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

February 15, 2013

Mr. Joseph W. Shea
Corporate Manager, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3D-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3 – ISSUANCE OF AMENDMENTS REGARDING DELETION OF LOW PRESSURE COOLANT INJECTION MOTOR-GENERATOR SETS FOR BROWNS FERRY PLANT, UNITS 2 AND 3 (TAC NOS. ME9176 AND ME9177)

Dear Mr. Shea:

The Commission has issued the enclosed Amendment Nos. 309 and 268 to Renewed Facility Operating Licenses (FOL) Nos. DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant (BFN), Units 2 and 3, respectively. These amendments are in response to your application dated February 25, 2011, as supplemented by letters dated September 15, 2011, July 30, 2012, and January 24, 2013. The enclosure to the July 30, 2012, letter superseded, in its entirely, the enclosure to the February 25, 2011, letter.

The amendments delete the BFN, Units 2 and 3, Technical Specification (TS) Surveillance Requirement 3.5.1.12, which requires the verification of the capability to automatically transfer the power supply from the normal source to the alternate source for each Low-Pressure Coolant Injection subsystem inboard injection valve and each recirculation pump discharge valve on a 24-month frequency. In addition, these amendments approve the use of a modified loss-of-coolant accident (LOCA) methodology that requires revising TS 5.6.5b to include a reference to the modified LOCA methodology. Also, the amendments revise TSs 3.3.1.1, 5.6.5a, and 5.6.5b to include the modified LOCA methodology and the oscillation power range monitor upscale function period based detection algorithm setpoint limits.

Further, the amendment for BFN, Unit 3 approves changes to the FOL to add the licensee's proposed license condition regarding implementation of the amendment at BFN, Unit 3.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Nuclear Regulatory Commission's biweekly *Federal Register* notice.

Sincerely,

Siva P. Lingam for

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296

Enclosures:
1. Amendment No. 309 to
   License No. DPR-52
2. Amendment No. 268 to
   License No. DPR-68
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 309
Renewed License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated February 25, 2011, as supplemented by letters dated September 15, 2011, July 30, 2012, and January 24, 2013, 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-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 309 , 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 entering Mode 3 (i.e., Hot Shutdown) from the spring 2013 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Jessie F. Quichocho, 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: February 15, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 309

RENEWED FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace Page 3 of Renewed Operating License DPR-52 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
3.3-8	3.3-8
3.5-7	3.5-7
5.0-24	5.0-24
5.0-24a	5.0-24a
---	5.0-24b

sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 309, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 253 to Facility Operating License DPR-52, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 253. For SRs that existed prior to Amendment 253, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 253.

3) The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's

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.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.5.1.10	-------------------------NOTE------------------------ Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.11	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each ADS valve opens when manually actuated.	24 months
SR 3.5.1.12	(Deleted)	

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. 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.

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

3. 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.

(continued)

5.6 Reporting Requirements (continued)

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

5. 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.

6. 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.

7. 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.

8. 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.

9. 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.

10. 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.

11. 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.

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

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

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5.6 Reporting Requirements (continued)

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

15. 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 February 15, 2013.

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

17. 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.

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

(continued)

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 268
Renewed License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated February 25, 2011, as supplemented by letters dated September 15, 2011, July 30, 2012, and January 24, 2013, 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 Operating license and Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 268 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. For License Amendment 268 , the licensee shall implement changes to BFN, Unit 3 TSs 5.6.5 and 3.3.1.1 within 60 days of approval. The remaining BFN, Unit 3, changes will be implemented upon completion of required supporting modification work and prior to entering Mode 3 (i.e., Hot Shutdown) from the spring 2014 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Jessie F. Quichocho, 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: February 15, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 268

RENEWED FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace the following pages of Renewed Operating License DPR-68 with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3	3
6	6

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
3.3-8	3.3-8
3.5-7	3.5-7
5.0-24	5.0-24
5.0-24a	5.0-24a
---	5.0-24b

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3458 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 268 , are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 212 to Facility Operating License DPR-68, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 212. For SRs that existed prior to Amendment 212, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 212.

Following Implementation:

(a) The *first* performance of SR 3.7.4.4, in accordance with TS 5.5.13.c.(i), shall be within a specific frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from November 10, 2003, the date of the most recent successful tracer gas test.

(b) The first performance of the periodic assessment of the Control Room Envelope CRE) Habitability, Technical Specification 5.5.13.c.(ii), shall be with 9 months following the initial implementation of the TS Change. The next performance of the periodic assessment will be in a period specified by the CRE Program. That is 3 years from the last successful performance of the Technical Specification 5.5.13.c.(ii) tracer gas test.

(c) The first performance of the periodic measurement of CRE pressure, TS 5.5.13.d, shall be within 24 months, plus 180 days allowed by SR 3.0.2 as measured from the date of the most recent successful pressure measurement test.

(d) For License Amendment 268 , the licensee shall implement changes to BFN, Unit 3 TSs 5.6.5 and 3.3.1.1 within 60 days of approval. The remaining BFN, Unit 3, changes will be implemented upon completion of required supporting modification work and prior to entering Mode 3 (i.e., Hot Shutdown) from the spring 2014 refueling outage.

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71 (e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than July 2,2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

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.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.10	-------------------------NOTE------------------------ Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.11	-------------------------NOTE------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each ADS valve opens when manually actuated.	24 months
SR 3.5.1.12	(Deleted)	

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. 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.

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

3. 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.

(continued)

5.6 Reporting Requirements (continued)

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

5. 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.

6. 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.

7. 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.

8. 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.

9. 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.

10. 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.

11. 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.

(continued)

5.6 Reporting Requirements (continued)

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

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

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

15. 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 February 15, 2013.

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

17. 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.

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

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 309 TO RENEWED FACILITY OPERATING

LICENSE NO. DPR-52 AND AMENDMENT NO. 268 TO RENEWED

FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 2 AND 3

DOCKET NOS. 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated February 25, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110600408), the Tennessee Valley Authority (TVA, the licensee) requested an amendment to Technical Specifications (TS) for the Browns Ferry Nuclear Plant (BFN), Units 2 and 3 (Renewed Facility Operating License Nos. DPR-52 and DPR-68). The proposed license amendment request (LAR) would delete BFN, Units 2 and 3, TS Surveillance Requirement (SR) 3.5.1.12, which currently requires the verification of the capability to automatically transfer the power supply from the normal source to the alternate source for each low pressure coolant injection (LPCI) subsystem inboard injection valve and each recirculation pump discharge valve on a 24-month frequency. Based on improved loss-of-coolant accident (LOCA) analysis, the automatic transfer of power is no longer required and will be deleted. Due to the proposed change, LPCI motor-generator (MG) Sets associated with the automatic scheme would also be deleted.

In response to Nuclear Regulatory Commission (Commission, NRC) staff's request for additional information (RAI), the licensee provided supplemental information in a letter dated September 15, 2011 (ADAMS Accession No. ML11263A159). In its supplemental letter dated July 30, 2012 (ADAMS Accession No. ML12215A005), the licensee provided a reference to the more recent LOCA analysis (modified EXEM BWR [Boiling-Water Reactor]-2000 Evaluation Model) approved by NRC on April 27, 2012, for use on BFN, Unit 1. In the letter dated July 30, 2012, the licensee also requested approval for the use of a modified LOCA methodology that requires revising TS 5.6.5b to include a reference to the modified LOCA methodology. Also, the licensee requested to revise TSs 3.3.1.1, 5.6.5a, and 5.6.5b to include the modified LOCA methodology and the oscillation power range monitor upscale function period based detection algorithm setpoint limits. The enclosure to the July 30, 2012, letter superseded, in its entirely, the enclosure to the February 25, 2011, letter. Further, by letter dated January 24, 2013, TVA provided clarification regarding deleting current Reference 1 from TS 5.6.5b.

Enclosure 3

The application dated February 25, 201, was originally published on May 3, 2011 in *Federal Register* (76 FR 24930). However, the supplement dated July 30, 2012, did expand the scope of the application as originally noticed, therefore, a revised hazards consideration determination was published in the *Federal Register* (FR) on November 5, 2012 (77 FR 66490). The supplement dated January 24, 2013, provided additional information that clarified the licensee's July 30, 2012, submittal, did not expand the scope of the application as noticed and did not change the NRC staff's proposed no significant hazards consideration determination as published in the FR on November 5, 2012 (77 FR 66490).

2.0 REGULATORY EVALUATION

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) SRs; 4) design features; and 5) administrative controls.

The LPCI system is among the emergency core cooling systems (ECCSs) provided to mitigate postulated LOCAs at BFN, Units 2 and 3. The systems are provided and analyzed in accordance with NRC requirements, which are promulgated at 10 CFR 50.46. In addition to referencing the required and acceptable features of ECCS evaluation models (10 CFR 50.46(a)(1)(i) and (ii)), and providing the analytic acceptance criteria for such evaluations (10 CFR 50.46(b)), 10 CFR 50.46(d) states as follows:

> The requirements of this section are in addition to any other requirements applicable to ECCS set forth in [Part 50]. The criteria set forth in [10 CFR 50.46(b)], with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance and design set forth in [Part 50], including in particular Criterion 35 of Appendix A.

The construction permits for BFN, Units 2 and 3 were issued by the Atomic Energy Commission (AEC) on May 10, 1967, and July 31, 1968, respectively. BFN, Units 2 and 3 were designed and constructed based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (AEC draft GDC). The AEC published the final rule that added Appendix A to 10 CFR Part 50, General Design Criteria for Nuclear Power Plants, in the *Federal Register* (36 FR 3255) on February 20, 1971 (Appendix A GDC).

Appendix A, GDC12, "Suppression of reactor power oscillations," states that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Appendix A, GDC 17, "Electric power systems," requires, in part, that nuclear power plants have onsite and offsite electric power systems to permit the functioning of structures, systems, and components that are important safety. The onsite system is required to have sufficient independence, redundancy, and testability to perform its safety function, assuming a single failure.

Appendix A, GDC 35, "Emergency core cooling," states:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be 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.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Although BFN was licensed before the promulgation of Appendix A GDCs, and Commission policy precludes the NRC from imposing them upon BFN, the TVA committed to meet the intent of the GDCs as documented in the BFN Updated Final Safety Analysis Report, Appendix A. The design criteria in the BFN licensing basis that are analogous to GDC 17, 18, and 35, are AEC draft GDC 7, 39, and 44.

AEC draft GDC 7, "Suppression of Power Oscillations (Category B)," states 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 draft GDC 39, "Emergency Power for Engineered Safety Features (ESF)," states that alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the ESF. As a minimum, the onsite power system and offsite power system shall each, independently provide this capacity assuming a failure of a single active component in each power system.

AEC draft GDC 44 states, in part, that at least two ECCSs, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each ECCS and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reactor to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each ECCS shall be evaluated conservatively in each area of uncertainty.

Finally, the guidance in NRC Generic Letter (GL) 1988-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," applies, because TVA requested a revision to the core

operating limits report (COLR) references list in the TS to reflect the implementation of an ECCS evaluation that justifies the proposed SR deletion.

3.0 TECHNICAL EVALUATION

3.1 Description of Affected Systems

3.1.1 Emergency Core Cooling System

BFN, Units 2 and 3 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 (SBLOCA), 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 SBLOCAs. 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 SBLOCA.

The CS system is designed to provide top-down spray cooling in the core to mitigate the effects of large break LOCAs, and for SBLOCAs 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 BFN is comprised of two separate divisions, each providing emergency core coolant injection to a single recirculation loop. 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.

3.1.2 480 Volt (V) Electrical Distribution System

At BFN, the safety-related power is distributed by eight 4160V Shutdown Boards (four for Units 1 and 2, and four for Unit 3). Each board is powered from a normal and an alternate offsite power source, and also backed-up by an emergency diesel generator. The 4160V Shutdown Boards feed the 4 kilovolt safety-related pump loads, and the downstream 480V safety-related Shutdown Boards through 4160-480V transformers.

As shown in Figures 1 and 3 of the licensee's supplemental letter dated July 30, 2012, for the current configuration of the Electrical Distribution System, which is pertinent to the proposed change for Units 2 and 3, there are five 480V Reactor Motor Operated Valve (RMOV) Boards (A through E) powered by 480V safety-related Shutdown Boards A and B for Unit 2, and five similar RMOV Boards for Unit 3. The 480V RMOV Boards A and D are normally powered from 480V Shutdown Board A (Division I power), with the alternate power supply source from 480V Shutdown Board B. The 480V RMOV Boards B, C, and E are normally powered from 480V Shutdown Board B (Division II power), with the alternate power supply source from 480V Shutdown Board A.

Currently, 480V RMOV Boards D and E are supplied from 480V Shutdown Boards A and B via MG sets. There are four MG sets in Unit 2, and four in Unit 3. Two MG sets are fed from 480V Shutdown Board A and act as a normal power source for 480V RMOV Board D and as an alternate power source to 480V RMOV Board E. Two MG sets are fed from 480V Shutdown Board B and act as a normal power source for 480V RMOV Board E and as an alternate power source to 480V RMOV Board D.

Currently, 480V RMOV Boards D and E automatically transfer the power supply from the normal source to the alternate source upon detection of an under voltage condition from the normal source. The MG sets act as electrical isolators to prevent a fault from propagating between electrical divisions during an automatic transfer.

The 480V RMOV Board D provides Division I power to the following loads:

- Flow Control Valve (FCV) 68-79, Recirculation Pump Discharge Valve;
- FCV-74-7, RHR Pumps A and C Minimum Flow Bypass Valve; and
- FCV-74-53, RHR LPCI Injection Valve.

The 480V RMOV Board E provides Division II power to the following loads:

- FCV-68-3, Recirculation Pump Discharge Valve;
- FCV-74-30, RHR Pumps B & D Minimum Flow Bypass Valve; and
- FCV-74-67, RHR LPCI Injection Valve.

The automatic transfer capability for 480V RMOV Boards D and E was designed to ensure that the LPCI injection occurred from both loops with at least one pump in each loop. If one loop's LPCI injection valve (either FCV-74-53 or FCV-74-67), RHR minimum flow valves (FCV-74-7 and 30) and the associated reactor recirculation loop discharge valve (either FCV-68-79 or FCV-68-3) lost power (from either 480V RMOV Boards D or E), the RMOV board would automatically transfer to the opposite division's power supply to ensure operation of the valves. With this

transfer scheme in place, the concern was that the automatic transfer could propagate an electrical fault to both divisions of power supply. As a result, Unit 2 and Unit 3 LPCI MG sets were included in the design for both the normal and alternate power supplies to the RMOV Boards to provide electrical isolation between the two divisions.

3.2 Evaluation of Changes to LPCI MG Sets

3.2.1 Description of Proposed Changes

In the LAR, the licensee stated that the automatic transfer of the power supply for the LPCI inboard injection valves, RHR minimum flow valves, and recirculation pump discharge valves was once a requirement to comply with 10 CFR Part 50 Appendix K, "ECCS Evaluation Models," and 10 CFR 50.46, using older LOCA analysis methods. However, based on improved LOCA analysis methods, the current ECCS performance analysis does not take credit for the LPCI automatic transfer mechanism. The 10 CFR 50.46 regulatory requirements are met by the two independent electrical power divisions for the ECCS equipment without the automatic transfer of the power supply for the LPCI inboard injection valves, RHR minimum flow valves, and recirculation pump discharge valves (fed from RMOV Boards D and E).

The licensee has proposed to delete TS SR 3.5.1.12, "Verify automatic transfer of the power supply from the normal source to the alternate source for each LPCI subsystem inboard injection valve and each recirculation pump discharge valve" every 24 months, for Units 2 and 3. This change will allow the removal of the LPCI MG sets, which were once a requirement for electrical divisional isolation between the Class 1E normal and alternate power feeds to RMOV Boards D and E. The licensee proposed to connect RMOV Boards D and E directly to their normal power supplies, and keep both the alternate supply breakers to these RMOV Boards normally open to provide isolation between Electrical Divisions I and II.

3.2.2 Evaluation of the proposed TS SR 3.5.1.12 Changes

The NRC staff evaluated the impact of the proposed deletion of the TS SR 3.5.1.12 on the Electrical Distribution System and ECCS performance of BFN, Units 2 and 3. The impact of proposed change on the ECCS performance is evaluated in Section 3.3 of this safety evaluation (SE).

The primary purpose of the existing LPCI MG sets is to provide isolation between Division I and Division II power supplies to the RMOV Boards in case of an automatic transfer due to loss of the normal source of power to RMOV Boards D and E. This design ensured that propagation of an electrical fault from one division to the other did not occur. Since automatic transfer of power supplies to the RMOV Boards is no longer credited in the revised ECCS performance analysis, LPCI MG sets are not required for isolation purpose. The licensee has proposed to power RMOV Boards D and E directly from the corresponding 480V Shutdown Boards through normal feeder breakers. All feeder breakers for 480V for RMOV Boards D and E will be modified from electrically operated to mechanically operated, at the applicable 480V Shutdown Boards. The alternate feeder breakers for 480V for RMOV Boards D and E will be changed from normally closed to normally open.

The NRC staff identified a concern about maintaining the isolation between the two divisions of the 480V RMOV boards, after the proposed change. In response to an NRC staff RAI, the licensee, by letter dated September 15, 2011, provided the following clarification:

- The proposed modification to the breakers at the shutdown boards, feeding the RMOV Boards D and E, is for isolation purposes.

- The similar 480V Shutdown Board feeder breakers for Units 2 and 3, 480V RMOV Boards A, B, and C are of a mechanically-operated type for isolation purposes.

- After the breakers are modified to a mechanically operated breaker type, the breakers can be operated locally at the 480V shutdown or the RMOV boards. The operating procedures will be revised to have the operators operate these breakers locally and to address the deletion of the MG sets and their power-seeking auto-transfer function.

Based on the above clarification, the NRC staff concern is resolved because double isolation will be maintained between the two divisions of 480V RMOV boards by keeping both of the alternate source breakers to these boards open, to avoid propagation of an electrical fault from one division to the other.

The NRC staff finds the proposed change to the electrical distribution system, and the proposed deletion of the TS SR 3.5.1.12, acceptable.

3.3 Emergency Core Cooling System Evaluation

3.3.1 Description of Proposed Changes

In the July 30, 2012, supplement, the licensee requested approval to apply a modified version of the EXEM BWR-2000 ECCS evaluation model to BFN, Units 2 and 3. The July 30, 2012, submittal stated that issues with the application of the ECCS evaluation model had been identified, and were addressed in the modified evaluation. This evaluation is documented in AREVA report ANP-3015 (NP), "Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis" (ADAMS Accession Nos. ML12215A006). This document, which is applicable to BFN, Units 1, 2, and 3, was approved for implementation at BFN, Unit 1 (Amendment No. 281 dated July 3, 2012). This is a plant-specific application of the NRC-approved AREVA ECCS evaluation model documented in EMF-2361, "EXEM BWR-2000 ECCS Evaluation Model," May 2001 (ADAMS Accession No. ML012050396). The specific issues and the licensee's proposed resolution are evaluated by the NRC staff in Section 3.3.2, below.

3.3.2 Evaluation of changes to ECCS Model

The NRC staff technical review and conclusions finding ANP-3015 acceptable for BFN, Unit 1 are discussed in the SE for Amendment No. 281 (ADAMS Accession No. ML12129A149). As discussed in the following sections, based on the similarity of the three BFN units, and because all three units now use AREVA fuel and analytic methods, the NRC staff conclusions as to acceptability of BFN, Unit 1 also apply to Units 2 and 3. Section 3.8.1 of the SE for BFN, Unit 1, Amendment 281, also discusses that the evaluation documented in ANP-3015 acceptably demonstrates compliance with the acceptance criteria contained in 10 CFR 50.46(b).

3.3.2.1 LOCA Analyses Review

The licensee submitted an ECCS evaluation that was performed using a modified version of the AREVA EXEM BWR-2000 evaluation model. The modified evaluation used certain calculations for the heat transfer characteristics that EXEM BWR-2000 did not appear capable of representing accurately or conservatively. ANP-3015 identifies the unacceptable calculations and modifies the ECCS evaluation model to address the calculational issues. The following summarizes the NRC staff review of the licensee's ECCS evaluation as modified to support Units 2 and 3.

EXEM BWR-2000 ECCS Evaluation Model Validation

The EXEM BWR-2000 ECCS evaluation model 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 and one small break simulations. 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 HPCI and low pressure spays and injection. The small break was simulated on a BWR/6-type facility.

Given the limiting nature of the SBLOCA at BFN – a somewhat unique result among BWRs – the NRC staff reviewed the RELAX/HUXY assessment against the 6SB1 test, in particular for the BFN review. 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 [evaluation model] is reasonable for small breaks. Based on AREVA's claim, the NRC staff determined that the EXEM BWR-2000 ECCS evaluation model successfully simulated the 6SB1 test with reasonable agreement. The SE approving EXEM BWR-2000 documents this determination.

In a review of a power uprate application submitted by another facility, the NRC staff identified issues with the approach to modeling counter-current spray cooling that is used in EXEM BWR-2000. In this prior review, the NRC staff determined that there were differences between its

confirmatory calculations and AREVA's for certain, nonlimiting small breaks. The NRC staff confirmatory peak cladding temperature (PCT) predictions resulted in higher temperature predictions than AREVA's. This was attributed to the counter-current flow limitation correlations employed by AREVA that the NRC staff determined were not applicable for certain modeling configurations. However, the submitted uprate application was approved because the effects of counter-current spray cooling were nonlimiting for that facility. Since the limiting PCT case for BFN previously appeared to rely on spray cooling, and the limiting accident for BFN is an SBLOCA, the same conclusion could not be drawn for the BFN ECCS evaluation.

The Sector Steam Test Facility 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 Units 2 and 3 limiting small break, as calculated using the unmodified version of EXEM BWR-2000, which was referenced in the February 25, 2011, license amendment request, did not illustrate this trend.

As described above, the ECCS evaluation model has been modified as documented in ANP-3015(NP). The modification results in the restriction of counter-current liquid flow from entering the hot bundle. Because the modification addresses the concern with counter-current liquid flow in the BFN small-break limiting configuration, and because the evaluation model is NRC-approved and otherwise acceptable, the NRC staff concludes that the modified evaluation model is acceptable for application to BFN, Units 2 and 3.

BFN ECCS Analysis

AREVA report ANP-3015(NP), Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis, Revision 0 (ADAMS Accession No. ML11286A109), provides a revised ECCS Evaluation Summary to address the issues with the counter-current flow limitation model. This revised summary also provided a description of other break sizes, locations and other properties. The NRC staff reviewed the analysis to determine whether the licensee had calculated the most severe postulated LOCA.

The analytic results documented in ANP-3015(NP) indicate that the limiting recirculation line break was a 0.21 ft^2 split break in the pump discharge with a batter board 'A' failure and a top-preaked axial power shape. The limiting PCT was 1891 °F. Local cladding oxidation was 1.15 percent, and core-wide oxidation was 0.66 percent. These results conform to the acceptance criteria promulgated in 10 CFR 50.46(b), which require a PCT less than 2200 °F, local oxidation less than 17 percent, and less than 1 percent core wide oxidation. Based on the licensee's results and the use of an acceptable evaluation model as described above, the NRC staff determined that the proposed analysis documented in ANP-3015(NP) 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 of 10 CFR 50.46(a)(1)(ii). The NRC staff also determined that the licensee's analytic results conform to the 10 CFR 50.46(b) acceptance criteria.

In order to implement the ECCS evaluation documented at ANP-3015 for BFN, Units 2, and 3, TVA determined that it is necessary to revise the COLR References List contained in BFN, Units 2 and 3 TS 5.6.5b. Although this SE refers to both proprietary and non-proprietary versions of ANP-3015 and EMF 2361 (EXEM BWR-2000), the references listed in COLR uses the proprietary version of the EMF-2361. Therefore, reference to EMF 2361(P)(A) on TS page 5.0-24b would state:

> 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 February 15, 2013.

This proposed change reflects the guidance contained in GL 1988-16, which indicates that plant-specific analytic methods should include a reference to the NRC staff safety evaluation that approves the use of such methods. On this basis, the NRC staff finds the proposed TS revision acceptable.

3.3.3 Evaluation of Removal of SR 3.5.1.2 on ECCS

A SR similar to SR 3.5.1.12 in BFN, Units 2 and 3 TSs previously existed for BFN, Unit 1, but had been removed by Amendment 254, issued by letter dated June 20, 2005 (ADAMS Accession No. ML051580047). The SE included with this letter contains the NRC staff technical review and conclusions finding the SR deletion acceptable based on an evaluation similar to that in Section 3.2.2 of this SE. Because the modification proposed by TVA, and the supporting TS revisions, will make BFN, Units 2 and 3 similar to BFN, Unit 1, the NRC staff concludes that the conclusions presented in Amendment 254 apply to BFN, Units 2 and 3. As discussed below, the proposed SR deletion is consistent with the applicable design criteria discussed in Section 2.0 of this SE and the ECCS evaluation demonstrates acceptable performance relative to the 10 CFR 50.46(b) acceptance criteria.

With the deletion of Boards D and E, the automatic transfer capability for the power supply to the LPCI injection valves and recirculation pump discharge valves will also be deleted. The RMOV Boards A and B (to which the loads are transferred) do not have automatic transfer capability. The impact on the number of the Emergency Core Cooling Subsystems available under various contingencies due to the loss of automatic transfer capability is evaluated in Section 3.3.1 of this SE.

The emergency core cooling subsystems are designed to limit peak clad temperature over the complete spectrum of possible break sizes, including the design basis break. The design basis break is defined in ANP-3015(NP) and evaluated at the conclusion of Section 3.3.2.1 of this SE. The current analysis does not credit the availability of the automatic transfer capability.

The NRC staff reviewed the impact of the proposed changes on the capabilities of the ECCS at BFN, Units 2 and 3. Because the minimum equipment requirements set forth in GDC 17 and 35 are met and the requirements of 10 CFR 50.46 are satisfied, the NRC staff has determined that the proposed change is acceptable. The proposed SR elimination will allow TVA to make a modification to the LPCI system that would reduce, in the words of GDC 35, the redundancy of the LPCI system.

The NRC staff has reviewed the licensee's proposed TS changes and supplemental information. Based on the evaluation discussed above, the NRC staff finds the proposed deletion of TS SR 3.5.1.12, and deletion of LPCI MG sets acceptable. The proposed changes satisfy the regulatory requirements of 10 CFR 50.36, 10 CFR 50.46, and 10 CFR Part 50, Appendix A, GDC 17 and 35.

3.4 Technical Specification Changes

In addition to TS changes described in Sections 3.2 and 3.3 of this SE, TVA revised all the methodology references in TS 5.6.5 to include a revision number and a revision date. This change adds specificity to the TS COLR References, and is consistent with an NRC position documented in a letter from the NRC to the Technical Specifications Task Force (TSTF), dated August 4, 2011 (ML110660285). The TSTF is a nuclear power industry-sponsored group that submits proposed revisions to the Standard Technical Specifications (STSs) to the NRC for review, approval, and subsequent incorporation into the STSs (http://www.nrc.gov/reactors/operating/licensing/techspecs/post-revision3-sts.html). This letter acknowledges a withdrawal of a STS revision that permitted the elimination of revision numbers and dates from TS COLR references. The revision that TVA proposes, in which revision numbers and dates are added to the TS COLR References, is acceptable because it is more restrictive and it reflects the position documented in the August 4, 2011, letter.

A list of the analytical methods that are used to determine input to the COLR are contained in TS 5.6.5b. The NRC staff notes that the original intent was to add only the titles of the analytical methods without specifying the revision number. In the August 4, 2011, letter, 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. NRC approval is required for all TS changes, including editing the COLR reference list. The NRC staff reviews the methodology to ensure it is applicable to the facility of a given licensee. Also, the NRC staff reviews and evaluates whether the licensee has properly satisfied all implementation conditions and limitations associated with a given methodology. 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 determined that the revision number for the listed methodologies is necessary. In its letter dated July 30, 2012, the licensee revised the proposed TS pages to reflect the addition of the revision numbers.

The proposed TS requires that listed analytical models in Attachments 1 and 2 to the submittal are applied consistent with the NRC approving SEs and the model limitations and conditions and the NRC staff finds those methodologies acceptable for use. Because 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 1 and 2 to the submittal to TS 5.6.5b acceptable.

In addition, the licensee in its letter dated July 30, 2012, stated that as part of TS change TS-473, AREVA Fuel Transition for BFN, Unit 1, TVA committed to make the following changes to BFN, Units 2 and 3 TS 3.3.1.1, TS 5.6.5a and TS 5.6.5b to include the AREVA methodology for the oscillation power range monitor (OPRM) upscale function period based set point limits.

- Function 2f of Table 1 of TS 3.3.1.1, "Reactor Protection Systems Instrumentation," and TS 5.6.5. "Core Operating Limits Report (COLR)," will be revised to indicate that the OPRM Upscale Function period based detection algorithm setpoint limits are included in COLR.

- TS 5.6.6b will be revised to include the AREVA stability related Topical Reports that describe the analytical methods used for determining the OPRM period based detection algorithm setpoint limits.

The revisions to the OPRM-related TS are acceptable because they will incorporate the analytic methods used to evaluate reactor stability in the COLR. The TS will now specify which methods are used in order to ensure compliance with GDC 12. Also, the proposed revisions are acceptable because the stability-related topical reports are approved by NRC for use at boiling water reactors.

In addition, the licensee in its letter dated January 24, 2013, clarified removal of the current Item 1 of TS 5.6.5b. The licensee stated that General Electric fuel is no longer used in BFN, Units 2 and 3. Therefore, TVA requested deletion of the reference to NEDE-24011 for TS 5.6.5b because this methodology is no longer applicable to BFN, Units 2 and 3. The NRC staff finds the proposed deletion of NEDE-24011 acceptable because it is no longer appropriate for the fuel used and will preclude the licensee from using General Electric analytic methods to analyze AREVA fuel.

4.0 LICENSE CONDITION

TVA in its letter dated July 30, 2012, stated that the proposed amendment would be implemented by the next refueling outage for each unit. For BFN, Unit 2, it will be implemented prior to entering Mode 3 (Hot Shutdown) from the spring 2013 refueling outage. However, the licensee proposed to add a license condition to BFN, Unit 3 facility operating license to permit partial implementation of the proposed amendment. The licensee proposed to implement BFN, Unit 3 TS 5.6.5 and TS 3.3.1.1 within 60 days of approval. The licensee stated that these TS changes are needed to resolve a BFN, Unit 3 degraded/nonconforming condition involving the AREVA LOCA Analysis. TVA proposed to implement the remaining changes, deletion of SR 3.5.1.12 and required supporting modification work, prior to entering Mode 3 from the spring 2014 refueling outage.

The above BFN, Unit 3 degraded/nonconforming condition is documented in details under Unresolved Item 2011003-03, in the NRC Integrated Inspection Report 2012002, dated April 27, 2012 for BFN, Units 1, 2 and 3. The licensee, in Enclosure 2 of its letter dated April 18, 2012, in part, described this issue and stated that the AREVA's revised analysis (Section 3.3.2 of this SE) would be submitted to the NRC for application for BFN, Unit 3. The licensee stated that it would continue to treat this issue as a degraded/nonconforming condition until NRC approval of the revised analysis. The NRC staff finds the proposed license condition acceptable because it resolves the degraded/nonconforming condition associated with the LOCA analysis for BFN, Unit 3 in a timely manner. Therefore, the following license condition will be added to the Renewed Facility Operating License No. DPR 68 for BFN, Unit 3:

(d) For License Amendment 268 , the licensee shall implement changes to BFN, Unit 3 TSs 5.6.5 and 3.3.1.1 within 60 days of approval. The remaining BFN, Unit 3, changes will be implemented upon completion of required supporting modification work and prior to entering Mode 3 (i.e., Hot Shutdown) from the spring 2014 refueling outage.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (77 FR 66490). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Benjamin T. Parks
Vijay K. Goel

Date: February 15, 2013

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Nuclear Regulatory Commission's biweekly *Federal Register* notice.

Sincerely,

/RA by Siva Lingam for/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-260 and 50-296

Enclosures:
1. Amendment No. 309 to License No. DPR-52
2. Amendment No. 268 to License No. DPR-68
3. Safety Evaluation

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