reverse manner.

The full length and part-length fuel rods (PLFRs) contain fuel pellets composed of sintered uranium dioxide (UO_2) or uranium dioxide - gadolinia (UO_2-Gd_2O_3) and clad in Zircaloy-2 (Zr-2). The PLFRs have a special lower end fitting that engages in bushings in the LTP grid to keep them in the proper axial location in the fuel assembly with the lower ends engaged in the lowermost spacer grid.

The fuel channel is a square duct with rounded corners, open at both ends and encloses the sides of each fuel assembly for the main purpose of providing a flow boundary between the active coolant flow and the core bypass flow. Gussets are welded at two opposite corners of the top end of the fuel channel for support and attachment to the fuel assembly. The fuel channel outer geometry is designed for compatibility with the control blade.

3.1.2 Fuel Rod Evaluation

The ATRIUM-10 fuel rod design was evaluated according to the AREVA BWR generic mechanical design criteria contained in ANF-89-98(P)(A), *Generic Mechanical Design Criteria for BWR Designs, Advanced Nuclear Fuels Corporation*, Revision 1 and Supplement 1, May 1995 (Reference 4). A summary of fuel rod mechanical design evaluation is given below. Internal hydriding is prevented by careful control of moisture and other hydrogenous impurities during fuel fabrication. A fabrication limit of less than or equal to 2.0 parts per million hydrogen is specified for fuel pellet fabrication.

Cladding collapse is evaluated using the RODEX2A and COLAPX codes XN-NF-85-74(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model*, Exxon Nuclear Company, August 1986 and XN-NF-82-06(P)(A), *Qualification of Exxon Nuclear Fuel for Extended Burnup*, Revision 1 and Supplements 2, 4, and 5, Exxon Nuclear Company, October 1986 (References 9 and 6, respectively). Cladding creep collapse and subsequent potential failure is avoided by restricting the formation of pellet/clad axial gaps during the first **[[** **]]** rod exposure. Once the fuel reaches an exposure **[[** **]]**, the majority of the initial densification is considered to be complete and there is no longer a concern for the formation of axial gaps in the fuel column due to densification. In Reference 2, AREVA reports that the gap condition is evaluated at the first calculation step just **[[**

]] when creep collapse can be a concern.

Prevention of potential fuel failure from overheating of the cladding is accomplished by minimizing the probability of exceeding thermal margin limits on limiting fuel rods during normal operation

and AOO.

Overheating of the fuel pellets is prevented by maintaining the fuel centerline temperature below the melting point of UO_2. The normal operating temperatures are calculated using the design power history. For AOOs, in addition to the design power history, additional calculations are performed at elevated power levels as a function of exposure corresponding to the Protection Against Power Transients (PAPT) linear heat generation rate (LHGR) limit. RODEX2A code has the capability to evaluate **[[**

]].

A design power history, which is used as input to RODEX2A for the fuel rod analyses, is illustrated in Figure 3.1 of Reference 3. This power history was derived from the normal operating LHGR limit as shown in Figure 1.1 of Reference 3. Figure 3.1 of Reference 3 also shows a gadolinia fuel rod power history. **[[**

]]

AREVA uses a **[[**

]] The LHGR limitation for gadolinia rods also occurs in relation to the PAPT limit as shown in Figure 1.1 of Reference 3. Figure 1.1 shows that the PAPT limit is **[[**

]].

During the transient **[[**

]]. The calculated fuel centerline temperatures from RODEX2A are plotted in Figures 3.2, 3.3, and 3.4 of Reference 3 for the UO_2, gadolinia and PLFR, respectively. In all cases, the calculated temperatures at the overpower levels are less than the fuel melting temperature.

Cladding strain caused by transient-induced deformations of the cladding is calculated using the RODEX2 and RAMPEX codes. In addition to transient strain analysis, a steady state creep strain analysis is performed using RODEX2A using the design power history. Results from the analysis, summarized in Table 3.3 of Reference 3, show that the cladding strain satisfies the strain limit of 1 percent. Also, it has been shown that for pellet exposures greater than **[[** **]]** the cladding transient strain is less than **[[** **]]**.

Cladding stresses are calculated at the beginning of life (BOL) and at end of life (EOL) at the cladding outer and inner diameter in the three principal directions. The end cap stresses are evaluated for loadings from differential pressure, differential thermal expansion rod weight, and plenum spring force. Results from the analyses compare with the acceptance criteria for cladding stress.

AREVA does not have explicit limits for the in-reactor densification or swelling. Implicitly, other design criteria such as those related to cladding strain or fuel temperature protect against fuel rod failure due to fuel densification and swelling. Appendix K of [*XN-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, RODEX2, Fuel Rod Thermal-Mechanical Evaluation Model, Exxon Nuclear Company, March 1994*] Reference 11 describes the densification model. When the fuel is fabricated, it is sintered to achieve a large fraction of the theoretical density with the remainder of the volume occupied by small voids and defects. Densification is defined to occur as a result of the size of change of annihilation of small voids during irradiation. The amount of in-reactor densification is based on out-of-reactor resinter tests. The UO_2 fuel exhibit a higher change in density from the resinter test than the gadolinia fuel. Although the actual resinter values are different, the analyses for UO_2 and gadolinia fuel use the same upper specification limits for resinter. Therefore, UO_2 and gadolinia rods are treated the same way with respect to densification.

Gadolinia fuel normally exhibits lower fission gas release than UO_2 fuel rods mainly due to the fact that the gadolinia fuel rods are at lower power levels and lower discharge exposures. [[

]]. The two commitments made by AREVA to the NRC in RODEX2A methodology regarding the gas pressure in the gadolinia bearing fuel and gadolinia power history was found to be observed in connection with the Unit 1 ATRIUM-10 design.

The NRC staff has determined that the fuel rod mechanical design analysis performed using NRC approved design analysis methodology satisfied the acceptance criteria needed for fuel system damage and fuel rod failure with respect to design aspects such as internal hydriding, cladding collapse, overheating of fuel and cladding, stress and strain limits, cladding rupture, and fuel densification and swelling.

3.1.3 Fuel Assembly Evaluation

The ATRIUM-10 fuel system mechanical design satisfies the acceptance criteria set forth in Reference 4. Table 3.3 of Reference 3 lists the fuel system description, the design criteria, and the results of the analyses. A summary of the fuel assembly evaluations is listed below:

- The structural integrity of fuel assembly components is assured by establishing limits on stresses and deformations due to handling, operational, and accident loads. Load and stress limits, as applicable, are derived from the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel* (B&PV) Code Section III.

- Fuel rod cladding fatigue as calculated using the RODEX2 and RAMPEX codes, indicate that the fatigue design curve includes a factor of safety of 2 on stress or 20 on the number of cycles.

- Since the design basis for fretting wear is that fuel rod failures due to fretting shall not occur and there is no specific wear limit, the acceptability of fretting resistance is verified.

- RODEX2A analyses indicate that AREVA BWR fuel cladding oxidation is low (Reference 9) and the maximum amount of wall thinning is taken into account in the cladding steady-state stress analysis for EOL conditions.

- Rod closure measurement data taken on lead fuel assemblies has validated the application of the rod bow model to the ATRIUM-10 design and is assessed for impact on thermal margin.

- Three growth calculations considered for the ATRIUM-10 design: (1) minimum fuel rod clearance between the LTP and UTP, (2) minimum engagement of the fuel channel with the LTP seal spring, and (3) external channel engagement (e.g., channel fastener springs) have shown that the fuel assembly components, including the fuel channel, shall maintain clearances and engagements throughout the design life [Reference 3 and *EMF-85-74(P), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model,* Siemens Power Corporation, Revision 0 Supplement 1(P)(A) and Supplemental 2(P)(A), February 1998 (Reference 10).

- Using RODEX2A and design power history as input, the rod internal pressure is limited to the **[[** **]]**. In addition to evaluating the maximum rod pressure, **[[**

  ]].

- Under both normal operating conditions (including AOOs), the calculated net force is confirmed to be in the downward direction, indicating no liftoff. Under mixed core conditions, the liftoff is considered on a specific basis as determined by the plant and the other fuel types. Analyses indicate that there is large margin to liftoff under normal operating conditions.

- Fuel assembly liftoff analyses under faulted conditions (including seismic and LOCA) for a combination of contributions from gravity, buoyancy, hydraulic forces and momentum have resulted in uplift that is limited to **[[**

  ]].

The NRC staff has reviewed the fuel assembly design evaluated according to the AREVA BWR generic mechanical design criteria and determined that the fuel assembly design has satisfied the acceptance criteria.

3.1.4 Fuel Coolability

Core coolability and ability to insert control blades are essential for accidents in which severe fuel damage might occur. Section 4.2 of the SRP lists areas important to coolability, such as, embrittlement, violent expulsion of fuel, fuel ballooning and structural deformations. The requirements for cladding embrittlement are associated with LOCA requirements 10 CFR 50.46.

For a severe reactivity-initiated accident, the radially averaged energy deposition at the highest axial location is restricted according to the guidelines contained in Regulatory Guideline 1.77, *Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors*. The potential cladding ballooning and flow blockage and cladding rupture are considered in the LOCA analysis according to 10 CFR Part 50 Appendix K requirements.

The ATRIUM-10 fuel assembly was evaluated for structural integrity during external loading by testing and analysis to obtain the dynamic characteristics of the fuel assembly and spacer grids. Tests are done with and without a fuel channel. The dynamic models are compared to the test results to ensure an accurate characterization of the fuel. **[[**

]].

The testing and analyses have shown the dynamic response of the ATRIUM-10 design to be very similar to BWR fuel designs that have the same basic channel configuration and weight. This includes the previously analyzed GNF fuel at BFN. Because the fuel assembly weight and channel stiffness do not vary significantly from prior AREVA fuel designs (or other co-resident fuel types), the maximum loads and deflections for the ATRIUM-10 fuel assembly will be essentially unchanged from before.

The NRC staff has determined that the licensee has satisfied the AEC GDC comparable to the acceptance criteria with respect to specific areas important to fuel coolability, such as, embrittlement, violent expulsion of fuel, fuel ballooning, and structural deformations.

3.1.5 Fuel Channel and Fastener

The stress limits during normal operation are obtained from the ASME B&PV Code, Section III, Division 1, Subsection NG for Level A Service. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation. As an alternative to the elastic analysis stress intensity limits, a plastic analysis may be performed as permitted by paragraph NB-3228.3 of the ASME B&PV Code. **[[**

]].

The fatigue life is based on the O'Donnel and Langer curve, which includes a factor of 2 on stress amplitude or a factor of 20 on the number of cycles, whichever is more conservative. **[[**

]].

The NRC staff has determined that the fuel channel and fastener design for Unit 1 has been implemented as per requirements of appropriate ASME criteria for bending moment, pressure loading, and stress criteria. Therefore, the NRC staff finds that the Unit 1 design has satisfied the comparable AEC GDC and 10 CFR 50.46 requirements to assure that fuel system dimensions

remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.

3.2 Mechanical Tests

AREVA has tested the fuel components to qualify the mechanical design or evaluate assembly characteristics that included:

- Fuel assembly axial load structural strength test

An axial load test was conducted by applying an axial tensile load between the LTP grid and UTP handle of a fuel assembly cage specimen. The load was slowly applied while monitoring the load and deflection. No significant permanent deformation was detected for loads up to [[ ]] times the dry weight of the fuel assembly and fuel channel.

- Spacer grid lateral impact strength test

Spacer grid impact strength was determined by a transverse dynamic load test. A vibration machine is used for the test. [[

]].

- Tie plate lateral load strength tests and LTP axial compression test

The three separate tests that were conducted on the tie plates are (1) The UTP was loaded laterally to obtain a limit for accident conditions, (2) the LTP was subjected to an axial compressive load to simulate handling loads, and (3) the LTP was loaded laterally at the nozzle to determine a load limit for accident conditions. The details of the test are given in Reference 3. Results from the testing were adjusted, accounting for reactor operating conditions, to determine the load limits reported in Table 3.3 of Reference 3.

- Fuel assembly fretting test

The results from a [[

]].

- Fuel assembly static lateral deflection test

AREVA conducted a lateral deflection test to determine the fuel assembly stiffness, by supporting the fuel assembly at the two ends in a vertical position and applying a side displacement at the center spacer and measuring the corresponding force. Results from this test are input to the fuel assembly structural model.

- Fuel assembly lateral vibration tests

The vibration test was performed using a vibration machine to determine the natural frequencies, damping and spacer grid impact stiffness for a fuel assembly. The lateral vibration test was performed on a full-scale fuel assembly in both water and air, and with and without a fuel channel. Two tests, a lateral vibration and a fuel assembly impact test, were performed. Results from the

test were used a basis for selecting fuel assembly stiffness values and damping for the structural model.

- Fuel assembly impact tests

Impact tests for the ATRIUM-10 fuel assembly were done with the same vibration machine as above. Tests were done in air and water. Measured impact loads were used in establishing the spacer in-grid stiffness.

The NRC staff has determined that the mechanical tests performed on the fuel assembly and its components have assured conformance to the fuel design criteria. Therefore the NRC staff finds that the licensee has acceptably verified that existing design-basis limits, analytical models, and evaluation methods remain applicable for the specific design for normal operation, AOOs, and postulated accidents.

3.3 Thermal-Hydraulic Design of ATRIUM-10 Fuel Design

The TVA intends to utilize the AREVA ATRIUM-10 fuel design in Unit 1 starting in the fall 2012 refueling outage. The licensee intends that the ATRIUM-10 fuel assemblies will be coresident with GE14 fuel design. Thermal-hydraulic analyses were performed to verify that the design criteria are satisfied and to establish thermal operating limits with acceptable margins of safety during normal reactor operation and AOOs. The applicable generic thermal-hydraulic design criteria have been approved by the NRC in XN-NF-80-19(P)(A), *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, Exxon Nuclear Company, Volume 4, Revision 1, June 1986 (Reference 12). In Enclosure 8 to the submittal (Reference 13) the licensee determined that the ATRIUM-10 fuel assemblies should be hydraulically compatible with coresident GE14 fuel for the entire range of the licensed power-to-flow operating map for Unit 1. This section summarizes the design criteria that are applicable to the ATRIUM-10 fuel design and how they are satisfied for Unit 1.

3.3.1 Hydraulic Characterization

The design criteria for hydraulic characterization is such that hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to the existing fuel in the reactor such that there is no significant impact on total core flow or the flow distribution among assemblies in the core.

The basic geometric parameters and loss coefficients for both ATRIUM-10 and GE14 fuel designs are listed in Tables 3.2 and 3.3 to Reference 13, respectively. The listed loss coefficients based on the tests are modified to **[[**

]]. The bare rod friction, ULTRAFLOW spacer, and UTP losses for ATRIUM-10 are based on flow tests. The local losses for the BFN ATRIUM-10 FUELGUARD LTP are based on pressure drop tests performed at AREVA's Portable Hydraulic Test Facility. The loss coefficients for the GE14 fuel are based on flow test results.

3.3.2 Hydraulic Compatibility

The thermal hydraulic analyses methodology used by AREVA for the calculation of pressure drop in BWR fuel assemblies are presented in XN-NF-79-59(P)(A), *Methodology for Calculation of*

Pressure Drop in BWR fuel Assemblies, Exxon Nuclear Company, November 1983 (Reference 14). This methodology is implemented in the XCOBRA code, which predicts steady-state thermal-hydraulic performance of the fuel assemblies of BWR cores at various operating conditions and power distributions. Detailed hydraulic compatibility analyses were performed for core GE14, full core ATRIUM-10 and for a mixed core of ATRIUM-10 and GE14 fuel to demonstrate that the thermal-hydraulic design criteria are satisfied for a transition core configuration.

The input conditions for the thermal hydraulic design calculations listed in Table 3.4 of Reference 13 considered two statepoints: 100-percent power/100-percent flow and 62-percent power/37.3-percent flow. Evaluations were made with the bottom-, middle-, and top-peaked axial power distributions presented in Figure 3.1 of Reference 13. Reference 13 presets results from the analyses for bottom-peaked axial power distributions, however, it is reported that the results for middle- and top-peaked axial power distributions show similar trends.

Results presented in Reference 13 for all core configurations indicate that the core average results and differences between ATRIUM-10 and GE14 fuel for rated power results are within the range considered compatible. Similar agreement occurs at other power levels, rated and off-rated conditions. Since the overall pressure drop in the **[[**

]].

Based on the reported changes in pressure drop and assembly flow caused by the transition from GE14 to ATRIUM-10, the ATRIUM-10 design is found hydraulically compatible with the GE14 design, since the thermal-hydraulic design criteria are satisfied.

3.3.3 Thermal Margin Performance

Thermal margin analyses were performed using the methodology prescribed in the thermal-hydraulic analysis code, XCOBRA. Thermal margin performance or critical power ratio (CPR) performance is established by means of an empirical correlation based on the results of boiling transition test program. The CPR empirical correlation use a basic functional relationship between boiling transition and nominal fuel bundle operating conditions of coolant inlet mass flow rate, pressure and inlet enthalpy. This is accomplished by determining correlation coefficients fit to measure critical heat flux data with specific additive constants used to account for specific fuel design parameters.

The CPR values for ATRIUM-10 and GE14 fuel designs are calculated using the SPCB critical power correlation in EMF-2209(P)(A), *SPCB Critical Power Correlation*, Revision 2, Framatome ANP, September 2003 (Reference 15), and the GE14 CPR values are calculated with the SPCB correlation in EMF-2245(P)(A), *Application of Siemens Power Corporation's Critical Power Correlations to Co-resident Fuel*, Revision 0, Siemens Power Corporation, August 2000 (Reference 16). As discussed in Reference 15, assembly design features are incorporated in the CPR calculations through the F-eff [correlation parameter] term. For hydraulic compatibility evaluation, steady state analyses for ATRIUM-10 and GE14 assemblies for radial peaking factors between **[[** **]]** were performed. As indicated in Reference 13, analysis results indicate ATRIUM-10 fuel will not cause thermal margin problems for the coresident GE14 fuel.

To account for the mixed-core configuration as Unit 1 transitions from a full core of GE14 fuel to a full core of ATRIUM-10 fuel, each fuel type is explicitly modeled. This enables the impact of the differences in mechanical design on geometry and loss coefficients are explicitly accounted for. The application of the SPCB critical power correlation to the coresident GE14 fuel is based on the indirect process described in Reference 16.

3.3.4 GE14 Fuel Thermal Mechanical Information

The licensee indicates that ATRIUM-10 fuel assemblies are hydraulically compatible with coresident GE14 fuel in the Unit 1 reactor for the entire range of the licensed power-to-flow operating map. Thermal Overpower (TOP) and Mechanical Overpower (MOP) [[

]].

Conformance to TOP/MOP limits is based on results [[

]].

TVA provided GNF with [[ ]], calculated by AREVA transient code for a spectrum of fast transients. TVA also provided the internal heat generation as a function of time histories for each transient. GNF determined the [[

]]. Both rated and off rated conditions were analyzed.

Internal power generation rates and corresponding AREVA transient code TOP and MOP results were provided for eight (8) feedwater control failure (FWCF) and nine (9) load rejection with no bypass (LRNB) transients. The equations that govern the linear heat generation rate thermal limit flow/power dependent adjustments and multipliers (LHGRFAC) are listed in ANP-2859(P) Revision 0, *Browns Ferry Unit 1 Cycle 9 Fuel Cycle Design (105 [percent] OLTP), AREVA NP Inc., September 2009* (Reference 20). The calculated TOP and MOP for FWCF and LRNB type transients are contrasted for the cases at rated power conditions in Figures 2 through 5 of Enclosure 14 to the submittal (Reference 20). The LHGRFAC values derived based on the equations provided to AREVA and Cylindrical heat transfer (CHT) are compared in Figures 6 and 7 of Reference 20. Based upon [[ ]] AREVA GNF results, GNF defined [[ ]], these limits assure that when AREVA-Transient code is applied to GE14 fuel in Unit 1 that fuel melting and cladding plastic strain licensing limits are satisfied.

3.3.5 Stability

Unit 1 is a detect and suppress Option III plant that that utilizes a power range neutron

monitor-based system. The Option III system is based upon combining groups of local power range monitors (LPRMs). These monitors are combined into cells known as oscillation power range monitors (OPRMS), which generates a signal that is examined for the characteristics of a reactor instability event, and if detected, a reactor trip is generated. Cycle-specific calculations are performed to determine the required Option III system setpoint(s) necessary to ensure that the MCPR Safety Limit is not exceeded. This includes the calculation of the DIVOM (delta over initial minimum CPR (MCPR) versus oscillation magnitude) as well as delta-MCPR response to a two recirculation pump trip event using the cycle-specific licensing basis core. For times in which the primary OPRM system is not available, regions are defined on the power-flow map in accordance with the Backup Stability Protection. These regions have been conservatively defined for Unit 1 and are confirmed on a cycle-specific basis.

The stability performance is a function of the core power, core flow, core power distribution, and to a lesser extent, the fuel design. Only the parameters that are fuel-design specific, such as component dimensions, hydraulic loss coefficients, and void coefficient, are varied for a design comparison. A comparative stability analysis was performed with the NRC-approved code. The study shows that the ATRIUM-10 fuel design has decay ratios equivalent to or better than other approved AREVA fuel designs. Since the stability performance of a core is strongly dependent on the core power, core flow, and power distribution in the core, core stability is evaluated on a cycle specific basis and addressed in the reload licensing report.

The NRC staff evaluated the licensee's thermal-hydraulic design analysis for the Unit 1 transition core containing ATRIUM-10 and GE14 fuel. The NRC staff also evaluated the hydraulic compatibility, thermal margin performance and the stability performance of the transition core. The NRC staff finds that the Unit 1 transition core is conservatively designed to demonstrate that the thermal-hydraulic design and compatibility criteria continue to be met.

3.4 Fuel Cycle Design

The fuel cycle design and fuel management calculations for Unit 1 Cycle 9 operation have been performed with the approved AREVA neutronics methodology in EMF-2158(P)(A), *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, Revision 0, Siemens Power Corporation, October 1999 (Reference 18) and Reference15. The CASMO-4 lattice depletion code was used to generate nuclear data including cross sections and local power peaking factors. The MICROBURN-B2 three dimensional core simulator code, combined with the application of the SPCB critical power correlation, was used to model the core. The Cycle 9 results for core loading, projected control rod patterns and evaluations of thermal and reactivity margins based on Cycle 8 operational history are presented in ANP-2859(P), *Browns Ferry Unit 1 Cycle 9 Fuel Cycle Design (105 [-percent] OLTP)*, AREVA NP Inc., Revision 0, September 2009 (Reference 19).

The loading pattern for Cycle 9 maintains full core symmetry within a scatter load fuel management scheme. The fuel loading in conjunction with control rod patterns presented in Appendix A of Reference 19 shows acceptable power peaking and associated margins to limits for projected Cycle 9 operation. The licensee provided the specific core location of the fresh assemblies in Cycle 9 is provided in Appendix C of Reference 19.

Cycle specific shutdown margin (SDM) calculations are performed as part of the reload licensing analyses. The results are reported in Section 3.3 of Reference 19. The SDM analysis is performed using the NRC approved AREVA CASMO-4/MICROBURN-B2 neutronics analysis methodology documented in Reference 18. The SDM analyses for Cycle 9 were repeated for the case of short, nominal, and long Cycle 8 end-of-cycle (EOC) conditions and at cold conditions with no xenon. A series of one-rod-out cold calculations is performed with MICROBURN-B2 in order to identify the most limiting control rod location in the core. The difference between the cold critical k-effective and the limiting one-rod-out calculated k-effective at cold conditions determines the SDM. Key values for SDM and R-value results are presented in Table 2.1 of Reference 19. The SDM for Cycle 9 is in agreement with the TS limit of greater than 0.38 percent delta k/k at the beginning-of-cycle (BOC). The core hot excess reactivity was calculated at full power with all rods out, 102.5 Mlb/hr core flow, and with equilibrium xenon. Tables 3.4 through 3.6 of Reference 19 summarize the Cycle 9 reactivity margins versus cycle exposure, including the standby liquid control SDM for the cycle.

The NRC staff has reviewed the information and analyses performed for the Cycle 9 fuel cycle analysis and found that the analyses demonstrate adequate hot excess reactivity and cold SDM throughout the cycle. The NRC staff has determined that the projected control rod patterns for Cycle 9 achieve optimum operating flexibility consistent with a conservative margin to thermal limits. It should be noted that the actual Unit 1 Cycle 9 outage began in fall of 2010 and restarted with a full core of GE14 fuel. For the purpose of this review, Unit 1 Cycle 9 represents the first transition core containing both ATRIUM-10 and CE14 fuel. For the purpose of this review, Unit 1 Cycle 9 represents the first transition core containing both ATRIUM 10 and GE14 fuel. The licensee intends to first use this transition core during the Unit 1 Cycle 10 refueling outage proposed for fall 2012.

3.5 <u>Verification of Calculated TIP Distribution Uncertainty</u>

In response to NRC staff's request for more information on the use of EMF-2158 [EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICOBURN-B2*, Siemens Power Corporation, October 1999] methodology (Reference 21), the licensee responded with a comparison between the uncertainties in the measured assembly power and the topical report values for the D-lattice plants (Reference 2). The licensee reports (Reference 2) that the uncertainty components determined using traversing incore probe (TIP) measurements that are taken at or near full power conditions for LPRM calibration have been demonstrated to be independent of the fuel type and are within the values reported in Reference 21. The NRC staff found the uncertainty calculations acceptable.

3.6 <u>Fuel Design Conclusion</u>

The NRC staff reviewed the licensee's request for utilizing the ATRIUM-10 fuel design in Unit 1 Cycle 10. The NRC staff has determined that ATRIUM-10 fuel design maintains neutronic, thermal-hydraulic and mechanical compatibility with the coresident fuel. Further, the NRC staff found that the fuel design has met the criteria specified in the applicable AEC GDC as the licensee has adequately demonstrated that the fuel will not fail during normal operation and AOOs.

3.7 Non-LOCA Safety Analysis

3.7.1 Anticipated Operational Occurrences

The plant responses to the limiting AOOs are analyzed for each reload cycle. The results are used to establish the operating limit minimum critical power ratio (OLMCPR) and to confirm that the ASME overpressure criterion is met. To support the request, the licensee performed a reload transient analysis to cover the projected operating conditions within the licensed power-to-flow map, equipment out of service options, and SCRAM speed options (i.e., TS-required scram speed and nominal scram speed). For the initial application of AREVA fuel and methodology to BFN, the reload analysis consisted of simulation of transient events to cover the rated and off-rated operating conditions. The results were used to determine the OLMCPR limits for ATRIUM-10 and co-resident GE14 fuel.

The OLMCPRs are established so that less than 0.1 percent of the fuel rods in the core are expected to experience boiling transition during an AOO initiated from rated or off-rated conditions. The operating limits are based on the TSs two-loop operation safety limit minimum critical power ratio (SLMCPR) of 1.09 and the TSs single-loop operation of 1.11. The transient delta CPR is added to the SLMCPR to determine the OLMCPR for that transient event. The limiting event sets the OLMCPR for plant operation. The AOO events analyzed for OLMCPR include: load rejection no bypass, turbine trip no bypass, feedwater control failure, loss of feedwater heating, and control rod withdrawal error.

The NRC staff finds the licensee determined the OLMCPR based on the approach that was accepted by the NRC staff. The licensee evaluated MCPR limits at power levels greater than 25 percent. The licensee determined that MCPR limits need not be evaluated below this point. The limiting delta CPR for loss of feedwater heating occurs at 25 percent power and is 0.26.

ASME overpressure analysis is performed to demonstrate compliance with the ASME B&PV Code. The analysis also demonstrates that the plant-specific safety/relief valve configuration has sufficient capacity and performance to prevent the reactor vessel pressure from reaching the safety limit of 110 percent of the design pressure. Main steam isolation valve (MSIV) closure and turbine stop valve closure (without bypass) events were analyzed with the COTRANSA2 systems code assuming 102 percent rated power at both 81 percent and 105 percent rated core flow at the highest cycle exposure. The maximum pressure of 1328 pounds per square inch gauge (psig) occurred in the lower plenum for the MSIV closure event at the 105 percent core flow condition. The results demonstrated that the maximum vessel pressure limit of 1375 psig is not exceeded. The NRC staff finds that ASME overpressure protection is adequately demonstrated.

As discussed in the section above, the NRC staff finds that: (1) the licensee used approved codes and methodologies to perform the AOO analyses; (2) the values used for the input parameters are appropriate in predicting the consequences; (3) the calculated responses of key reactor and system parameters are reasonable; and (4) the results of the analysis show that the fuel integrity and RCS pressure boundary integrity acceptance criteria will be met. Therefore, the NRC staff concludes that the impact of the proposed changes on the AOOs is acceptable.

3.7.2 Thermal-Hydraulic Stability

Unit 1 is currently operating under the requirements of the reactor stability LTS Option III approved by the NRC staff in the GE Licensing Topical Report NEDO-32465-A, *Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications*, BWR Owners Group (BWROG), GE Nuclear Energy, August 1986 (Reference 17). The BFN solution is used when the OPRM is inoperable. The OPRM system is designed to provide for an automatic scram when the system detects power oscillations above the setpoint. The licensee stated that the conditions evaluated are for a postulated oscillation at 45 percent core flow steady state operation and following a two recirculation pump trip from the limiting full power operation state point.

General Electric calculated the DIVOM curves for the licensee. It was found that the curves are not necessarily conservative for the current plant operating conditions for plants implementing Stability Option III. The non-conservatism was addressed by performing calculations for the relative change in CPR as a function of the calculated hot channel oscillation magnitude (HCOM). The NRC staff reviewed Reference 17 and determined that the HCOM portion of the Option III calculation is based on hardware-specific items such as the LPRM assignments and the reactor protection system (RPS) trip logic. The NRC staff therefore finds that the HCOM portion of the Option III calculation is indeed hardware specific and need not change as a result of the proposed changes.

The OPRM system is armed only when plant operation is within the Option III trip-enabled region where the plant is susceptible to instability. Setpoints for the OPRM system are determined in a two-step process that is based on the MCPR. The MCPR margin that exists prior to the onset of oscillations is determined for two scenarios: a two recirculation pump trip from full power at the highest rod line, and steady-state operation at 45 percent core flow with the core at the OLMCPR. From these MCPR values, the change in CPR during an oscillation is assessed to determine the DIVOM curve. The optimum setpoint is high enough to allow sufficient time for reliable oscillation detection, but low enough to preclude the violation of the MCPR safety limit. The setpoint determination is cycle-specific.

The licensee relies on the RAMONA5-FA computer code to calculate the CPR response of the core to regional oscillations on a cycle-specific basis. The licensee confirmed that the application of RAMONA5-FA for Browns Ferry is consistent with BAW-10255PA Revision 2, *Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code,* AREVA NP, May 2008. AREVA's DIVOM approach **[[**

]] to determine the licensing CPR response. The NRC staff finds that the stability based OLMCPR is determined for Browns Ferry consistent with the approved NEDO-32465-A methodology.

When the OPRM system is inoperable, the plant may use an alternate approach to address stability. The current practice with the Option-III system is to use the Backup Stability Protection (BSP) as the backup method. The BSPs include specific requirements for operator action as well as restrictions on operation in certain regions of the power/flow map. These BSP regions are determined using the STAIF methodology documented in EMF-CC-074(P)(A) Volume 4, Revision 0, *BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2,* Siemens Power Corporation, August 2000. The licensee evaluated BSP curves using STAIF to

determine endpoints meeting decay ratio criteria for the BSP Base Minimal Region I (scram region) and Base Minimal Region II (controlled entry region). Stability boundaries based on these endpoints can then be determined using the generic shape generating function. The licensee performed analyses to support operation for both nominal and reduced feedwater temperature conditions (both final feedwater temperature reduction and feedwater heater out of service. The NRC staff finds that the BSP approach is acceptable for Unit 1.

Based on the information discussed above, the NRC staff finds that the stability analysis and evaluation performed provides reasonable assurance that the proposed transition in fuel and methods will not adversely impact BFN ability to satisfy the associated AEC GDC related to reactor design and suppression of reactor power oscillations.

3.7.3 Anticipated Transients Without SCRAM

An ATWS is defined as an AOO followed by the failure of the RPS of the protection system consistent with AEC GDC-14, 15, and 20. The NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME service level C limit of 1500 psig; (2) the peak clad temperature (PCT) is within the 10 CFR 50.46 limit of 2200 degrees Fahrenheit (°F); (3) the peak suppression pool temperature is less than the design limit (281°F for BFN); and (4) the peak containment pressure is less than the containment design pressure (56 psig for BFN). AREVA does not have a generically approved long-term ATWS containment evaluation methodology. Therefore, the NRC staff reviewed the licensee's long term evaluation on a cycle-specific basis.

The licensee provided plant- and cycle-specific ATWS analyses for Units 2 and 3 conditions. For Unit 3 Cycle 14, the ATWS overpressure analysis was performed based on approved COTRANSA2 methodology documented in ANP-2860(P), *Browns Ferry Unit 1- Summary of Response to Requests for Additional Information*, October 2009, ADAMs Accession No. ML093130535). The licensee reported a calculated peak vessel pressure of 1477 psig, assuming pressure regulator failure open followed by pressurization due to MSIV closure. The licensing acceptance criterion of 1500 psig is met; therefore, the NRC finds it acceptable. Additionally, the licensee stated in ANP-2863(P), Revision 1, *Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105 [percent] OLTP*, March 2010 (Reference 25) that the PCT and cladding oxidation responses during an ATWS are bounded by LOCA. Based on fuel performance analyses conducted for BWR ATWS events, the NRC staff finds that there is reasonable assurance that 10 CFR 50.46 criteria will not be challenged during ATWS, and therefore finds the licensee's conclusion acceptable.

In addition to the short-term vessel overpressure and PCT analysis, the long-term suppression pool performance must be evaluated for acceptability during ATWS. Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This, in turn, may impact the amount of steam discharged to the suppression pool and containment. The licensee stated that **[[**

]].

The licensee provided an evaluation to compare [[

]].

The results showed that [[

]]. The licensee concluded that the fuel design difference [[

]] The licensee therefore concluded that the introduction of ATRIUM-10 fuel will not significantly impact the long term ATWS response (suppression pool temperature and containment pressure) and the current analysis remains applicable.

The NRC staff reviewed the results of the licensee's ATWS analysis of record performed by GE. For Unit 1, the limiting ATWS suppression pool temperature is 214.1°F with the design limit at 281°F. The limiting ATWS containment pressure is 21.4 psig with the design limit at 56 psig. The NRC staff finds that without any system modifications, the primary affect of fuel design [[

]] the NRC staff also finds that relatively minor variations in [[

]] can be accommodated by the current margins available to the suppression pool and containment limits.

The NRC staff concludes that the licensee has demonstrated that the required systems are currently installed and that they will continue to meet the requirements of 10 CFR 50.62. In addition, the NRC staff reviewed the information submitted by the licensee related to ATWS and concludes that the licensee adequately accounted for the effects of the proposed fuel and methodology transition on ATWS. Therefore, the NRC staff finds the proposed changes acceptable with respect to ATWS.

3.7.4 Conclusion

Based on the considerations discussed above, the NRC staff finds that the licensee's proposed transition to AREVA fuel and safety analysis methods is acceptable with respect to the non-LOCA safety analyses.

3.8 LOCA Safety Analysis

The LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents.

The NRC staff's review covered:

- The licensee's determination of break locations and break sizes;
- Postulated initial conditions;
- The sequence of events;
- The analytical models used for the analyses and calculations of the reactor power, pressure, flow, and temperature transients;
- Calculations of peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling;
- Functional and operational characteristics of the reactor protection and the emergency core cooling system (ECCS); and,
- Operator actions.

As described in more detail in Section 2.0, the NRC's acceptance criteria are based on AEC GDC-40 and 29 and the current GDC-35 (10 CFR 50.46 incorporates GDC-35 by reference).

3.8.1 LOCA Analyses Review

The licensee submitted an original ECCS evaluation, which used certain calculations for the heat transfer characteristics that the evaluation model did not appear capable of representing accurately or conservatively. Based on the NRC staff's detailed review, the licensee identified the unacceptable calculations and modified the ECCS evaluation model to address the calculational issues. The following addressed the NRC staff's review of the licensee's ECCS evaluation as modified to support Unit 1.

3.8.1.1 EXEM BWR-2000 ECCS Evaluation Model Validation

The EXEM BWR-2000 ECCS evaluation model [EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model, Revision 0, July 1991 (ADAMS Accession No. ML012050396)] is NRC-approved and conforms to the required and acceptable features of ECCS evaluation models as set forth in 10 CFR Part 50, Appendix K. It includes the RELAX system code that is used to calculate the blowdown and reflood stages of the event, and the HUXY code, which is used to calculate the fuel rod heatup in the limiting plane of the hot bundle. Initial fuel rod stored energy is calculated with the RODEX2 code. The EXEM BWR-2000 evaluation model is validated against three runs of the Full Scale Integral Test (FIST), including two large break simulations and one small break. Summaries of FIST and other relevant emergency core cooling experiments are described in NUREG-1230, *Compendium of ECCS Research for Realistic LOCA Analysis*, December 1988.

The two large break simulations included a representative test of a large break scenario in a BWR/6 facility with high and low pressure CSs, and in a BWR/4 configuration with high pressure coolant injection and low pressure spays and injection. The small break was simulated on a BWR/6-type facility.

Given the limiting nature of the SBLOCA, the NRC staff reviewed the RELAX/HUXY assessment

against the 6SB1 test in particular. The 6SB1 test shows an effectively constant pressure as the break opens. The liquid flowing out the break reduces the vessel inventory, while the pressure control system maintains a constant pressure. The level in the vessel falls due to the reduction in liquid, and the reactor trips and the main steamlines isolate on low level signals. The pressure begins a slight increase, but because the two-phase level in the downcomer decreases, the break eventually uncovers. This slows the inventory reduction in the vessel, and causes the pressure to begin falling rapidly. After the receipt of a low-level signal and a 120-second delay, the ADS actuates to further reduce the vessel pressure. The event is shown to be mitigated by the high pressure and low pressure CSs, with enhanced effectiveness provided by the ADS.

The cladding heatup calculations were within the variance observed in the thermocouples at any given plane in the experiment. AREVA claimed that because of conservatisms in the bounding 20-percent multiplier on the decay heat and because the SBLOCA resulted in low predicted peak cladding temperatures, the RELAX/HUXY EM is reasonable for small breaks. Based on the above, the NRC staff finds that the EXEM BWR-2000 ECCS evaluation model successfully simulated the 6SB1 test with reasonable agreement.

3.8.1.2 Browns Ferry ECCS Analysis

In the EXEM BWR-2000 code assessment, AREVA modified the RELAX model to **[[** **]]** duplicate the experimental observations. The results provided by the licensee suggest that a similar modification was not performed for the Unit 1 ECCS evaluation. In particular, the pressure can be seen to **[[** **]]** at the initiation of the break [see Figure 6.1, ANP-2908(P), *Limiting TLO [Two-Loop Operation]*, R*ecirculation Line Break Upper Plenum Pressure*, Revision 0, ADAMS Accession No. ML101160447].

The NRC staff questioned the apparent differences between the RELAX/HUXY validation run that replicated 6SB1 and the analyzed SBLOCA events for BFN. In Enclosure 2 to Reference 23, supplement, the licensee stated that a main steamline isolation was assumed to occur at the initiation of the event. This assumption caused the reactor vessel pressure to increase to the main steam relief valve setpoints and remain there until the high pressure coolant injection cut-in at 65.4 seconds. Additionally in Enclosure 2 to Reference 23, TVA stated that the reactor scrams on the MSIV position, when 90 percent open.

Finally, TVA compared the assumption of the proposed analysis to one more reflective of the 6SB1 case, where the steam lines are modeled in Enclosure 2 to the Reference 23. In the case considered for comparison, TVA asserted that vessel thermal-hydraulic conditions would be similar between the two cases by 250 seconds, which is before the predicted heatup. Based on this, TVA believes that the two events would be largely similar. TVA also cited sensitivity studies that were performed to confirm this assertion.

The NRC staff evaluated TVA's responses concerning the effect of the MSIV modeling on the postulated accident. The early reactor scram and higher reactor pressure may be considered non-conservative relative to modeling main steam pressure control functions. The reactor scram would terminate power generation in the core earlier, both reducing the fuel stored energy and providing a slight reduction in the decay heat load. The higher pressure in the reactor may also serve to accelerate system mass loss as flow exits the break, provided that the break flow is not choked.

The 6SB1 test shows both a top-down and a bottom-up quench during the reflood phase of the accident. The quench fronts appear to meet 1/3 of the way down the fuel bundle. Based on figures provided by the licensee from the limiting transient, it was indeterminate whether the quench front was advancing during the accident. Most notably, the channel temperature is seen to [[ ]] (See Figure 6.16, ANP-2908(P)). At this time, there is an insignificant liquid mass present in the upper plenum and not very much liquid in the lower plenum. It is not clear from the results, therefore, what the source of the liquid is. Note also that, at this time, approximately [[ ]] (amount obtained by integrating the curve depicted in Figure 6.5, ANP-2098(P)).

In Enclosure 2 to the May 12, 2011 supplement, the licensee confirmed that, due to low predicted steam flows in the hot channel, no counter-current flow limitation was predicted, and the hot assembly was being cooled by top-down spray cooling from the CS injection.

The Sector Steam Test Facility (SSTF) experiments showed that spray cooling liquid tended to accumulate in the upper plenum prior to a period at which counter-current flow limitations broke down, and the spray coolant dropped into the core region (NUREG-1230). The coolant initially dropped at the core periphery, followed by the average channels, and finally in the central, high-powered channels. This supports the conjecture that a bottom-up reflood is likely to be more dominant, especially in the hot assemblies. The information provided for the Unit 1 limiting small break did not illustrate this trend.

Per EXEM BWR-2000, the [[ ]]. The spray cooling phenomenon being calculated is likely exacerbated by the tendency of the code [[ ]]. Therefore, spray cooling is being achieved, and its effects are being calculated explicitly, and SSTF has shown that this is a three-dimensional phenomenon that requires distinction between average core, periphery, bypass, and hot channel. Figure 4.2, ANP-2908(P), shows that the RELAX model core region is comprised of [[ ]]. This is not a sufficiently detailed nodalization to predict analytically the countercurrent spray flow distribution that SSTF illustrated was possible.

The NRC staff has identified issues previously with AREVA's approach to modeling countercurrent spray cooling. In its review of an uprate application for another facility, the NRC staff determined that there were differences between its confirmatory calculations and AREVA's for certain, non-limiting small breaks. The NRC staff's confirmatory peak cladding temperature predictions resulted in higher temperature predictions than AREVA's. This was attributed to the counter-current flow limitation correlations employed by AREVA, which the NRC staff did not find acceptable. However, the uprate application for another facility was still approved, because it was clear that the effects of counter-current spray cooling were non-limiting for that facility. Since the limiting PCT case for Browns Ferry benefits from spray cooling, this conclusion cannot be drawn.

3.8.1.3 Small Break Limited Configuration Review

Based on the concern regarding the applicability of the EXEM BWR-2000 evaluation model to the

Browns Ferry small break-limited ECCS configuration, the NRC staff performed an audit of the calculations in detail. TVA prepared a modified ECCS evaluation and augmented the BFN ECCS evaluation with analyses that address concerns with top-down cooling performance.

The NRC staff's audit focused on the analytic results of the model to determine whether the EXEM BWR-2000 ECCS evaluation model has been applied in a manner that demonstrates (1) that the Browns Ferry-specific ECCS performance evaluation conforms to the required and acceptable features of an ECCS evaluation model set forth in Appendix K to 10 CFR Part 50, and (2) that the Browns Ferry ECCS evaluation appropriately demonstrates compliance with the requirements of 10 CFR 50.46. To address the NRC staff concerns, TVA prepared a modified ECCS evaluation and augmented the ECCS evaluation with analyses to address the staff concerns with top-down cooling performance [Audit report (ADAMS Accession No. ML12100A103)].

The ECCS evaluation model had been modified such that, **[[**

]]. This modification resulted in the restriction of countercurrent liquid flow from entering the hot bundle.

In Enclosure 6 to Reference 24, the licensee submitted ANP-3015, *Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis*, Revision 0 (ADAMS Accession No. ML11286A109). This document submitted the revised ECCS Evaluation Summary to address NRC staff concerns with the pressure control assumptions on the break spectrum and the analyzed top-down cooling mechanisms. This revised summary also provided a description of other break sizes, locations and other properties. The analysis was reviewed to determine whether the licensee had calculated the most severe postulated LOCA.

3.8.1.3.1 Updated Break Spectrum Evaluation

The licensee evaluated the LOCA break spectrum at 102 percent of the current licensed thermal power, 3458 MWt, comprised entirely of ATRIUM-10 fuel at BOL conditions. This provides an analysis licensing basis power of 3527 MWt. The 2 percent increase reflects the maximum uncertainty in monitoring reactor power, as required by 10 CFR Appendix K. The limiting fuel assembly in the core was assumed to be operated at a maximum average planar linear heat generation rate limit of 12.5 kW/ft. **[[**

]]. The operating domain of the power/flow map is applicable for the ATRIUM-10. **[[**

]].

The NRC staff reviewed the analysis results performed according to the new method. The results indicated that the limiting recirculation line break was a 0.21 ft^2 split break in the pump discharge with a battery board 'A' failure and a top-peaked axial power shape. The limiting PCT was 1891°F.

This compares to other recent results as follows:

Percent OLTP	ADS Condition	Single Failure	Limiting Break Size	PCT
105	Operable	HPCI/LPCI	0.21 ft^2	1891°F
105	Degraded	HPCI/LPCI	0.5 ft^2	1809°F
105	Degraded	LPCI/ADS	0.25 ft^2	1973°F
120	Operable	HPCI/LPCI	0.5 ft^2	1998°F

The NRC staff reviewed the results and did not identify any significant issues with the results. Minor issues identified by the staff are discussed in the following subsections.

Comparison of Revised Analytic Results to Prior Results

As discussed in ANP-2908(P), the prior, degraded ADS analyses indicated that the fuel cladding would exhibit a single heatup due to the lack of steam cooling associated with the prompt depressurization and flashing brought about by timely ADS initiation. However, the ADS-operable analyses indicated two cladding heatups: one associated with **[[**

]], and a second heatup as the system **[[**
]]. The second heatup typically extended through the time CS reached rated flow, at which point the HUXY heatup analysis begins to use Appendix K spray heat transfer coefficients. The heatup typically continued for twenty to fifty seconds until a core reflood occurred, at which time the cladding surface temperature would decrease.

Overall, the modified analysis resulted in a change of the limiting single failure case, and a small adjustment to the limiting break size. Due to the modeled hardware modification, a PCT benefit was realized in the overall break spectrum.

Break Spectra Discrepancies

Previous review activities associated with the BFN extended power uprate amendment requests revealed that there was a discrepancy in the limiting break size among LOCA break spectra calculated by the NRC staff and GE. Section 50.46(a)(1)(i) requires licensees to analyze a number of postulated loss of coolant accidents of different sizes, locations and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. A review of the calculation disparities caused the NRC staff to question whether the AREVA break spectrum, provided the requisite assurance that the most severe postulated LOCA had been calculated.

While at 105 percent OLTP, the degraded ADS analyses had indicated reasonable agreement among the various break spectra, the analyzed restoration of the ADS appeared to re-introduce this discrepancy. Therefore, the NRC staff reviewed this discrepancy to determine its apparent causes and safety and regulatory significance. The most readily identifiable difference between the NRC staff and GE analyses, which were consistent, and the AREVA analysis, which was discrepant, was a difference in the limiting break size. AREVA had been predicting a limiting break size of 0.5 ft^2, and currently predicts a limiting break size of 0.21 ft^2, whereas the NRC staff and GE predicted much smaller limiting break sizes. It was determined that a significant difference in the AREVA model **[[**

]] .

In a consequence-driven event, the main steam lines do not isolate until the level one setpoint, which is also the setpoint for the reactor trip, is reached. Until this time, the pressure regulator attempts to control reactor pressure by throttling the turbine steam admission valve closed gradually, allowing coolant to continue leaving through the main steam lines and reducing the overall reactor vessel inventory.

This modeling difference appears to cause the various evaluation models to differ in their predictions of the smaller break sizes. The NRC staff questioned whether the absence of pressure control modeling would cause the model to predict a different break spectrum, and whether a comparison among various break sizes would illustrate that the AREVA method was conservative – predicted a higher PCT – when analyzing a given break size.

The licensee provided the results of a sensitivity study demonstrating the effect of the EXEM-BWR 2000 pressure control assumptions on the break spectrum. The results for the limiting break size as determined using the modified EXEM-BWR 2000 analyses (0.21 ft^2) and for a smaller break size that would result in a delayed pressurization following a level-driven main steamline isolation were included. The analyses compared the effect that a [[ ]] would have on the results at the current licensed thermal power level (CLTP) for a 0.21 ft^2 break and a 0.05 ft^2 break. The results indicated that, for the [[ ]] cases, the peak cladding temperature was slightly lower. As expected, the [[ ]] assumption resulted in a reduction in the vessel mass and exacerbated the heatup, the analytic results also showed that, when pressure control was assumed, the ADS system actuated earlier, which reduced the vessel pressure and permitted CS cut-in sooner.

The heatup trend for the 0.05 ft^2 break was roughly comparable to the trend predicted for the same single failure assumptions using the GE analytic methods for the [[ ]] break, although the GE analysis predicted [[ ]] peak cladding temperatures. It should be noted, that the licensee has not used GE methods to evaluate the full break spectrum up to [[ ]] and larger small breaks, so that a methodical comparison is not possible. As the analysis was performed at CLTP, the analytic differences when evaluating the AREVA methods do not extend to extended power uprate conditions.

Based on the similarity in event sequence timing and the performance of the cladding temperature trends for similar cases, and on the licensee's analytically supported conclusion that assume [[ ]] peak cladding temperature, the NRC staff finds that the assumption that [[ ]] at the initiation of the break will delay ADS actuation and exacerbate the transient. This is an acceptably conservative modeling assumption for use at the CLTP level.

Modified ECCS Evaluation Results Review

The NRC staff questioned the following observations associated with the modified ECCS evaluation review:

(1) The break spectrum for mid-peaked power shaped, split discharge breaks appeared to have abrupt temperature changes when plotted as a function of break size; and

(2) For the subset of break sizes from [[ ]] a smaller, intermediate temperature transient was observed to occur between the [[ ]].

Abrupt Temperature Changes With Break Size

As shown below, the predicted PCT versus break size for the mid-peaked pump discharge split breaks on the recirculation discharge line was plotted using Table B.1 of Enclosure 2 to Reference 23. The plot indicated a drop in PCT of approximately 200°F between two small breaks. This drop appeared, based on the remaining break spectrum results, to be rather abrupt. The NRC staff requested that the licensee explain why the break spectrum results exhibit slightly discontinuous behavior in the range of break sizes from [[ ]].

[[

]]

The licensee evaluated the [[ ]] events, which exhibited a [[ ]] drop in PCTs, to determine the cause of the difference in the predicted behavior. In Enclosure 2 to Reference 23 the licensee determined that a difference in liquid in the bottom of the average core was caused by differences in steam generation and counter current flow limitation (CCFL) at the assembly inlet. The differences were being caused by an [[

]]. As the licensee's analysis confirms that the abrupt change is due to required and acceptable modeling features, and not due to numerical issues with the code, the NRC staff finds the differences acceptable.

Intermediate Temperature Transient

As discussed previously, a review of Figure 2.5-3 of Enclosure 2 to Reference 23 revealed an intermediate temperature transient. The intermediate temperature transient is best described as

follows. [[

]].

The licensee evaluated the results to determine the cause of the different heatup behaviors. As discussed in Enclosure to Reference 23, the licensee concluded that the heatup behavior was attributable to a difference in the calculational technique used to determine the temperature trend at the given time in the calculation. [[

]]. Given that the final temperature peak is based on the Appendix K coefficients, the licensee's response confirms that the calculated results are obtained using acceptable model features specified in Appendix K to 10 CFR Part 50.

3.8.1.3.2 Results and Conclusions

The limiting case results are as follows:

Parameter	Value	10 CFR 50.46(b) Acceptance Criterion
PCT	1891°F	2200°F
Local Cladding Oxidation	1.15 percent	17 percent
Core-Wide Oxidation	0.66 percent	1.0 percent

Based on the review described above, the NRC staff determined that the proposed implementation of EXEM BWR-2000 for Unit 1, as supported by the modified analysis described in ANP-3015(P), is acceptable. This determination is based on the modified analysis conforming to the required and acceptable features of ECCS evaluation models described in Appendix K to 10 CFR Part 50, consistent with the requirement at 10 CFR 50.46(a)(1)(ii). The NRC staff further finds that the licensee's analytic results demonstrate conformance to the acceptance criteria set forth in 10 CFR 50.46(b). Based on these considerations, the NRC staff finds the proposed ECCS evaluation model implementation acceptable.

3.8.2 Safety Limit Minimum Critical Power Ratio Review (SLMCPR)

3.8.2.1 SLMCPR Methodology

The SLMCPR is imposed to protect at least 99.9 percent of the fuel rods in the core from boiling transition during steady state and transient conditions. In a letter dated August 8, 1990, the NRC approved topical report ANF-524(P)(A), *ANF Critical Power Methodology for Boiling Water Reactors,* Revision 2, November 1990 (ADAMS Accession No. ML031290327). This topical report identifies the fuel and non-fuel related uncertainties and the statistical process used to determine a MCPR safety limit. In the submittal, the licensee demonstrated compliance to each restriction in the NRC staff's approving SE.

3.8.2.2 SLMCPR

The SLMCPR analysis uses the SPCB critical power correlation additive constants and additive constant uncertainty for ATRIUM-10 fuel. The SPCB correlation is designed for application in steady-state, transient, and LOCA critical heat flux (CHF) calculations for the ATRIUM-10 fuel design. The SPCB additive constants and additive constant uncertainty for the coresident GE14 fuel were developed using the indirect approach described in EMF-2245(P)(A), *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*, Siemens Power Corporation, August 2000, Revision 0 (ADAMS Accession No. ML031290332). The additive constants for ATRIUM-10 and GE14 fuel are [[ ]] and [[ ]], respectively. The SLMCPR analysis addresses the mixed core effects and is performed each cycle using the cycle-specific core configuration. Each fuel type present in the core is modeled using appropriate geometric data, thermal hydraulic characteristics, and power distribution information (from CASMO-4 and MICROBURN-B2 analyses). The CPR is evaluated for each assembly using fuel type specific SPCB correlation coefficients. Plant and fuel type specific uncertainties are considered in the statistical analysis performed to determine the SLMCPR.

The NRC staff evaluated the impact of the proposed introduction of ATRIUM-10 fuel on the power distribution uncertainties used in the analyses. The licensee determined the SLMCPR by including the effects of channel bow using the following assumptions: no fuel channels used for more than one fuel bundle lifetime, and assembly average burnup remains less than 55 GWd/MTU for central ATRIUM-10 and GE14 fuel types. The channel bow local peaking uncertainty is a function of the nominal and bowed local peaking factors and the standard deviation of the channel bow.

The radial power uncertainty used in the analysis includes the effects of up to 40 percent of the TIP channels out-of-service, up to 50 percent of the LPRMs out-of-service, and a 2500 effective full power hours LPRM calibration interval. The assembly radial peaking for two-loop operation is [[ ]] and for single loop operations is [[ ]]. The analysis for two and single loop operation yields SLMCPR and percentage of rods in boiling transition values of 1.09, 0.086, 1.11, and 0.069, respectively.

As discussed in further detail in this SE, the NRC staff finds that the appropriate values of the MICROBURN-B2 uncertainties are used for the BFN SLMCPR, Therefore, the staff finds the licensee's application of the SLMCPR methodology, as documented in ANF-524(P)(A) in support of the fuel transition request, acceptable.

3.8.2.3 SLMCPR Conclusion

The NRC staff finds that the licensee appropriately applied an NRC-approved method to ensure adherence to the SLMCPR. The methodology used should ensure that 99.9 percent of the fuel rods in the core avoid transition boiling, consistent with GDC-10 regarding specified acceptable FDLs.

3.9 Technical Specification Changes

In order to enable TVA to use this fuel in Unit1, several changes to the TSs were proposed. An administrative correction to the header and section number is proposed for TS 3.2.3. In a letter

dated April 18, 2012, the licensee determined that proposed administrative change was not necessary, therefore that portion of the request was rescinded.

Two changes were identified to support the way the AREVA methodologies added in section TS 5.6.5 monitor and enforce thermal limits. As those methodologies are acceptable for use, the NRC staff approves the modification of the Unit 1 TS 3.3.4.1, *End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation*, and TS 3.7.5, *Main Turbine Bypass System* to require an LHGR adjustment when operating with EOC-RPT out of service and operating with turbine bypass out of service, respectively.

Changes were proposed to TS 3.3.1.1, *Reactor Protection Systems Instrumentation*, and TS 5.6.5.a to include the OPRM period based detection algorithm setpoint limits in the COLR. Since plant operation continues to be limited in accordance with the values of cycle-specific parameter limits established by NRC-approved methodologies, the NRC staff concludes that the proposed TS changes have no adverse impact on plant safety. Therefore, the NRC staff concludes that the proposed changes are acceptable.

A list of the analytical methods which are used to determine input to the COLR are contained in TS 5.6.5.b. Currently, the TS 5.6.5.b contains only GNF analytical methods that pertain to GE14 fuel design. In support of the transition to ATRIUM-10 fuel, a change to the TS 5.6.5.b to include appropriate NRC approved AREVA analytical methodologies that are listed in Attachments 2 and 3 to the submittal was proposed. The NRC staff notes that the original intent was to add only the titles of the analytical methods without specifying the revision number. In a letter dated August 14, 2011 (ADAMS Accession No. ML110660285), the NRC provided a discussion on the format for listing analytical methodologies in the TSs. Maintaining a list of the methodologies in the TSs requires licensees to obtain NRC approval prior to editing the reference list. Among others, one reason that NRC approval is required prior to editing the reference list is so that the NRC staff can review the methodology and ensure that it is applicable to the facility of a given licensee. Additionally, the NRC staff can verify that the licensee has properly satisfied all implementation conditions and limitations associated with a given methodology. Because there is no inherent requirement to ensure that the implementation conditions and limitations associated with methodology revisions are maintained the same as previous revisions to the same methodology, or that the applicability of subsequent methodology revisions remains the same as earlier methodologies, the NRC staff finds that the revision number for the listed methodologies is necessary. In a letter dated April 18, 2012, the licensee revised the proposed submitted TS pages to reflect the addition of the revision numbers.

Provided that the use of the listed analytical models in Attachments 2 and 3 to the submittal are applied consistent with the NRC approving safety evaluations and the model limitations and conditions, the NRC staff finds those methodologies acceptable for use. Therefore, as the additions are consistent with the guidance of GL 88-16 and the associated SE dated May 20, 1993, the NRC staff finds the addition of the analytical models in Attachments 2 and 3 to the submittal to TS 5.6.5.b acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding January 10, 2011 (76 FR 1467). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter from R. M. Krich (TVA) to U.S. NRC, "Technical Specification Change TS-473 – AREVA Fuel Transition," Tennessee Valley Authority, April 16, 2010.

2. Letter from R. M. Krich (TVA) to U.S. NRC, "Response to NRC Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel (TAC No. ME3775)," Tennessee Valley Authority, February 23, 2011.

3. ANP-2877P, Revision 0, "Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105% OLTP)" (Enclosure 6 to Reference 1), AREVA NP, Inc., November 2009.

4. ANF-89-98(P)(A), Revision 1, Supplement 1, "Generic Mechanical Design Criteria for BWR Designs," Advanced Nuclear Fuels Corporation, May 1995.

5. EMF-93-177(P)(A), Revision 1, "Mechanical Design for BWR Fuel Channels," Framatome ANP, Inc., August 2005.

6. XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.

7. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup, Supplement 1, Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988.

8. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, September 1986.

9. XN-NF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Exxon Nuclear Company, August 1986.

10. EMF-85-74(P), Revision 0, Supplement 1(P) (A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation, February 1998.

11. XN-NF-81-58(P)(A), Revision 2, Supplements 1 and 2, "RODEX2, Fuel Rod Thermal-Mechanical Evaluation Model, Exxon Nuclear Company, March 1994.

12. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

13. ANP-2821(P), Revision 0, "Browns Ferry Unit1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies (105% OLTP)" (Enclosure 8 to Reference 1), AREVA NP Inc., June 2009.

14. XN-NF-79-59(P)(A), "Methodology for Calculation of Pressure Drop in BWR fuel Assemblies," Exxon Nuclear Company, November 1983.

15. EMF-2209(P)(A), Revision 2, "SPCB Critical Power Correlation," Framatome ANP, September 2003.

16. EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, August 2000.

17. NEDO-32465-A, Licensing Topical Report, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," BWROG, GE Nuclear Energy, August 1986.

18. 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.

19. ANP-2859(P), Revision 0, "Browns Ferry Unit 1 Cycle 9 Fuel Cycle Design (105 [percent] OLTP), AREVA NP Inc.," September 2009.

20. GNF 0000-0111-8036-R0-P (Enclosure 14 to Reference 1), "GE14 Fuel Thermal-Mechanical Information," Global Nuclear Fuel, January 2010.

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

22. Letter from F. Hebdon (U.S. NRC) to M. Medford (TVA), "Issuance of Amendments Regarding the Core Operating Limits Report," May 20, 1993.

23. Letter from R. M. Krich (TVA) to U.S. NRC, "Browns Ferry Unit 1 - Response to NRC Request for Additional Information Regarding a Request to Transition to AREVA Fuel (TAC No. ME3775)," Tennessee Valley Authority, May 12, 2011.

24. Letter from R. M. Krich (TVA) to U.S. NRC, "Browns Ferry Unit 1 - Response to NRC Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel (TAC No. ME3775)," Tennessee Valley Authority, October 7, 2011.

25. ANP-2863(P), Revision 1, "Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105 [percent] OLTP," March 2010.

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Date: April 27, 2012

It should be noted that this is a unit-specific review and the conclusions are not applicable to other units or facilities, or at extended operating conditions. A copy of the Safety Evaluation is also enclosed. A proprietary version of this letter and safety evaluation was issued in a letter to the TVA dated April 27, 2012. This letter transmits a non-proprietary version. **Proprietary information was removed, as indicated by empty bold-face double-brackets, such as [[ ]].** A Notice of Issuance was included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Eva A. Brown, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-259

Enclosures:
1. Amendment No. 281 to DPR-33
2. Safety Evaluation

cc w/ encls: Distribution via Listserv

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