TVA

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

July 30, 2012

10 CFR 50.90

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

Browns Ferry Nuclear Plant, Units 2 and 3
Facility Operating License Nos. DPR-52 and DPR-68
NRC Docket Nos. 50-260 and 50-296

Subject: **Supplement to Technical Specification Change TS-429 – Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3**

Reference:

1. Technical Specification Change TS-429 – Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3, dated February 25, 2011

2. Letter from TVA to NRC, "Technical Specification Change TS-473, AREVA Fuel Transition," dated April 16, 2010

3. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Unit 1 -Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel (TAC NO. ME3775)," Request for Additional Information (RAI) Regarding TS-473, AREVA Fuel Transition (TAC No. ME3775)," ML110180585, dated August 23, 2011

4. Letter from TVA to NRC, "Response to NRC Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel," dated October 7. 2011

5. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to AREVA Fuel," dated April 27, 2012

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6. Letter from NRC to Technical Specifications Task Force, "Implementation of Travelers TSTF-363, Revision 0, "Revise Topical Report References in ITS 5.6.5, COLR [Core Operating Limits Report]," TSTF-408, Revision 1, "Relocation of LTOP [Low-Temperature Overpressure Protection] Enable Temperature and PORV [Power-Operated Relief Valve] Lift Setting to the PTLR [Pressure-Temperature Limits Report]," and TSTF-419, Revision 0, "Revise PTLR Definition and References in ITSS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR," dated August 4, 2011.

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

8. Letter from TVA to NRC, "Revised Commitment Date for Updated Loss of Coolant Accident Analysis for Browns Ferry Nuclear Plant, Units 2 and 3," dated June 29, 2012

By letter dated February 25, 2011 (Reference 1), the Tennessee Valley Authority (TVA) submitted a proposed Technical Specifications (TS) amendment to delete Browns Ferry Nuclear (BFN), Units 2 and 3, TS Surveillance Requirement (SR) 3.5.1.12. SR 3.5.1.12 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..

The enclosure to the Reference 1 letter identified ANP-2908(P) Revision 0, "Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Break Spectrum Analysis," AREVA NP Inc., dated March 2010, as the current Loss of Coolant Accident (LOCA) analysis of record for BFN, Units 2 and 3. ANP-2908(P) applied the EXEM BWR-2000 Evaluation Methodology to produce the Emergency Core Cooling System (ECCS) Model used to perform the LOCA Analysis.

By letter dated April 16, 2010 (Reference 2), the Tennessee Valley Authority (TVA) submitted "Technical Specification Change TS-473, AREVA Fuel Transition," to the NRC requesting approval of a license amendment to support using AREVA Fuel in Unit 1 at BFN. As part of the NRC review of the BFN Unit 1 ATRIUM™-10 fuel transition License Amendment Request (LAR), the staff conducted an onsite audit of the AREVA EXEM BWR-2000 emergency core cooling system evaluation model insofar as it has been applied to support the transition to AREVA fuel and safety analysis methods at Browns Ferry Nuclear Plant, Unit 1. The audit was conducted the week of July 18, 2011, at AREVA's Richland, Washington, facilities. During the audit, the NRC questioned the analyzed top-down cooling mechanisms of the EXEM BWR-2000 LOCA methodology. This question is documented in the August 23, 2011 Request for Additional Information (RAI) letter from the NRC (Reference 3) related to Technical Specification Change TS-473.

In order to address the issue raised by the NRC, the EXEM BWR-2000 Evaluation Model was modified for specific application to BFN, Units 1, 2, and 3. The result of the LOCA Analysis using the modified EXEM-2000 Evaluation Model is documented in ANP-3015(P) "Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis." ANP-3015(P) was submitted to the NRC in Reference 4. The application of the modified EXEM BWR-2000 Evaluation Model was approved for use on BFN, Unit 1, by the NRC in a License Amendment issued on April 27, 2012 (Reference 5). In addition to the TS Changes previously requested in reference 1, TVA is requesting approval to apply the modified EXEM BWR-2000 Evaluation Methodology to BFN, Units 2 and 3.

To incorporate a BFN, Units 2 and 3, specific approval, item 16 of the TS 5.6.5b, "Core Operating Limits Report (COLR)," will be revised to reference the NRC Approved Safety Evaluation. Reference 11 of TS Bases 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," will also be revised to incorporate the reference to the NRC Safety Evaluation described above. Additionally, based on the NRC position documented in Reference 6, the methodology references in TS 5.6.5 are revised to include revision number and revision date.
By letter dated April 18, 2012 (Reference 7), TVA committed to provide the modified LOCA Methodology to the NRC by June 30, 2012. By letter dated June 29, 2012 (Reference 8), TVA revised the commitment date for providing the modified LOCA Methodology to July 30, 2012.

As part of Technical Specification Change TS-473, AREVA Fuel Transition (Reference 4), TVA committed to revise the BFN Units 2 and 3 TS 3.3.1.1, 5.6.5.a, and 5.6.5.b to include the AREVA Methodolgy for the Oscillation Power Range Monitor (OPRM) Upscale Function period based detection algorithm setpoint limits. The enclosure to this letter provides the evaluation for the proposed changes. Attachments 1 through 4 of the enclosure to this letter provides the marked-up proposed TS and Bases pages, and the retyped proposed TS and Bases pages for BFN, Units 2 and 3. The evaluation for the proposed changes includes a description of the proposed changes, the technical evaluation, the no significant hazards determination, and the environmental evaluation. The enclosure to this letter supersedes, in its entirety, the enclosure to the February 25, 2011 letter.

Attachment 5 of the enclosure to this letter contains information that AREVA NP considers to be proprietary in nature and subsequently, pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4), it is requested that such information be withheld from public disclosure.

Attachment 6 of the enclosure to this letter contains the redacted version of the proprietary. Attachment 5 of the enclosure to this letter with the proprietary material removed, suitable for public disclosure.

Attachment 7 of the enclosure to this letter provides the affidavit, supporting this request.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the proposed TS change qualifies for categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), "Notice for public comment; State consultation," a copy of this application, with attachments, is being provided to the designated State of Alabama official.

TVA requests the approval of the proposed License Amendments by December 16, 2012 to support BFN, Unit 2, implementation of required supporting modification work during the BFN, Unit 2, refueling outage currently scheduled March 16, 2013. For BFN, Unit 2, implementation of the proposed License Amendment will be implemented prior to entering Mode 3 (i.e., Hot Shutdown) for this spring 2013 refueling outage. TVA proposes a BFN, Unit 3, license condition to permit partial implementation of the TS changes in accordance with the following schedule. BFN, Unit 3 TSs 5.6.5 and 3.3.1.1 will be implemented within 60 days of approval. The remaining BFN, Unit 3, changes will be implemented for BFN Unit 3, upon completion of required supporting modification work and prior to entering Mode 3 (i.e., Hot Shutdown) from the spring 2014 refueling outage.

There is no new regulatory commitment in this license amendment request. If you should have any questions regarding this submittal, please contact Tom Hess at (423) 751-3487.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on this 30th day of July, 2012.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure: Revised Evaluation for Technical Specification Change TS-429, Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3

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

REVISED EVALUATION FOR TECHNICAL SPECIFICATION CHANGE TS-429
Deletion of Low Pressure Coolant Injection Motor-Generator Sets for
Browns Ferry Nuclear Plant, Units 2 and 3

ATTACHMENT 7

Affidavit

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the report ANP-3015(P) Revision 0, entitled, "Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis," dated September 2011 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 30th day of September, 2011.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/12

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-10-12
STATE OF WASHINGTON

ENCLOSURE

REVISED EVALUATION FOR TECHNICAL SPECIFICATION CHANGE TS-429
Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3

ATTACHMENTS:

1. Proposed Technical Specifications and Bases Page Markups for BFN, Unit 2
2. Proposed Technical Specifications and Bases Page Markups for BFN, Unit 3
3. Retyped Proposed Technical Specifications and Bases Pages for BFN, Unit 2
4. Retyped Proposed Technical Specifications and Bases Pages for BFN, Unit 3
5. ANP-3015(P), Revision 0, "Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis," Proprietary
6. ANP-3015(NP), Revision 0, "Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis," Non-Proprietary
7. Affidavit

1.0 SUMMARY DESCRIPTION

This evaluation supports the proposal to amend Operating License DPR-52 for Browns Ferry Nuclear Plant (BFN), Unit 2, and Operating License DPR-68 for BFN, Unit 3. The proposed amendment would delete BFN, Units 2 and 3, Technical Specifications (TS) Surveillance Requirement (SR) 3.5.1.12. This SR 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. In addition, the BFN, Units 2 and 3, TS Bases 3.5.1, "ECCS - Operating," and TS Bases 3.8.7, "Distribution Systems - Operating," are modified to reflect the disabling of this automatic transfer capability and the deletion of the LPCI Motor-Generator (MG) Sets.

TS 5.6.5.b, "Core Operating Limits Report" (COLR), will be revised to reference the NRC Safety Evaluation which approved the plant specific application of the modified EXEM BWR-2000 LOCA methodology and revision number and revision dates for all COLR references. Function 2.f of Table 1 of TS 3.3.1.1, Reactor Protection Systems Instrumentation, and TS 5.6.5, COLR, will be revised to indicate that the Oscillation Power Range Monitor (OPRM) Upscale Function period based detection algorithm setpoint limits are included in the COLR. TS 5.6.5.b will be revised to include the AREVA stability related Topical Reports which describe the analytical methods used for determining the OPRM period based detection algorithm setpoint limits.

1.1 Equipment Historical Background

The current design of BFN, Units 2 and 3, provided for automatic transfer of the power supply for the Low Pressure Coolant Injection (LPCI) inboard injection, Residual Heat Removal (RHR) minimum flow valves, and recirculation pump discharge valves to the alternate source when low voltage is detected on the primary source. The design included MG Sets (i.e., LPCI MG Sets) to provide electrical divisional isolation between the 1E class normal and alternate power feeds to Reactor Motor-Operated Valve (RMOV) Boards D and E, while allowing for the operability of both Emergency Core Cooling System (ECCS) electrical trains when the power supply was swapped over. Currently, these LPCI MG Sets for BFN, Unit 2, and BFN, Unit 3, are obsolete and are high maintenance equipment.

1.2 Design and Licensing Bases

Currently, BFN, Units 2 and 3, RMOV Boards D and E automatically transfer the power supply from the normal source to the alternate source upon detection of low voltage at the normal power source. 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 50 Appendix K, "ECCS Evaluation Models," and 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," using older LOCA analysis methods.

Based on improved Loss of Coolant Accident (LOCA) analysis methods, the automatic transfer of power is no longer required. This is demonstrated in the AREVA LOCA Break Spectrum Analysis for BFN, Units 1, 2, and 3 (Reference 1). This analysis does not require the automatic transfer of the power supply for the LPCI inboard injection valves, RHR minimum flow valves, and recirculation pump discharge valves.

10 CFR 50.46 regulatory requirements are met by the use of two independent electrical power divisions for the ECCS equipment.

Deletion of the requirement for the automatic transfer function of RMOV Boards D and E will not change the number of ECCS subsystems credited in the current BFN licensing basis for BFN, Units 2, or BFN, Unit 3, since the automatic transfer function is no longer credited for the BFN LOCA Break Spectrum Analysis.

2.0 DETAILED DESCRIPTION

The proposed change eliminates the requirement to maintain an automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves. The specific proposed TS changes are described below. The associated TS Bases changes are provided for information.

2.1 Proposed Technical Specification Changes

The proposed change is to delete TS SR 3.5.1.12 for BFN, Units 2 and 3. This SR requires the verification of the capability to automatically transfer the power supply from the normal source to the alternate source for each LPCI subsystem inboard injection valve and each recirculation pump discharge valve on a 24-month frequency. In addition, TS Bases 3.5.1 and 3.8.7 for BFN, Units 2 and 3, are modified to reflect the disabling of the automatic transfer capability and any reference to the LPCI MG Sets.

The Tennessee Valley Authority (TVA) is requesting that TS SR 3.5.1.12 for BFN, Units 2 and 3, be deleted to support a modification to allow for the removal of the LPCI MG Sets. 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, while allowing for the operability of both ECCS electrical trains when the power supply is swapped over, will be removed from service since the automatic transfer function is no longer credited for the BFN LOCA Break Spectrum Analysis. BFN, Units 2 and 3, RMOV Boards D and E will be connected directly to their power supplies with both of the alternate supply breakers normally open to provide isolation between electrical divisions.

ANP-2908(P) Revision 0, "Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Break Spectrum Analysis," AREVA NP Inc., dated March 2010, is the current LOCA analysis of record for BFN, Units 2 and 3. ANP-2908 applied the EXEM BWR-2000 Evaluation Methodology to produce the ECCS Model used to perform the LOCA Analysis. By letter dated April 16, 2010, the Tennessee Valley Authority (TVA) submitted "Technical Specification Change TS-473, AREVA Fuel Transition," to the NRC requesting approval of a license amendment to support using AREVA Fuel in Unit 1 at BFN. As part of the NRC review of the BFN Unit 1 ATRIUM™-10 fuel transition License Amendment Request (LAR), the staff conducted an onsite audit of the AREVA EXEM BWR-2000 emergency core cooling system evaluation model insofar as it has been applied to support the transition to AREVA fuel and safety analysis methods at Browns Ferry Nuclear Plant, Unit 1. The audit was conducted the week of July 18, 2011, at AREVA's Richland, Washington, facilities. During the audit, the NRC questioned the analyzed top-down cooling mechanisms of the EXEM BWR-2000 LOCA methodology. This question is documented in the August 23, 2011 Request for Additional Information letter from the NRC (Reference 8) related to Technical Specification Change TS-473. In order to address the issue raised by the NRC, the EXEM BWR-2000 Evaluation Model was modified for specific application to BFN, Units 1, 2, and 3. The result of the LOCA Analysis using the modified EXEM-2000 Evaluation Model is documented in ANP-

3015(P) "Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis." ANP-3015(P) was submitted to the NRC in reference 9. The application of the modified EXEM BWR-2000 Evaluation Model was approved for use on BFN, Unit 1, by the NRC in a License Amendment issued on April 27, 2012 (Reference 6). In addition to the TS Changes previously requested in, TVA is requesting approval to apply the modified EXEM BWR-2000 Evaluation Methodology to BFN, Units 2 and 3.

To incorporate the modified EXEM BWR-2000 Evaluation Methodology to BFN Units, 2 and 3, item 16 of TS 5.6.5b, "Core Operating Limits Report (COLR)," is revised to reference the NRC Approved Safety Evaluation. Reference 11 of TS Bases 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," is revised to incorporate the reference to the NRC Safety Evaluation described above. TVA revised all of the methodology references in TS 5.6.5 to include a revision number and revision date consistent with an NRC position documented in a letter from the NRC to the TS Task Force, dated August 4, 2011.

As part of Technical Specification Change TS-473, AREVA Fuel Transition (Reference 7), TVA committed to make the following revisions the BFN Units 2 and 3 TS 3.3.1.1, 5.6.5.a, and 5.6.5.b to include the AREVA Methodolgy for the Oscillation Power Range Monitor (OPRM) Upscale Function period based detection algorithm setpoint limits:

- Function 2.f 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 5.6.5.b will be revised to include the AREVA stability related Topical Reports which describe the analytical methods used for determining the OPRM period based detection algorithm setpoint limits.

TVA received approval of a similar TS change to support the deletion of the automatic transfer function and the associated LPCI MG Sets on June 20, 2005 (Reference 2) for BFN, Unit 1. The BFN, Unit 1, LPCI MG Sets and their RMOV Boards 1D and 1E were then removed from service. For BFN, Unit 1, loads that were once on RMOV Boards 1D and 1E are now powered from BFN, Unit 1, RMOV Boards A and B.

However, BFN, Units 2 and 3, will retain their RMOV boards in the planned modification, which will eliminate the LPCI MG Sets. Currently, the BFN, Units 2 and 3 RMOV Boards D and E are being powered by the LPCI MG Sets. After the modification, the BFN, Units 2 and 3, RMOV Boards D and E will be powered directly from the 480V Shutdown Boards. Loads presently on BFN, Units 2 and 3, RMOV Boards D and E will remain on the respective RMOV boards.

Mark-ups of the proposed changes to the TS and Bases are provided in Attachments 1 and 2 for BFN, Units 2 and 3, respectively. Attachments 3 and 4 provide the retyped TS and Bases pages reflecting the incorporation of the proposed changes for BFN, Units 2 and 3, respectively.

TVA requests the approval of the proposed License Amendments by December 16, 2012 to support BFN, Unit 2, implementation of required supporting modification work during the BFN, Unit 2, refueling outage currently scheduled March 16, 2013. For BFN, Unit 2, implementation of the proposed License Amendment will be implemented prior to entering Mode 3 (i.e., Hot Shutdown) for this spring 2013 refueling outage. TVA proposes a BFN, Unit 3, license condition to permit partial implementation of the TS changes in accordance with the following schedule.

BFN, Unit 3 TSs 5.6.5 and 3.3.1.1 will be implemented within 60 days of approval. The proposed partial implementation schedule is needed to resolve a BFN, Unit 3, degraded/ nonconforming condition involving the AREVA LOCA Analysis. The remaining BFN, Unit 3, changes will be implemented for BFN Unit 3, upon completion of required supporting modification work and prior to entering Mode 3 (i.e., Hot Shutdown) from the spring 2014 refueling outage.

3.0 TECHNICAL EVALUATION

3.1 Current Electrical Distribution System

BFN is a three-unit plant. As discussed in Updated Final Safety Analysis Report (UFSAR) Sections 8.4, "Normal Auxiliary Power System," and 8.5, "Standby AC Power Supply and Distribution," there are several sources of offsite and onsite power for BFN.

During normal operation, station auxiliary power is taken from the main generator through the unit station service transformers. During startup and shutdown, auxiliary power is supplied from the 500-kV system through the main transformers to the unit station service transformer with the main generators isolated by the main generator breakers. Auxiliary power is also available through the two common station service transformers which are fed from the 161-kV system. Standby (onsite) power is supplied by eight diesel generator units (four for BFN Units 1 and 2, and four for BFN, Unit 3).

There are five 480V RMOV Boards (A through E) powered by 480V Shutdown Boards A and B for BFN, Unit 2, and BFN, Unit 3. The 480V RMOV Boards A and D are normally powered from 480V Shutdown Board A with Division I power. The 480V Shutdown Board B is the alternate power supply. The 480V RMOV Boards B, C and E are normally powered from 480V Shutdown Board B with Division II power. The 480V Shutdown Board A is the alternate power supply. (Note that the designations used for the boards, valves and MG Sets in the text of this submittal have been generalized to improve readability. The actual RMOV board designations are 2A through 2E on BFN, Unit 2, and 3A through 3E on BFN, Unit 3. The valves and MG Set designations are also prefixed with the associated unit number.)

Currently, power to BFN, Unit 2, and BFN, Unit 3, 480V RMOV Boards D and E are supplied from 480V Shutdown Boards A and B via MG Sets. There are four MG Sets in BFN, Unit 2, and four in BFN, Unit 3. Two MG Sets are fed from 480V Shutdown Board A and act as a normal power source for 480V RMOV Board D (MG Set DN) and as an alternate power source to 480V RMOV Board E (MG Set EA). Two MG Sets are fed from 480V Shutdown Board B and act as a normal power source for 480V RMOV Board E (MG Set EN) and as an alternate power source to 480V RMOV Board D (MG Set DA).

Currently, BFN, Unit 2, and BFN, Unit 3, 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.

3.2 Design of the Emergency Core Cooling System

The BFN ECCS consists of the following:

- High Pressure Coolant Injection (HPCI);
- Automatic Depressurization System (ADS);
- Low Pressure Core Spray (LPCS); and
- LPCI, which is an operating mode of RHR.

The ECCS subsystems are designed to limit clad temperature over the complete spectrum of possible break sizes in the nuclear system process barrier, including the design basis break. The design basis break is defined as the complete and sudden circumferential rupture of the largest pipe connected to the reactor vessel (i.e., one of the recirculation loop pipes) with displacement of the ends so that blow down occurs from both ends.

The low-pressure ECCS consists of LPCS and LPCI. The LPCS consists of two independent loops. Each loop consists of two pumps, a spray sparger inside the core shroud and above the core, piping, and valves to convey water from the pressure suppression pool to the sparger, and the associated controls and instrumentation. When the system is actuated, water is taken from the pressure suppression pool. Flow then passes through a normally open motor-operated valve in the suction line to each 50% capacity pump.

The RHR System is designed for five modes of operation (i.e., shutdown cooling, containment spray and suppression pool cooling, LPCI, standby cooling, and supplemental fuel pool cooling). During LPCI operation, four RHR pumps take suction from the pressure suppression pool and discharge to the reactor vessel into the core region through both of the recirculation loops. Two pumps discharge to each recirculation loop.

The design function for the equipment powered from BFN, Unit 2, and BFN, Unit 3, 480V RMOV Boards D and E is as follows.

- Recirculation Pump Discharge Valves (FCV-68-79 and 3) - After receipt of a LPCI initiation signal, a signal is transmitted to the recirculation pump discharge valve control logic in each loop of the Recirculation System to close each valve once the reactor vessel pressure has sufficiently decreased.

- RHR Pump Minimum Flow Bypass Valves (FCV-74-7 and 30) - The RHR pump minimum flow bypass line header isolation valves are automatically controlled by control logic to start or stop flow through the two RHR pump minimum flow bypass lines of the associated loop. The isolation valve is automatically opened if its associated loop injection flow is less than approximately 3,500 gpm, concurrent with indication that either of the two RHR pumps in the respective loop is running. The isolation valve is automatically closed if its associated loop injection flow is greater than the set point.

- RHR Inboard Valves (FCV-74-53 and 67) - The RHR Inboard Valves are opened upon receipt of a LPCI initiation signal once the reactor vessel pressure has sufficiently decreased.

3.3 Historical Basis for the Electrical and Emergency Core Cooling Systems Design

As discussed in UFSAR Section 1.5, "Principal Design Criteria," sufficient redundancy and independence is provided for essential safety functions to ensure that no single failure of active components can prevent the required actions. For systems or components to which IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations," is applicable, single failures of passive electrical components are also considered.

Following initial startup and operation, the electrical system design was modified to satisfy the more stringent limitations required by 10 CFR 50, Appendix K, and to resolve other regulatory issues (late 1970s). BFN was using the General Electric (GE) SAFE/CHASTE/REFLOOD LOCA analysis methodology when it modified the ECCS logic. In order to obtain acceptable results utilizing the SAFE/CHASTE/REFLOOD LOCA analysis methodology, TVA had to ensure that at least one RHR pump would be operating in each LPCI loop prior to the postulated single failure to mitigate the consequences of a recirculation suction line break.

The automatic transfer capability for BFN, Unit 2, and BFN, Unit 3, 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, TVA was concerned that the automatic transfer could propagate an electrical fault to both divisions of power supply. As a result, BFN, Unit 2 and BFN, Unit 3 LPCI MG Sets were included in the design for both the normal and alternate power supplies to provide electrical isolation between the associated 480V Shutdown Board and the RMOV Board.

In 1996, TVA replaced the SAFE/CHASTE/REFLOOD LOCA analysis methodology with the SAFER/GESTR-LOCA methodology. The plant specific analysis to support the change to the SAFER/GESTR model and the associated TS changes were provided to NRC in per References 3 and 4. NRC issued the change in Reference 5. With the change to SAFER/GESTR, the BFN LOCA analyses no longer credited the automatic transfer of power for LPCI.

3.4 Proposed Change to the Emergency Core Cooling System Performance Analysis

ANP-2908(P) Revision 0, "Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Break Spectrum Analysis," AREVA NP Inc., dated March 2010, is the current LOCA analysis of record for BFN, Units 2 and 3. ANP-2908 applied the EXEM BWR-2000 Evaluation Methodology to produce the ECCS Model used to perform the LOCA Analysis. By letter dated April 16, 2010, TVA submitted "Technical Specification Change TS-473, AREVA Fuel Transition," to the NRC requesting approval of a license amendment to support using AREVA Fuel in Unit 1 at BFN. As part of the NRC review of the BFN Unit 1 ATRIUM™-10 fuel transition License Amendment Request (LAR), the staff conducted an onsite audit of the AREVA EXEM BWR-2000 emergency core cooling system evaluation model insofar as it has been applied to support the transition to AREVA fuel and safety analysis methods at Browns Ferry Nuclear Plant, Unit 1. The audit was conducted the week of July 18, 2011, at AREVA's Richland, Washington, facilities. During the audit, the NRC

questioned the analyzed top-down cooling mechanisms of the EXEM BWR-2000 LOCA methodology. This question is documented in the August 23, 2011 Request for Additional Information letter from the NRC related to Technical Specification Change TS-473. In order to address the issues raised by the NRC, the EXEM BWR-2000 Evaluation Model has been modified for specific application to BFN Units 1, 2, and 3. Section 4.0 of Attachment 5, "ANP-3015(P), Revision 0, "Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis," contains a detailed description of the changes made to the EXEM BWR-2000 LOCA methodology to resolve the issue with top-down cooling. Attachment 5 contains the results of the LOCA Break Spectrum Analysis performed using the modified EXEM BWR-2000 LOCA methodology. The application of the modified EXEM BWR-2000 Evaluation Model has been approved for use on BFN, Unit 1 (Reference 6).

To incorporate the modified EXEM BWR-2000 Evaluation Methodology to BFN Units, 2 and 3, item 16 of TS 5.6.5b, "Core Operating Limits Report (COLR)," is revised to reference the NRC Approved Safety Evaluation. Reference 11 of TS Bases 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," is revised to incorporate the reference to the NRC Safety Evaluation described above. TVA revised all of the methodology references in TS 5.6.5 to include a revision number and revision date consistent with an NRC position documented in a letter from the NRC to the TS Task Force, dated August 4, 2011.

As part of Technical Specification Change TS-473, AREVA Fuel Transition (Reference 7), TVA committed to make the following revisions the BFN Units 2 and 3 TS 3.3.1.1, 5.6.5.a, and 5.6.5.b to include the AREVA Methodolgy for the Oscillation Power Range Monitor (OPRM) Upscale Function period based detection algorithm setpoint limits:

- Function 2.f 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 5.6.5.b will be revised to include the AREVA stability related Topical Reports which describe the analytical methods used for determining the OPRM period based detection algorithm setpoint limits.

Operational equipment assumptions for the analyses are shown in Table 1, "BFN ECCS Credited for Recirculation Line Break LOCAs." Terminology for assumed single failures (SF) used in Table 1 is as follows.

- Backup battery power (SF-BATT)
    - Unit Battery supplying 250VDC RMOV Board 1A, 2A, or 3A (SF-BATT|BA)
    - Unit Battery supplying 250VDC RMOV Board 1B, 2B, or 3B Board B (SF-BATT|BB)
    - Unit Battery supplying 250VDC RMOV Board 1C, 2C, or 3C Board C (SF-BATT|BC)

  Note: There are three Unit Batteries (1, 2, and 3) shared between the three BFN units and supplying power to the 250VDC RMOV Boards.
- Opposite unit false LOCA signal (SF-LOCA)
- LPCI valve (SF-LPCI)
- Diesel Generator (SF-DGEN)
- HPCI System (SF-HPCI)
- ADS (SF-ADS)
    - Failure of ADS initiation logic (SF-ADS|IL)
    - Failure of a single ADS valve (SF-ADS|SV)

Table 1, BFN ECCS Credited for Recirculation Line Break LOCAs

Assumed Failure	Systems* † Remaining	
	Recirculation‡ Suction Break	Recirculation Discharge Break
SF-BATT|BA	6 ADS, 1 LPCS, 2 LPCI	6 ADS, 1 LPCS
SF-BATT|BB	HPCI, 1 LPCS, 2 LPCI, 4 ADS	HPCI, 1 LPCS, 4 ADS
SF-BATT|BC§	4 ADS, HPCI, 1 LPCS, 3 LPCI	4 ADS, HPCI, 1 LPCS, 1 LPCI
SF-LOCA	6 ADS, HPCI, 1 LPCS, 2 LPCI	6 ADS, HPCI, 1 LPCS
SF-LPCI	6 ADS, HPCI, 2 LPCS, 2 LPCI	6 ADS, HPCI, 2 LPCS
SF-DGEN	6 ADS, HPCI, 1 LPCS, 2 LPCI	6 ADS, HPCI, 1 LPCS
SF-HPCI	6 ADS, 2 LPCS, 4 LPCI	6 ADS, 2 LPCS, 2 LPCI
SF-ADS|IL	HPCI, 2 LPCS, 4 LPCI, 4 ADS	HPCI, 2 LPCS, 2 LPCI, 4 ADS
SF-ADS|SV	5 ADS, HPCI, 2 LPCS, 4 LPCI	5 ADS, HPCI, 2 LPCS, 2 LPCI

* Each LPCS means operation of two core spray pumps in a system. It is assumed that both pumps in a system must operate to take credit for core spray cooling or inventory makeup. Furthermore, 2 LPCI refers to two LPCI pumps into one loop, 3 LPCI refers to two LPCI pumps into one loop and one LPCI pump into one loop. 4 LPCI refers to four LPCI pumps into two loops, two per loop.

† 4 ADS, 5 ADS and 6 ADS means the number of ADS valves available for automatic activation.

‡ Systems remaining, as identified in this table for recirculation suction line breaks, are applicable to other non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed for recirculation suction breaks, less the ECCS in which the break is assumed.

§ BFN, Unit 3, systems remaining. Conservative for BFN, Units 1 and 2.

3.5 Proposed Change to the Electrical Distribution System

The following changes will be made to the Electrical Distribution System:

1. BFN, Units 2 and 3, LPCI MG Sets DN, DA, EN, and EA and their locally mounted instrumentation and controls will be removed from service and abandoned in place.
2. Instead of RMOV Boards D and E being powered from the LPCI MG Sets, they will be powered directly from the corresponding 480V Shutdown Boards through their normal feeds.
3. The feeder breakers for 480V RMOV Boards D and E at the applicable 480V Shutdown Boards will be modified from electrically operated to mechanically operated.
4. The alternate feeder breakers for 480V RMOV Boards D and E at the applicable 480V Shutdown Boards will be changed from normally closed to normally open.

The current configuration of the portion of the electrical distribution system associated with this planned change is shown in Figures 1 and 3 for BFN, Units 2 and 3, respectively.

After the planned change, the resulting configuration of the portion of the electrical distribution system associated with this planned change will be as shown in Figures 2 and 4 for BFN, Units 2 and 3, respectively.

Figure 1, Current Configuration of Portion of BFN, Unit 2, Electrical Distribution System Associated with Planned Change

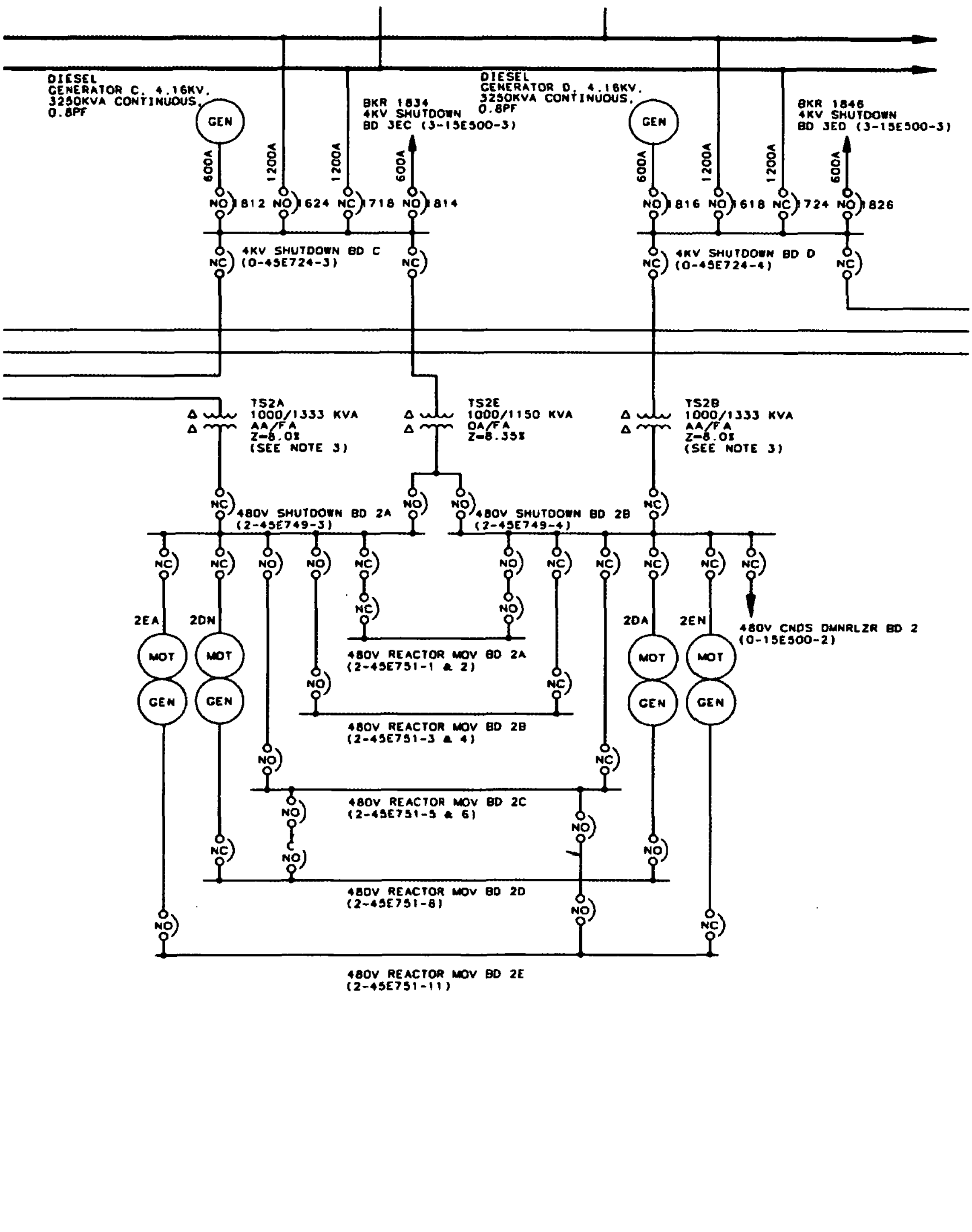


Figure 2, Resulting Configuration of Portion of BFN, Unit 2, Electrical Distribution System Associated with Planned Change

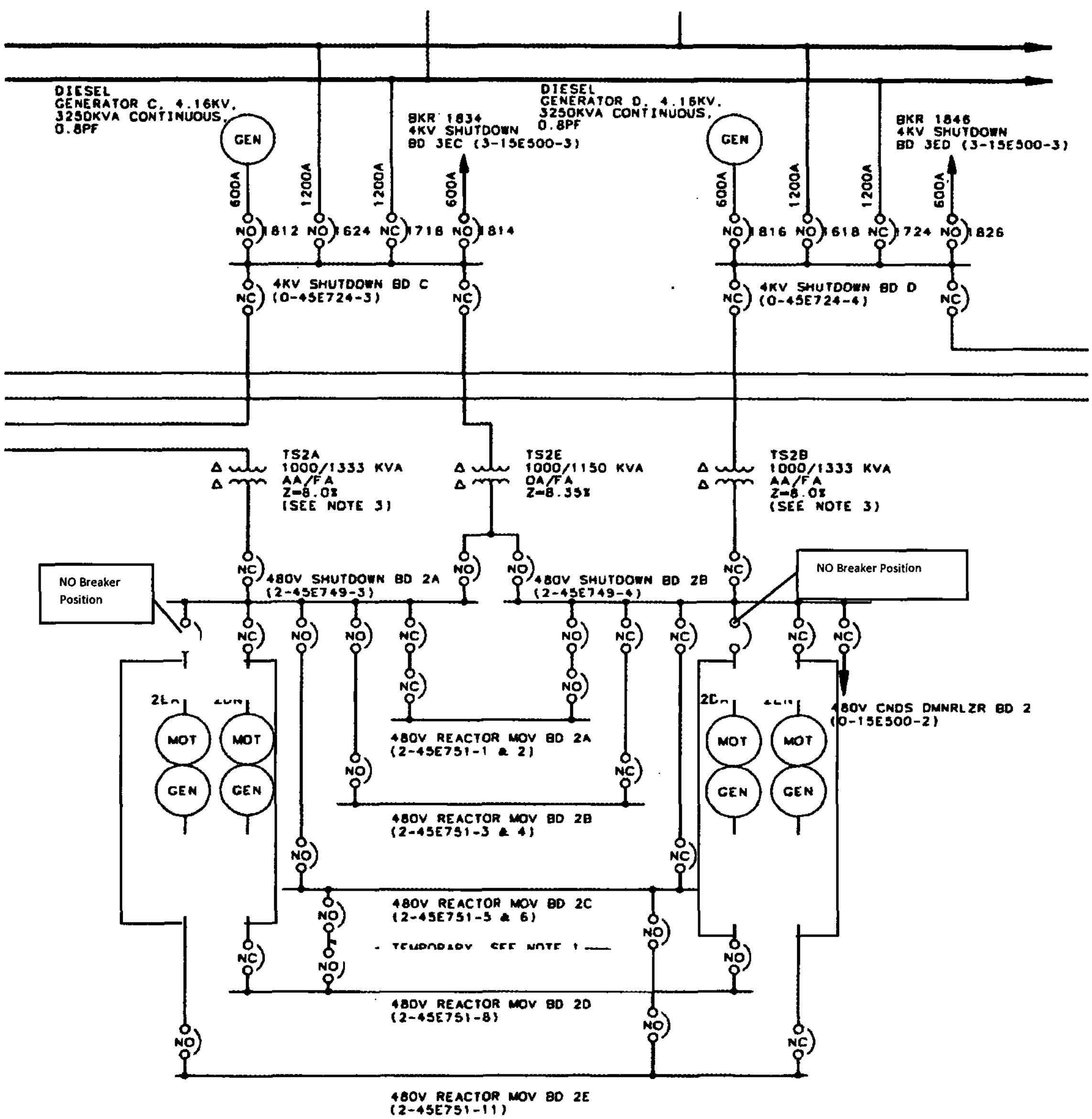


Figure 3, Current Configuration of Portion of BFN, Unit 3, Electrical Distribution System Associated with Planned Change

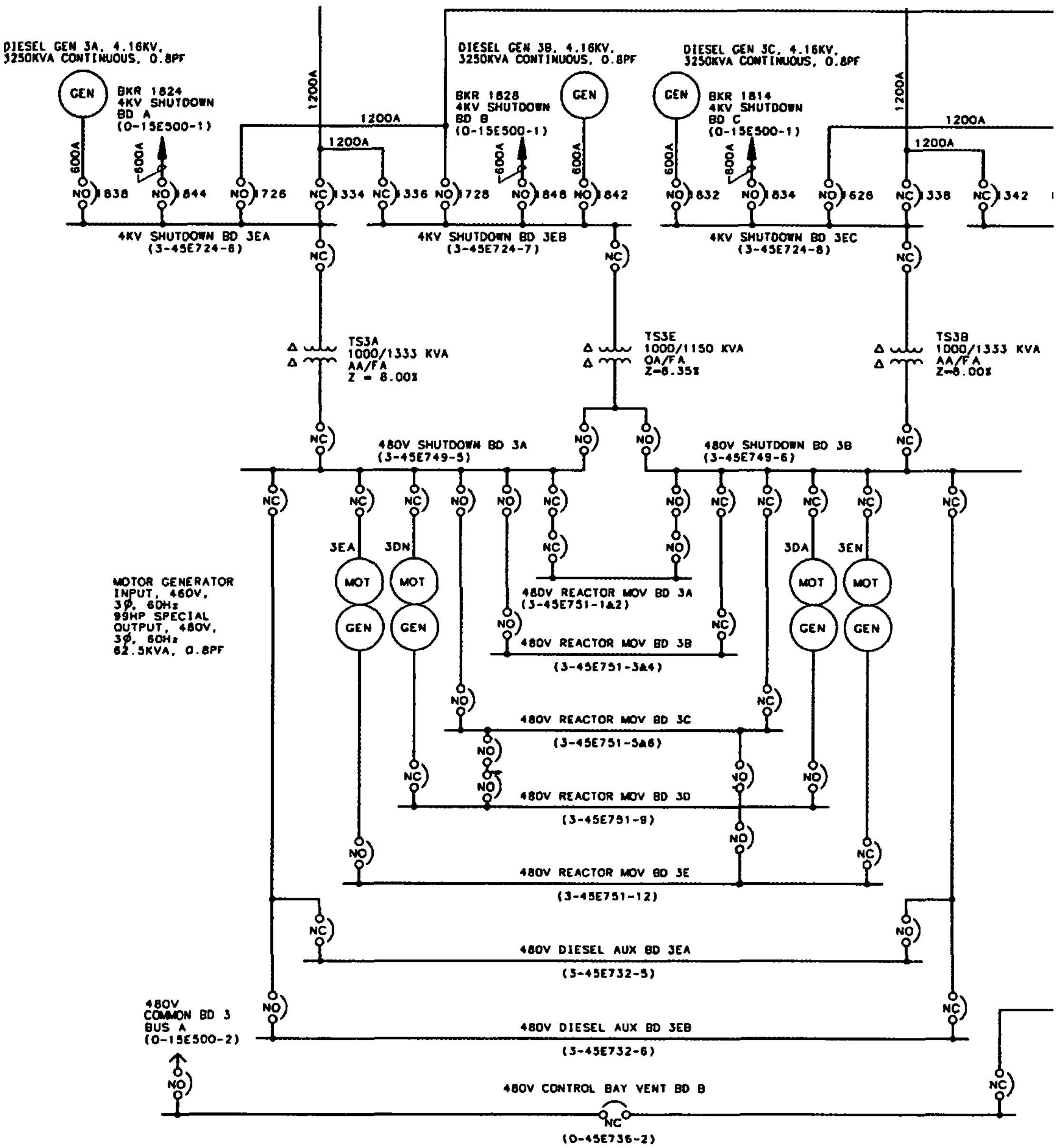
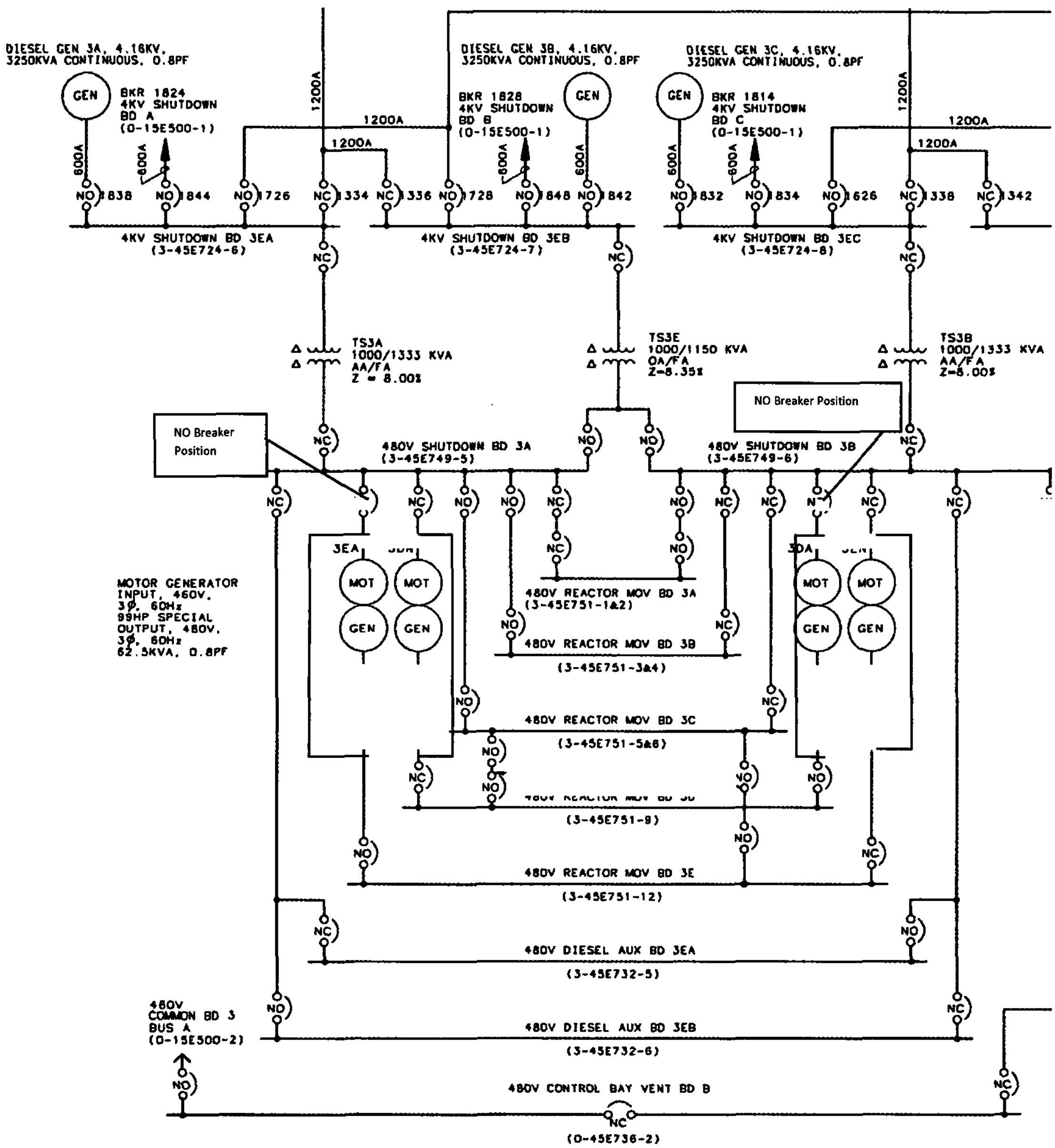


Figure 4, Resulting Configuration of Portion of BFN, Unit 3, Electrical Distribution System Associated with Planned Change



3.6 Evaluation of Proposed Change to the Emergency Core Cooling System Performance Analysis

The proposed implementation of EXEM BWR-2000 LOCA methodology, as supported by the modified analysis described in ANP-3015(P) meets the requirements and acceptable features of ECCS evaluation models described in Appendix K to 10 CFR Part 50, per 10 CFR 50.46(a)(1)(ii). The results of the LOCA Break Spectrum Analysis performed using the modified EXEM BWR-2000 LOCA methodology show that the requirements of 10 CFR 50.46(b) are maintained.

The proposed changes to TS 5.6.5 and 3.3.1.1 are necessary and appropriate to implement the AREVA fuel design, and associated analytical methodologies.

3.7 Evaluation of Planned Change on the Electrical Distribution System

Altering the configuration as planned above has the potential for introducing different failure modes than were previously considered. A failure modes evaluation was performed, which concluded that there will be no adverse effects as a result of the planned changes. The design function of the LPCI MG Sets was to provide electrical isolation between redundant divisions of the electrical distribution system in the event of a malfunction of the automatic transfer of 480V RMOV Boards D or E resulting in both normal and alternate supply breakers being closed at the same time. Without the LPCI MG Sets in the circuit, this malfunction would have allowed a fault to propagate from one division to the other. The new electrical system configuration eliminates that concern by changing the alternate feeder breakers for 480V RMOV Boards D and E at the applicable 480V Shutdown Boards from normally closed to normally open. These breakers will be modified to not automatically transfer, which ensures the redundant divisions remain electrically isolated from each other.

TVA's planned change is in conformance with 10 CFR 50.55a(h)(2), "Protection systems," and the BFN licensing basis. The BFN licensing basis for ECCS protection systems is described in UFSAR Sections 8.9, "Safety Systems Independence Criteria and Bases for Electrical Cable Installation," and 7.4, "Emergency Core Cooling Control and Instrumentation." These systems are designed to meet the intent of the Institute of Electrical and Electronics Engineers (IEEE) proposed Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE-279-1971).

3.8 Effect of Proposed Change on Actual Emergency Core Cooling System Performance and the Loss of Coolant Accident (LOCA) Analysis

Once the planned change is implemented, the loads powered from 480V RMOV Board D or E will not automatically transfer to continue to receive power. Available ECCS equipment, considering various single failure scenarios, and taking into account actual LOCA analyses assumptions, are described in Table 2, "ECCS Equipment Available and Credited in the LOCA Analysis for a Recirculation Suction Line Break Before and After the Planned Change," and Table 3, "ECCS Equipment Available and Credited in the LOCA Analysis for a Recirculation Discharge Line Break Before and After the Planned Change."

The ECCS equipment available following the postulated pipe break and single failure were determined by performing an analysis based on the physical configuration of the ECCS. The analysis started with the identification of ECCS equipment available prior to the postulated break. Then, each of the postulated break locations was evaluated. (Note: Break location plays a part in the analysis because the recirculation pump discharge pipe break results in the direct loss of a

LPCI loop, whereas recirculation pump suction pipe breaks do not result in the direct loss of any ECCS pump capability.) A loss of offsite power was also postulated to occur. One active single failure within the plant is postulated to occur concurrent with the pipe break. The single failure was determined based on ensuring that it results in the largest amount of equipment being lost. (For example, if two LPCI pumps (one loop) were lost as a result of the break location, a Diesel Generator supplying power to another pump in the opposite (unbroken) recirculation loop was selected as the single failure. This resulted in the largest amount of equipment lost due to the single failure). This analytical approach resulted in the identification of the minimum equipment remaining available for postulated break mitigation.

The planned change does not affect available equipment for eight of the nine most limiting postulated failures evaluated in the current LOCA analyses:

- The failure of unit battery board A (SF-BATT|BA);
- The failure of unit battery board B (SF-BATT|BB);
- The failure of unit battery board C (SF-BATT|BC);
- A spurious LOCA signal from another unit (SF-LOCA);
- The failure of a LPCI injection valve (SF-LPCI);
- The failure of the HPCI System (SF-HPCI);
- The failure of ADS initiation logic (SF-ADS|IL); or
- The failure of a single ADS valve (SF-ADS|SV).

The ninth limiting postulated failure is a LOCA (suction or discharge line break), without offsite power available, and the loss of a diesel generator is the assumed single failure. The scenario will cause the loss of power to either 480V RMOV Boards A and D or B and E. After the planned change is implemented, the loads powered from 480V RMOV Board D or E will not automatically transfer to receive power. Therefore, there will be one less LPCI pump actually available for injection into the vessel. However, as indicated in Tables 2 and 3, the planned change results in the same number of LPCI components available as is credited in the Reference 1 analysis of record.

Table 2, ECCS Equipment Available and Credited in the LOCA Analysis for a Recirculation Suction Line Break Before and After the Planned Change

Assumed Failure	ECCS Systems Actually Available Before the Planned Change	ECCS Systems Actually Available After the Planned Change	ECCS Systems Credited In the Analysis Before the Change	ECCS Systems Credited In the Analysis After the Change
SF-BATT|BA	6 ADS, 1 LPCS, 2 LPCI	(Same as available before the planned change)	6 ADS, 1 LPCS, 2 LPCI	(Same as credited before the planned change)
SF-BATT|BB	HPCI, 1 LPCS, 2 LPCI, 4 ADS	(Same as available before the planned change)	HPCI, 1 LPCS, 2 LPCI, 4 ADS	(Same as credited before the planned change)
SF-BATT|BC	4 ADS, HPCI, 1 LPCS, 3 LPCI	(Same as available before the planned change)	4 ADS, HPCI, 1 LPCS, 3 LPCI	(Same as credited before the planned change)
SF-LOCA	6 ADS, HPCI, 1 LPCS, 2 LPCI	(Same as available before the planned change)	6 ADS, HPCI, 1 LPCS, 2 LPCI	(Same as credited before the planned change)
SF-LPCI	6 ADS, HPCI, 2 LPCS, 2 LPCI	(Same as available before the planned change)	6 ADS, HPCI, 2 LPCS, 2 LPCI	(Same as credited before the planned change)
SF-DGEN	6 ADS, HPCI,1 LPCS, 3 LPCI	6 ADS, HPCI,1 LPCS, 2 LPCI	6 ADS, HPCI,1 LPCS, 2 LPCI	(Same as credited before the planned change)
SF-HPCI	6 ADS, 2 LPCS, 4 LPCI	(Same as available before the planned change)	6 ADS, 2 LPCS, 4 LPCI	(Same as credited before the planned change)
SF-ADS|IL	HPCI, 2 LPCS, 4 LPCI, 4 ADS	(Same as available before the planned change)	HPCI, 2 LPCS, 4 LPCI, 4 ADS	(Same as credited before the planned change)
SF-ADS|SV	5 ADS, HPCI, 2 LPCS, 4 LPCI	(Same as available before the planned change)	5 ADS, HPCI, 2 LPCS, 4 LPCI	(Same as credited before the planned change)

Table 3, ECCS Equipment Available and Credited in the LOCA Analysis for a Recirculation Discharge Line Break Before and After the Planned Change

Assumed Failure	ECCS Systems Actually Available Before the Planned Change	ECCS Systems Actually Available After the Planned Change	ECCS Systems Credited In the Analysis Before the Change	ECCS Systems Credited In the Analysis After the Change
SF-BATT|BA	6 ADS, 1 LPCS	(Same as available before the planned change)	6 ADS, 1 LPCS	(Same as credited before the planned change)
SF-BATT|BB	HPCI, 1 LPCS, 4 ADS	(Same as available before the planned change)	HPCI, 1 LPCS, 4 ADS	(Same as credited before the planned change)
SF-BATT|BC	4 ADS, HPCI, 1 LPCS, 1 LPCI	(Same as available before the planned change)	4 ADS, HPCI, 1 LPCS, 1 LPCI	(Same as credited before the planned change)
SF-LOCA	6 ADS, HPCI, 1 LPCS	(Same as available before the planned change)	6 ADS, HPCI, 1 LPCS	(Same as credited before the planned change)
SF-LPCI	6 ADS, HPCI, 2 LPCS	(Same as available before the planned change)	6 ADS, HPCI, 2 LPCS	(Same as credited before the planned change)
SF-DGEN	6 ADS, HPCI,1 LPCS, 1 LPCI	6 ADS, HPCI,1 LPCS	6 ADS, HPCI,1 LPCS	(Same as credited before the planned change)
SF-HPCI	6 ADS, 2 LPCS, 2 LPCI	(Same as available before the planned change)	6 ADS, 2 LPCS, 2 LPCI	(Same as credited before the planned change)
SF-ADS|IL	HPCI, 2 LPCS, 2 LPCI, 4 ADS	(Same as available before the planned change)	HPCI, 2 LPCS, 2 LPCI, 4 ADS	(Same as credited before the planned change)
SF-ADS|SV	5 ADS, HPCI, 2 LPCS, 2 LPCI	(Same as available before the planned change)	5 ADS, HPCI, 2 LPCS, 2 LPCI	(Same as credited before the planned change)

3.9 Technical Evaluation Summary

In summary, the automatic transfer of the power supply for the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves is not required to meet the modeling of these components in the safety analyses (LOCA). Regulatory requirements are met by the use of two independent divisions of ECCS equipment. Disabling of the automatic transfer function will not change the number of ECCS subsystems credited in the current BFN licensing basis.

The BFN, Unit 2, and BFN, Unit 3, 480V RMOV Boards D and E will be powered directly from the applicable 480V Shutdown Boards. This electrical alignment has been analyzed and determined to be acceptable for BFN, Unit 2, and BFN, Unit 3.

The modified ECCS LOCA methodology is in compliance with 10 CFR 50.46(a)(1)(ii) as an ECCS Evaluation Model conforming to the required and acceptable features of 10 CFR Appendix K. The results of the LOCA Break Spectrum Analysis performed using the modified EXEM BWR-2000 LOCA methodology show that the requirements of 10 CFR 50.46(b) are maintained.

The results of the deterministic evaluation provided in Sections 3.7 and 3.8 assure that the equipment required to safely shutdown the plant and mitigate the effects of a design basis accident, transient, or special event, will remain capable of performing their safety function with the deletion of the requirement to maintain an automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves.

The analytical methodologies to be used for design and licensing of ATRIUM-10 reloads are NRC approved and acceptable for establishing COLR limits. The proposed changes to TSs 5.6.5a and 3.3.1.1 are necessary and appropriate to implement the AREVA fuel design, and associated analytical methodologies.

4.0 REGULATORY EVALUATION

TVA is submitting a TS change request to licenses DPR-52 and DPR-68 for BFN, Unit 2, and BFN, Unit 3. The proposed TS change deletes a surveillance requirement to verify automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves.

TVA is also submitting a request to apply a modified version of the AREVA EXEM BWR-2000 LOCA methodology to BFN, Units 2 and 3. The modified methodology meets the requirements of 10 CFR 50.46(a)(1)(ii) as an ECCS Evaluation Model conforming to the required and acceptable features of 10 CFR 50 Appendix K.

4.1 Applicable Regulatory Requirements/Criteria

The proposed deletion of the requirement for an automatic transfer of the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves does not alter compliance with the requirements of 10 CFR 50, Appendix A, General Design Criterion 17, "Electric Power Systems," or the guidelines in Regulatory Guide 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants."

TVA's planned change is in conformance with 10 CFR 50.55a(h)(2) and the BFN licensing basis. The BFN licensing basis for ECCS protection systems is described in UFSAR Sections 8.9, "Safety Systems Independence Criteria and Bases for Electrical Cable Installation," and 7.4, "Emergency Core Cooling Control and Instrumentation." These systems are designed to meet the intent of the IEEE proposed Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE-279-1971).

The Normal Auxiliary Power System, Emergency AC Power System and the planned electrical distribution system will support the electrical loads necessary to mitigate the consequences of a design basis accident. The proposed deletion of a surveillance requirement to verify automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves does not change the number of ECCS subsystems credited in the BFN licensing basis. Therefore, the requirements of 10 CFR 50.46 and Appendix K continue to be met.

4.2 Precedent

The NRC previously approved similar changes for BFN, Unit 1, in the following License Amendments.

> "Browns Ferry Nuclear Plant, Unit 1 - Issuance of an Amendment Regarding Deletion of the Low Pressure Coolant Injection Motor-Generator Sets (TAC No. MC3822)(TS-427)," dated June 20, 2005. (ML051580047)
>
> "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to AREVA Fuel (TAC No. ME3775) (TS-473)," dated April 27, 2012. (ML12086A285)

4.3 Significant Hazards Consideration

The Tennessee Valley Authority (TVA) is submitting a Technical Specifications (TS) change request to licenses DPR-52 and DPR-68 for Browns Ferry Nuclear Plant (BFN), Unit 2, and BFN, Unit 3. The proposed TS change deletes a surveillance requirement to verify automatic transfer capability for the power supply to the Low Pressure Coolant Injection (LPCI) inboard injection valves, Residual Heat Removal (RHR) minimum flow valves and recirculation pump discharge valves. The proposed change will apply a new Emergency Core Cooling System (ECCS) Evaluation Model for the Loss of Coolant Accident (LOCA) Analysis and revise TS 5.6.5a and 3.3.1.1 to implement AREVA Analytical Methodologies. TS 5.6.5b is revised to include Revision Numbers and Revision Dates for AREVA Methodologies.

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed TS changes by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed Technical Specification change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The 480V RMOV Boards D or E, the equipment they power, or the automatic power transfer feature provided for these boards are not precursors to any accident previous evaluated in the Updated Final Safety Analysis Report (UFSAR). Therefore, the probability of an evaluated accident is not increased by modifying this equipment.

   The proposed deletion of a surveillance requirement to verify automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves does not change the number of Emergency Core Cooling System (ECCS) subsystems credited in the BFN licensing basis. The proposed change does not affect the operational characteristics or function of systems, structures, or components (SSCs), the interfaces between credited SSCs and other plant systems, or the reliability of SSCs. The proposed change does not impact the capability of credited SSCs to perform their required safety functions.

   The proposed change to the ECCS Evaluation Model meets the requirements of 10 CFR 50.46(a)(1)(ii) and ensures the limits of 10 CFR 50.46(b) are maintained. The proposed changes to TS 5.6.5a, 5.6.5b and 3.3.1.1 are required to implement AREVA Analytical Methodologies.

   Therefore, the proposed TS changes will not significantly increase the consequences of an accident previously evaluated.

2. Does the proposed Technical Specification change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed deletion of a surveillance requirement to verify automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves does not introduce new equipment, which could create a new or different kind of accident.

   The proposed change to the ECCS Evaluation Model meets the requirements of 10 CFR 50.46(a)(1)(ii) and ensures the limits of 10 CFR 50.46(b) are maintained. The proposed changes to TS 5.6.5a, 5.6.5b and 3.3.1.1 are required to implement AREVA Analytical Methodologies.

   The proposed change does not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. The capability of credited SSCs to perform their required function will not be affected by the proposed change. In addition, the proposed change does not affect the interaction of plant SSCs with other plant SSCs whose failure or malfunction can initiate an accident or transient. As such, no new failure modes are being introduced. No new external threats, release pathways, or equipment failure modes are created. Therefore, the proposed deletion of a surveillance requirement to verify automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and

recirculation pump discharge valves will not create a possibility for an accident of a new or different type than those previously evaluated.

3. Does the proposed Technical Specification change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change to the ECCS Evaluation Model and the deletion of a surveillance requirement to verify automatic transfer capability for the power supply to the LPCI inboard injection valves, RHR minimum flow valves and recirculation pump discharge valves does not change the conditions, operating configurations, or minimum amount of operating equipment credited in the safety analyses for accident or transient mitigation.

   The proposed change does not alter the assumptions contained in the safety analyses. The proposed change does not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined.

   The proposed change does not impact the safety analysis-credited redundancy or availability of SSCs required for accident or transient mitigation, or the ability of the plant to cope with design basis events as assumed in safety analyses. In addition, no changes are proposed in the manner in which the credited SSCs provide plant protection or which create new modes of plant operation. The requirements of 10 CFR 50.46 and Appendix K continue to be met. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

   The proposed changes to TS 5.6.5a, 5.6.5b and 3.3.1.1 are required to implement AREVA Analytical Methodologies.

Based on the above, TVA concludes that the proposed TS changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (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 TS changes will not be inimical to the common defense and security or the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed TS changes would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed TS changes do not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed TS changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed TS changes.

6.0 REFERENCES

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

2. Letter from M. H. Chernoff (NRC) to K. Singer (TVA), "Browns Ferry Nuclear Plant, Unit 1 - Issuance of an Amendment Regarding Deletion of the Low Pressure Coolant Injection Motor-Generator Sets (TAC No. MC3822)(TS-427)," dated June 20, 2005.

3. Letter from T. E. Abney (TVA) to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Adoption of the General Electric (GE) SAFER/GESTR Loss of Coolant Accident Methodology," dated March 11, 1997.

4. Letter from T. E. Abney (TVA) to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Revision to Technical Specification (TS) Bases (TS-389)," dated April 24, 1997.
5. Letter from J. F. Williams (NRC) to O.D. Kingsley (TVA), "Browns Ferry Nuclear Plant Units 1, 2 and 3 - Revision to Technical Specification Bases (TAC Nos. M97911, M97912, M97913, M98695 and M98696) (TS 388 and TS 389)," dated July 8, 1997.

6. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Unit 1 - Issuance of Amendments Regarding the Transition to AREVA Fuel," dated April 27, 2012.

7. Letter from TVA to NRC, "Technical Specification Change TS-473, AREVA Fuel Transition," dated April 16, 2010

8. Letter from NRC to TVA, "Browns Ferry Nuclear Plant, Unit 1 -Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel (TAC NO. ME3775)," Request for Additional Information (RAI) Regarding TS-473, AREVA Fuel Transition (TAC No. ME3775)," ML110180585, dated August 23, 2011.

9. Letter from TVA to NRC, "Response to NRC Request for Additional Information Regarding Amendment Request to Transition to AREVA Fuel," dated October 7, 2011.

REVISED EVALUATION FOR TECHNICAL SPECIFICATION CHANGE TS-429
Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3

ATTACHMENT 1

Proposed Technical Specifications and Bases Page Markups for BFN, Unit 2

Technical Specifications Pages:

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

Technical Specifications Bases Pages:

B 3.2-5, B 3.2-5a, B 3.5-3, B 3.5-21, B 3.8-86, B 3.8-87a, B 3.8-93

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~~	~~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.~~	~~24 months~~

5.6 Reporting Requirements (continued)

5.6.4 (Deleted).

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

INSERT:
(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

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; ~~and~~

~~(4)~~ 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:

Insert 1

1. ~~NEDE 24011 P A, General Electric Standard Application for Reactor Fuel.~~

2. ~~XN NF 81 58(P)(A), RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model.~~

3. ~~XN NF 85 67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.~~

4. ~~EMF 85 74(P)(A), RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model.~~

5. ~~ANF 89 98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.~~

(continued)

5.6 Reporting Requirements (continued)

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

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

8. ~~EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.~~

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

10. ~~XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.~~

11. ~~ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.~~

12. ~~ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.~~

13. ~~ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.~~

14. ~~EMF-2209(P)(A), SPCB Critical Power Correlation.~~

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

16. ~~EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.~~

17. ~~EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.~~

(continued)

INSERT 1

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.

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.**

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 [*insert SE approval date*], 2012**

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.**

BASES (continued)

REFERENCES

1. ~~NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.~~

~~2.~~ FSAR, Chapter 3.

2. ~~3.~~ FSAR, Chapter 14.

3~~4~~. FSAR, Appendix N.

4~~5~~. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 5, January 2002.

5~~6~~. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

6~~7~~. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.

7~~8~~. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

8~~9~~. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

9~~10~~. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

10~~11~~. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," (**as supplemented by the site specific approval in NRC safety evaluation [insert SE approval date]), 2012.**~~as identified in the COLR).~~

1**1**~~2~~. EMF-2292(P)(A), "ATRIUM™ -10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

(continued)

BASES

REFERENCES (continued)

12~~3~~. XN-NF-81-58(P)(A)~~ Revision 2 and Supplements 1 and 2~~, **"RODEX2 Fuel Rod Thermal-** ~~"RODEX2 Fuel Rod Thermal-~~Mechanical Response Evaluation Model," **(as identified**~~E~~ **in the COLR)**~~xxon Nuclear Company, March 1984~~.

13. 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," **(as identified in the COLR)**~~Exxon Nuclear Company, March 1983~~.

14. 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," **(as identified in the COLR)**~~Exxon Nuclear Company, June 1986~~.

BASES

BACKGROUND (continued)

at 0.2 seconds when offsite power is available and B, C, and D pumps approximately 7, 14, and 21 seconds afterwards and if offsite power is not available all pumps 7 seconds after diesel generator power is available). When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop.

The two LPCI pumps and associated motor operated valves in each LPCI subsystem are powered from separate 4 kV shutdown boards. Both pumps in a LPCI subsystem inject water into the reactor vessel through a common inboard injection valve and depend on the closure of the recirculation pump discharge valve following a LPCI injection signal. Therefore, each LPCI subsystem's common inboard injection valve and recirculation pump discharge valve are powered from one of the two 4 kV shutdown boards associated with that subsystem. ~~The ability to provide power to the inboard injection valve and the recirculation pump discharge valve from two independent 4 kV shutdown boards ensures that a single failure of a diesel generator (DG) will not result in the failure of both LPCI pumps in one subsystem.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.11 (continued)

The Frequency of 24 months is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience with these components supports performance of the Surveillance at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

~~SR 3.5.1.12~~

~~Verification every 24 months of the automatic transfer capability between the normal and alternate power supply (480 V shutdown boards) for the RMOV boards which supply power for each LPCI subsystem inboard injection valve and each recirculation pump discharge valve demonstrates that AC electrical power is available to operate these valves following loss of power to one of the 4 kV shutdown boards. The ability to provide power to the inboard injection valve and the recirculation pump discharge valve from two independent 4 kV shutdown boards ensures that single failure of an EDG will not result in the failure of both LPCI pumps in one subsystem. Therefore, the failure of the automatic transfer capability will result in the inoperability of the affected LPCI subsystem. The 24 month Frequency has been found to be acceptable based on engineering judgment and operating experience.~~

(continued)

Revision

BASES (continued)

LCO

The required electrical power distribution subsystems listed in Table B 3.8.7-1 ensure the availability of AC and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an abnormal operational transient or a postulated DBA. The AC and DC electrical power distribution subsystems are required to be OPERABLE.

Maintaining the AC and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

The AC electrical power distribution subsystems require the associated buses and electrical circuits to be energized to their proper voltages. ~~In addition, for the D or E RMOV Boards to be OPERABLE, they must be able to auto transfer on loss of voltage. This feature ensures that the failure of one Diesel Generator will not result in the loss of an RHR subsystem.~~ OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger.

Based on the number of safety significant electrical loads associated with each board listed in Table B 3.8.7-1, if one or more of the boards becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.7 is required. Other boards, such as motor control centers (MCC) and distribution panels which help comprise the AC and DC distribution systems may not be listed in Table B 3.8.7-1. The loss of electrical loads associated with these boards may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should

(continued)

BASES

LCO (continued)

When 480 V Shutdown Board 2B is aligned to the alternate supply 4.16 kV Shutdown Board C, a LOCA/LOOP with a failure of the Shutdown Board D Battery would disable the normal supply 4.16 kV Shutdown Board D, and would also prevent the 480 V Shutdown Board 2B from load shedding its 480 V loads which would overload the alternate supply Diesel Generator D. This would result in the loss of diesel generators C and D, associated 4.16 kV shutdown boards and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 2B is on its alternate supply.

2A, 2B, 2D and 2E

The Unit 2 480 V RMOV boards ~~2A and 2B~~ have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 2 250 V DC RMOV boards 2A, 2B, and 2C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

(continued)

BASES

ACTIONS

B.1 (continued)

Pursuant to LCO 3.0.6, the Distribution System Action C would not be entered even if the 480 V shutdown board was inoperable, resulting in de-energization of a 480 V RMOV board. Therefore, the Required Actions of Condition B are modified by a Note to indicate that when Condition B is entered with no power source to 480 V RMOV board 2D or 2E, Action C must be immediately entered. This allows Condition B to provide requirements for the loss of the 480 V shutdown board without regard to whether 480 V RMOV board 2D or 2E is de-energized. Action C provides the appropriate restrictions for a de-energized 480 V RMOV board 2D or 2E.

C.1

~~480 V RMOV board 2D or 2E is inoperable if the automatic transfer capability between the normal and alternate power supply (LPCI MG sets) is inoperable for any reason. (Refer also to bases for SR 3.5.1.12.)~~

With 480 V RMOV Board D or E inoperable, the respective RHR subsystem supported by each affected board is inoperable for LPCI. The overall reliability is reduced because of the loss of one LPCI/RHR subsystem. In this condition, the remaining OPERABLE ECCS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. Therefore, the associated RHR subsystem must be declared inoperable immediately, and the actions in the appropriate system specification taken.

(continued)

REVISED EVALUATION FOR TECHNICAL SPECIFICATION CHANGE TS-429
Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3

ATTACHMENT 2

Proposed Technical Specifications and Bases Page Markups for BFN, Unit 3

Technical Specifications Pages:

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

Technical Specifications Bases Pages:

B 3.2-5, B 3.2-5a, B 3.5-3, B 3.5-21, B 3.8-86, B 3.8-88, B 3.8-94

Facility Operating License

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
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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~~	~~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.~~	~~24 months~~

5.6 Reporting Requirements (continued)

5.6.4 (Deleted).

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

INSERT:
(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

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; and

(4) 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.

5

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:

INSERT 1

~~1. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel.~~

~~2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.~~

~~3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.~~

~~4. EMF-85-74(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.~~

~~5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.~~

(continued)

5.6 Reporting Requirements (continued)

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

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

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11. ~~ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.~~

12. ~~ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.~~

13. ~~ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.~~

14. ~~EMF-2209(P)(A), SPCB Critical Power Correlation.~~

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

16. ~~EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.~~

17. ~~EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.~~

(continued)

INSERT 1

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.

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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.

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 [*insert SE approval date*], 2012

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.

BASES (continued)

~~1.~~ REFERENCES ~~1. NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.~~

1~~2~~. FSAR, Chapter 3.

2. ~~3.~~ FSAR, Chapter 14.

3~~4~~. FSAR, Appendix N.

4~~5~~. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 5, January 2002.

5~~6~~. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

6~~7~~. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.

7~~8~~. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

8. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

9. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

10. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," **(as supplemented by the site specific approval in NRC safety evaluation [insert SE approval date]), 2012**~~as identified in the COLR~~).

11. EMF-2292(P)(A), "ATRIUM™ -10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

(continued)

BASES

REFERENCES (continued)

12. XN-NF-81-58(P)(A)~~ Revision 2 and Supplements 1 and 2~~, **"RODEX2 Fuel Rod Thermal-** "~~RODEX2 Fuel Rod Thermal-~~Mechanical Response Evaluation Model," **(as identified in the COLR)**~~Exxon Nuclear Company, March 1984~~.

13. 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," **(as identified in the COLR)**~~Exxon Nuclear Company, March 1983~~.

14. 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," **(as identified in the COLR)**~~Exxon Nuclear Company, June 1986~~.

BASES

BACKGROUND (continued)

at 0.2 seconds when offsite power is available and B, C, and D pumps approximately 7, 14, and 21 seconds afterwards and if offsite power is not available all pumps 7 seconds after diesel generator power is available). When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop.

The two LPCI pumps and associated motor operated valves in each LPCI subsystem are powered from separate 4 kV shutdown boards. Both pumps in a LPCI subsystem inject water into the reactor vessel through a common inboard injection valve and depend on the closure of the recirculation pump discharge valve following a LPCI injection signal. Therefore, each LPCI subsystem's common inboard injection valve and recirculation pump discharge valve are powered from one of the two 4 kV shutdown boards associated with that subsystem. ~~The ability to provide power to the inboard injection valve and the recirculation pump discharge valve from two independent 4 kV shutdown boards ensures that a single failure of a diesel generator (DG) will not result in the failure of both LPCI pumps in one subsystem.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.11 (continued)

The Frequency of 24 months is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience with these components supports performance of the Surveillance at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

~~SR 3.5.1.12~~

~~Verification every 24 months of the automatic transfer capability between the normal and alternate power supply (480 V shutdown boards) for the RMOV boards which supply power for each LPCI subsystem inboard injection valve and each recirculation pump discharge valve demonstrates that AC electrical power is available to operate these valves following loss of power to one of the 4 kV shutdown boards. The ability to provide power to the inboard injection valve and the recirculation pump discharge valve from two independent 4 kV shutdown boards ensures that single failure of an EDG will not result in the failure of both LPCI pumps in one subsystem. Therefore, the failure of the automatic transfer capability will result in the inoperability of the affected LPCI subsystem. The 24 month Frequency has been found to be acceptable based on engineering judgment and operating experience.~~

(continued)

BASES (continued)

LCO

The required electrical power distribution subsystems listed in Table B 3.8.7-1 ensure the availability of AC and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an abnormal operational transient or a postulated DBA. The AC and DC electrical power distribution subsystems are required to be OPERABLE.

Maintaining the AC and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

The AC electrical power distribution subsystems require the associated buses and electrical circuits to be energized to their proper voltages. ~~In addition, for the D or E RMOV Boards to be OPERABLE, they must be able to auto transfer on loss of voltage. This feature ensures that the failure of one Diesel Generator will not result in the loss of an RHR subsystem.~~ OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger.

Based on the number of safety significant electrical loads associated with each board listed in Table B 3.8.7-1, if one or more of the boards becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.7 is required. Other boards, such as motor control centers (MCC) and distribution panels which help comprise the AC and DC distribution systems may not be listed in Table B 3.8.7-1. The loss of electrical loads associated with these boards may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should

(continued)

BASES

LCO (continued)

generators 3A and 3C, associated 4.16 kV shutdown boards, and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 3A is on its alternate supply.

The Unit 3 diesel auxiliary boards have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

3A, 3B, 3D and 3E

The Unit 3 480 V RMOV boards ~~3A and 3B~~ have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 3 250 V DC RMOV boards 3A, 3B, and 3C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

(continued)

BASES

ACTIONS

B.1 (continued)

Pursuant to LCO 3.0.6, the Distribution System Action C would not be entered even if the 480 V shutdown board was inoperable, resulting in de-energization of a 480 V RMOV board. Therefore, the Required Actions of Condition B are modified by a Note to indicate that when Condition B is entered with no power source to 480 V RMOV board 3D or 3E, Action C must be immediately entered. This allows Condition B to provide requirements for the loss of the 480 V shutdown board without regard to whether 480 V RMOV board 3D or 3E is de-energized. Action C provides the appropriate restrictions for a de-energized 480 V RMOV board 3D or 3E.

C.1

~~480 V RMOV board 3D or 3E is inoperable if the automatic transfer capability between the normal and alternate power supply (LPCI MG sets) is inoperable for any reason. (Refer also to bases for SR 3.5.1.12.)~~

With 480 V RMOV Board D or E inoperable, the respective RHR subsystem supported by each affected board is inoperable for LPCI. The overall reliability is reduced because of the loss of one LPCI/RHR subsystem. In this condition, the remaining OPERABLE ECCS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. Therefore, the associated RHR subsystem must be declared inoperable immediately, and the actions in the appropriate system specification taken.

(continued)

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 within 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.

Insert attached License Condition (d)

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.

New License Condition:

(d) For License Amendment XXX, 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.

REVISED EVALUATION FOR TECHNICAL SPECIFICATION CHANGE TS-429
Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3

ATTACHMENT 3

Retyped Proposed Technical Specifications and Bases Pages for BFN, Unit 2

Technical Specifications Pages:

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

Technical Specifications Bases Pages:

B 3.2-5, B 3.2-5a, B 3.5-3, B 3.5-21, B 3.8-86, B 3.8-87a, B 3.8-93

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.

(continued)

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 [*insert SE approval date*], 2012

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)

BASES (continued)

REFERENCES

1. FSAR, Chapter 3.
2. FSAR, Chapter 14.
3. FSAR, Appendix N.
4. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 5, January 2002.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
6. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
7. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
8. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
9. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.
10. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," (as supplemented by the site specific approval in NRC safety evaluation [insert SE approval date]), 2012.
11. EMF-2292(P)(A), "ATRIUM™ -10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

(continued)

BASES

REFERENCES (continued)	12. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," (as identified in the COLR).
	13. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).
	14. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).

BACKGROUND (continued)

at 0.2 seconds when offsite power is available and B, C, and D pumps approximately 7, 14, and 21 seconds afterwards and if offsite power is not available all pumps 7 seconds after diesel generator power is available). When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop.

The two LPCI pumps and associated motor operated valves in each LPCI subsystem are powered from separate 4 kV shutdown boards. Both pumps in a LPCI subsystem inject water into the reactor vessel through a common inboard injection valve and depend on the closure of the recirculation pump discharge valve following a LPCI injection signal. Therefore, each LPCI subsystem's common inboard injection valve and recirculation pump discharge valve are powered from one of the two 4 kV shutdown boards associated with that subsystem.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.11 (continued)

The Frequency of 24 months is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience with these components supports performance of the Surveillance at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

LCO

The required electrical power distribution subsystems listed in Table B 3.8.7-1 ensure the availability of AC and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an abnormal operational transient or a postulated DBA. The AC and DC electrical power distribution subsystems are required to be OPERABLE.

Maintaining the AC and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

The AC electrical power distribution subsystems require the associated buses and electrical circuits to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger.

Based on the number of safety significant electrical loads associated with each board listed in Table B 3.8.7-1, if one or more of the boards becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.7 is required. Other boards, such as motor control centers (MCC) and distribution panels which help comprise the AC and DC distribution systems may not be listed in Table B 3.8.7-1. The loss of electrical loads associated with these boards may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should

(continued)

BASES

LCO (continued)

When 480 V Shutdown Board 2B is aligned to the alternate supply 4.16 kV Shutdown Board C, a LOCA/LOOP with a failure of the Shutdown Board D Battery would disable the normal supply 4.16 kV Shutdown Board D, and would also prevent the 480 V Shutdown Board 2B from load shedding its 480 V loads which would overload the alternate supply Diesel Generator D. This would result in the loss of diesel generators C and D, associated 4.16 kV shutdown boards and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 2B is on its alternate supply.

The Unit 2 480 V RMOV boards 2A, 2B, 2D and 2E have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 2 250 V DC RMOV boards 2A, 2B, and 2C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

(continued)

BASES

ACTIONS

B.1 (continued)

Pursuant to LCO 3.0.6, the Distribution System Action C would not be entered even if the 480 V shutdown board was inoperable, resulting in de-energization of a 480 V RMOV board. Therefore, the Required Actions of Condition B are modified by a Note to indicate that when Condition B is entered with no power source to 480 V RMOV board 2D or 2E, Action C must be immediately entered. This allows Condition B to provide requirements for the loss of the 480 V shutdown board without regard to whether 480 V RMOV board 2D or 2E is de-energized. Action C provides the appropriate restrictions for a de-energized 480 V RMOV board 2D or 2E.

C.1

With 480 V RMOV Board D or E inoperable, the respective RHR subsystem supported by each affected board is inoperable for LPCI. The overall reliability is reduced because of the loss of one LPCI/RHR subsystem. In this condition, the remaining OPERABLE ECCS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. Therefore, the associated RHR subsystem must be declared inoperable immediately, and the actions in the appropriate system specification taken.

(continued)

REVISED EVALUATION FOR TECHNICAL SPECIFICATION CHANGE TS-429
Deletion of Low Pressure Coolant Injection Motor-Generator Sets for
Browns Ferry Nuclear Plant, Units 2 and 3

ATTACHMENT 4

Retyped Proposed Technical Specifications and Bases Pages for BFN, Unit 3

Technical Specifications Pages:

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

Technical Specifications Bases Pages:

B 3.2-5, B 3.2-5a, B 3.5-3, B 3.5-21, B 3.8-86, B 3.8-88, B 3.8-94

Facility Operating License

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.

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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 [*insert SE approval date*], 2012

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)

BASES (continued)

REFERENCES

1. FSAR, Chapter 3.

2. FSAR, Chapter 14.

3. FSAR, Appendix N.

4. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 5, January 2002.

5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

6. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.

7. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

8. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

9. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

10. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," (as supplemented by the site specific approval in NRC safety evaluation [insert SE approval date]), 2012.

11. EMF-2292(P)(A), "ATRIUM™ -10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

(continued)

BASES

REFERENCES (continued)

12. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-"Mechanical Response Evaluation Model," (as identified in the COLR).

13. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).

14. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).

BASES

BACKGROUND (continued)

at 0.2 seconds when offsite power is available and B, C, and D pumps approximately 7, 14, and 21 seconds afterwards and if offsite power is not available all pumps 7 seconds after diesel generator power is available). When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop.

The two LPCI pumps and associated motor operated valves in each LPCI subsystem are powered from separate 4 kV shutdown boards. Both pumps in a LPCI subsystem inject water into the reactor vessel through a common inboard injection valve and depend on the closure of the recirculation pump discharge valve following a LPCI injection signal. Therefore, each LPCI subsystem's common inboard injection valve and recirculation pump discharge valve are powered from one of the two 4 kV shutdown boards associated with that subsystem.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.11 (continued)

The Frequency of 24 months is based on the need to perform the Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience with these components supports performance of the Surveillance at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

LCO

The required electrical power distribution subsystems listed in Table B 3.8.7-1 ensure the availability of AC and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an abnormal operational transient or a postulated DBA. The AC and DC electrical power distribution subsystems are required to be OPERABLE.

Maintaining the AC and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of ESF is not defeated. Therefore, a single failure within any system or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

The AC electrical power distribution subsystems require the associated buses and electrical circuits to be energized to their proper voltages. OPERABLE DC electrical power distribution subsystems require the associated buses to be energized to their proper voltage from either the associated battery or charger.

Based on the number of safety significant electrical loads associated with each board listed in Table B 3.8.7-1, if one or more of the boards becomes inoperable, entry into the appropriate ACTIONS of LCO 3.8.7 is required. Other boards, such as motor control centers (MCC) and distribution panels which help comprise the AC and DC distribution systems may not be listed in Table B 3.8.7-1. The loss of electrical loads associated with these boards may not result in a complete loss of a redundant safety function necessary to shut down the reactor and maintain it in a safe condition. Therefore, should

(continued)

BASES

LCO (continued)

generators 3A and 3C, associated 4.16 kV shutdown boards, and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 3A is on its alternate supply.

The Unit 3 diesel auxiliary boards have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 3 480 V RMOV boards 3A, 3B, 3D and 3E have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 3 250 V DC RMOV boards 3A, 3B, and 3C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

(continued)

BASES

ACTIONS

B.1 (continued)

Pursuant to LCO 3.0.6, the Distribution System Action C would not be entered even if the 480 V shutdown board was inoperable, resulting in de-energization of a 480 V RMOV board. Therefore, the Required Actions of Condition B are modified by a Note to indicate that when Condition B is entered with no power source to 480 V RMOV board 3D or 3E, Action C must be immediately entered. This allows Condition B to provide requirements for the loss of the 480 V shutdown board without regard to whether 480 V RMOV board 3D or 3E is de-energized. Action C provides the appropriate restrictions for a de-energized 480 V RMOV board 3D or 3E.

C.1

With 480 V RMOV Board D or E inoperable, the respective RHR subsystem supported by each affected board is inoperable for LPCI. The overall reliability is reduced because of the loss of one LPCI/RHR subsystem. In this condition, the remaining OPERABLE ECCS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. Therefore, the associated RHR subsystem must be declared inoperable immediately, and the actions in the appropriate system specification taken.

(continued)

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 within 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 XXX, the licensee shall implement changes to BFN Unit 3 Technical Specifications 3.3.1.1 and 5.6.5 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.

REVISED EVALUATION FOR TECHNICAL SPECIFICATION CHANGE TS-429
Deletion of Low Pressure Coolant Injection Motor-Generator Sets for Browns Ferry Nuclear Plant, Units 2 and 3

ATTACHMENT 5

ANP-3015(P), Revision 0, "Browns Ferry Units 1, 2, and 3, LOCA Break Spectrum Analysis", Proprietary