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

April 26, 2016

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNIT 1 – ISSUANCE OF AMENDMENT TO REVISE TECHNICAL SPECIFICATIONS RELATED TO CYCLE 12 SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (CAC NO. MF6760)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 295 to Renewed Facility Operating License No. DPR-33 for the Browns Ferry Nuclear Plant, Unit 1. This amendment is in response to your application dated September 25, 2015, as supplemented by letters dated December 28, 2015, and March 28, 2016.

This amendment revises the Technical Specification (TS) to change the Safety Limit Minimum Critical Power Ratio (SLMCPR) numeric values for Browns Ferry Nuclear Plant, Unit 1. The change decreases the numeric values of SLMCPR in TS Section 2.1.1.2 for single and two reactor recirculation loop operation based on the Cycle 12 SLMCPR evaluation.

The NRC staff has completed its review of the information provided by the licensee. The NRC staff's safety evaluation (SE) is enclosed. The NRC staff has determined that the enclosed SE (Enclosure 2) does not contain proprietary or other sensitive information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390, "Public inspections, exemptions, requests for withholding." However, the NRC will delay placing the enclosed SE in the public document room for a period of 10 working days from the date of this letter, to provide Tennessee Valley Authority the opportunity to comment on any sensitive aspects of the SE. If you believe that any information in Enclosure 2 contains sensitive information, please identify such information line-by-line and define the basis for withholding pursuant to the criteria of 10 CFR 2.390. After 10 working days, the enclosed SE will be made publicly available unless you notify us that it contains sensitive information.

The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions concerning this letter and the SE, contact me at 301-415-1447 or by E-mail at Farideh.Saba@nrc.gov.

Sincerely,

Farideh E Saba

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

Docket No. 50-259

Enclosures:
1. Amendment No. 295 to DPR-33
2. Safety Evaluation

cc w/enclosures **10 working days after issuance: Distribution via Listserv**

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 295
Renewed License No. DPR-33

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

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated September 25, 2015, as supplemented by letters dated December 28, 2015, and March 28, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 295, are hereby incorporated in this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and shall be implemented during the Unit 1 refueling outage in the fall of 2016.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-33
and Technical Specifications

Date of Issuance: April 26, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 295

RENEWED FACILITY OPERATING LICENSE NO. DPR- 33

DOCKET NO. 50-259

Replace the following page of the Renewed Facility Operating License No. DPR-33 with the attached revised page. The changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

Replace the following page of Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
Page 2.0-1	Page 2.0-1

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

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

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

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

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 295, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 234 to Facility Operating License DPR-33, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 234. For SRs that existed prior to Amendment 234, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 234.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.06 for two recirculation loop operation or ≥ 1.08 for single loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 295 TO FACILITY OPERATING LICENSE NO. DPR-33

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-259

1.0 INTRODUCTION

By letter dated September 25, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15268A566) (Reference 1), as supplemented by letters dated December 28, 2015 (ADAMS Accession No. ML15363A426) (Reference 2) and March 28, 2016 (ADAMS Accession No. ML16088A230) (Reference 10), Tennessee Valley Authority (TVA, the licensee) submitted a request for changes to the Browns Ferry Nuclear Plant Unit 1 (BFN Unit 1) Technical Specifications (TSs). The amendment revises TS 2.0, "Safety Limits (SLs)," by changing the Safety Limit Minimum Critical Power Ratio (SLMCPR) for both single and dual recirculation loop operation.

The supplements dated December 28, 2015, and March 28, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on January 5, 2016 (81 FR 276).

2.0 REGULATORY EVALUATION

Fuel design limits can be exceeded if the fuel produces heat equal to or greater than critical power. In a boiling-water reactor (BWR), heat produced by the fuel causes the water to partially vaporize in a stable process called nucleate boiling. As the amount of heat produced by the fuel increases, more of the water vaporizes and the vapor production changes the way the water boils. At a certain point, the efficiency of heat removal is impeded by vapor production and the temperature of the fuel cladding rises disproportionately to the heat generated. Critical power is a term used for the power at which the fuel departs from nucleate boiling and enters a transition to film boiling. For BWRs, the critical power may be predicted using a correlation of experimental data. BFN Unit 1 uses some correlations from AREVA Incorporated such as the ACE/ATRIUM 10XM Critical Power Correlation or the SPCB (Siemens Power Corporation Boilers) Critical Power Correlation. Due to core wide and operational variations, the margin to boiling transition is most easily described in terms of a critical power ratio (CPR), which is defined as the rod critical power as calculated by the correlation divided by the actual rod power. The more a CPR value exceeds 1.0, the greater is the margin to boiling transition. The

SLMCPR is calculated using a statistical process that takes into account operating parameters and uncertainties. The operating limit minimum CPR (OLMCPR) is equal to the SLMCPR plus a CPR margin for transients. At the OLMCPR, at least 99.9 percent of the rods avoid boiling transition during steady state operation and transients (Section 4.4, "Thermal and Hydraulic Design," of NUREG-0800, Revision 3, dated June 1996) caused by a single operator error or equipment malfunction.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 36(c)(1) requires that SLs are included in the TSs. The SLMCPR is verified on a cycle-specific basis, because it is necessary to account for the core configuration-specific neutronic and thermal-hydraulic response.

General Design Criterion (GDC) 10, in 10 CFR, Part 50, Appendix A, states, in part, that the reactor core and associated coolant, control, and protection systems shall be designed to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The following explains the use of GDC in Appendix A of 10 CFR Part 50 for BFN Unit 1. The construction permit for BFN Unit 1 was issued by the Atomic Energy Commission (AEC) on May 10, 1967, based on draft GDC published in July 1967 (Draft 70), and the operating license was issued on December 20, 1973. At the time of issuance of the operating license, the AEC had published the final rule that added Appendix A "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, in the FR on February 20, 1971 (36 FR 3255), with the rule effective on May 21, 1971. Consequently, as discussed in Section B2.2, "Safety Limit," of BFN Unit 1 TS Bases document, GDC 10 is used to evaluate the licensing basis for Reactor Core SLs of the plant.

The licensee's submittal dated September 25, 2015, states that, "The proposed change in the SLMCPR values in TS 2.1.1.2 complies with the requirements of GDC 10 and will continue to assure that fuel clad integrity is maintained." Therefore, the Nuclear Regulatory Commission (NRC) staff reviews amendments to the BFN Unit 1 license using the 10 CFR Part 50 Appendix A, GDC 10.

3.0 TECHNICAL EVALUATION

3.1 BFN Unit 1 Cycle 12 Core

BFN Unit 1 is a BWR/4 that has two recirculation loops. The licensee proposed to change the SLMCPR values in TS 2.1.1.2 from 1.09 to 1.06 for two-recirculation-loop operation, and from 1.11 to 1.08 for single-recirculation-loop operation.

The BFN Unit 1 Cycle 12 core loading will consist of 280 ATRIUM 10XM fuel bundles and 484 ATRIUM-10 fuel bundles in the core. Both fuel design types are manufactured by AREVA, Inc. This cycle will be the first time that the ATRIUM 10XM fuel design type is loaded in BFN Unit 1. The ATRIUM-10 fuel is all burned fuel initially loaded in Cycles 10 and 11.

Reference 4 documented NRC approval of the updated AREVA methodologies referenced in the License Amendment Request (LAR) for use at BFN Units 1, 2, and 3. At that time, the NRC

staff also approved a change to the SLMCPR values documented in the BFN Unit 2 TSs. After the BFN Unit 1 Cycle 12 core design was developed, a cycle specific SLMCPR analysis was performed using the updated methodologies. Consequently, the licensee is requesting a change to the BFN Unit 1 TSs to reduce the required SLMCPR to values that bound the Cycle 12 results.

3.2 Methodology

The licensee developed the BFN Unit 1 Cycle 12 SLMCPR values using the following NRC-approved methodologies and uncertainties:

- ANP-10307PA, Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors" (Reference 5)
- ANP-10298PA, Revision 0, "ACE/ATRIUM 10XM Critical Power Correlation" (Reference 6)
- ANP-3140(P), Revision 0, "Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation" (Reference 7)
- EMF-2209(P)(A), Revision 0, "SPCB Critical Power Correlation" (Reference 8)

A new approved revision to ANP-10298PA (Revision 1), was published on March 31, 2014, to address deficiencies in the K-factor calculational methodology. In this license amendment request, consistent with TS 5.6.5, the licensee references ANP-10298PA (Revision 0) for this review. The licensee has not adopted ANP-10298PA (Revision 1), as the K-factors for BFN Unit 1 were computed using the methodology in Reference 7, which also addresses the aforementioned deficiency.

Plant specific use of these methodologies must adhere to certain restrictions.

3.2.1 Methodology Restrictions

Based on the review (Reference 4) of the SAFLIM3D methodology for BFN Units 1, 2, and 3, the NRC staff identified the following restriction for the use of this methodology for SLMCPR calculations with BFN:

1. The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, Revision 0" (i.e., TS 5.6.5.b.11) to determine the SLMCPR shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined.

In addition, prior Safety Evaluations (SEs) of the topical reports describing the CPR correlations discussed the following limitations and conditions:

2. The ACE/ATRIUM 10XM methodology may only be used to perform evaluations of AREVA ATRIUM 10XM fuel design. The ACE/ATRIUM 10XM correlation may also be used to evaluate the performance of the co-resident fuel in mixed cores as discussed in Section 3.5 of the SE associated with Reference 6.

3. ACE/ATRIUM 10XM correlation shall not be used outside the range of applicability defined by the range of the test data prescribed in Table 2.1 of Reference 6.

Restriction 1

The licensee provided the SLMCPR analysis report for the proposed BFN Unit 1 Cycle 12 core. The analysis report confirms that the calculated fast fluence gradients for the aforementioned core design were compared to the upper and lower bounds of the channel bow database. All fuel assemblies for which the fluence gradients were calculated to be outside the bounds of the measurement database were augmented as described by the restriction, in a manner consistent with what was done for BFN Unit 2 (Reference 9).

As a result of the review of the licensee's submittals, the NRC staff finds that the licensee has appropriately addressed this restriction imposed on application of this SLMCPR methodology at BFN Unit 1.

Restriction 2

The BFN Unit 1 Cycle 12 core loading will consist of both ATRIUM-10 and ATRIUM 10XM fuel. The SLMCPR analysis report states that the ACE/ATRIUM 10XM correlation is only used for the ATRIUM 10XM fuel. The SPCB correlation is used for the ATRIUM-10 fuel, consistent with Reference 8, as required by TS 5.6.5.b.14.

Based on tits review of the licensee's report, the NRC staff finds that this restriction is satisfied.

Restriction 3

Most of the parameters listed in Table 2.1 of Reference 6 relate to core characteristics that do not change as a result of differences in nuclear design, and were previously addressed for the ATRIUM-10 and ATRIUM 10XM fuel assembly designs. One parameter that does vary from core loading to core loading is the maximum local peaking factor. The licensee provided information in Reference 2 that demonstrated that the BFN Unit 1 Cycle 12 core remains within the range of applicability provided by Table 2.1 of Reference 6, including the maximum peaking factor.

Based on the information provided by the licensee, the NRC staff finds that the licensee has demonstrated that this restriction is satisfied.

3.3 Precedents

As a precedent, the licensee cited Reference 4, which includes a similar SLMCPR LAR for Unit 2 at the same site. This reference documents NRC review and approval of the use of the methodology with ATRIUM 10XM fuel for all three units at BFN, as well as cycle-specific

analyses performed for Unit 2 resulting in a change to the TS values. As part of the evaluation of this LAR dated September 25, 2015, the NRC staff evaluated the use of the updated methodology with both fuel designs planned for use in the BFN Unit 1 Cycle 12 core, ATRIUM-10 and ATRIUM 10XM. Reference 4 and its supporting references were reviewed to confirm that the precedent was applicable to BFN Unit 1 Cycle 12. Except for the specific batch sizes, uranium and gadolinia loadings in the fuel rods, cycle-specific operating histories, and unit-specific uncertainty values, the application of the methodology at both units covers a similar range of plant parameters and operating configurations. The mentioned differences are not expected to affect the applicability of the use of this methodology with the stated fuel assembly designs, though they may affect the calculated cycle-specific SLMCPR values.

As a result of a review of the referenced SE and supporting documentation, the NRC staff has determined that the cited precedent satisfactorily demonstrates the applicability of the AREVA methodologies for analysis of the BFN Unit 1 Cycle 12 core.

3.4 Departures from NRC-Approved Methodology

No departures from NRC-approved methodologies were identified in the BFN Unit 1 Cycle 12 SLMCPR calculations.

3.5 Assembly Flow Rate Uncertainty

In the SLMCPR analysis report submitted by the licensee for BFN Unit 1 Cycle 12, the provided value for the assembly flow rate uncertainty was slightly lower than the value provided for the Unit 2 precedent. ANP-10307PA, Revision 0, requires a higher value for cores containing fuel not loaded by AREVA (Reference 5). Even though the Unit 2 precedent was for a core that only had AREVA fuel, use of the higher value was conservative and acceptable. The BFN Unit 1 Cycle 12 evaluation used the appropriate value for cores consisting solely of AREVA fuel.

Thus, the NRC staff finds that the use of the lower assembly flow rate uncertainty is acceptable for use in the BFN Unit 1 SLMCPR analyses.

3.6 Core Monitoring System

For BFN Unit 1 Cycle 12, the AREVA POWERPLEX core monitoring system will be used as the core monitoring system. The POWERPLEX system has been in use at BFN since 2004. POWERPLEX utilizes MCPR operating limits generated by approved licensing analysis methods. POWERPLEX uses the approved methodologies in Reference 5 to determine specific MCPR for each assembly for comparison to the licensing MCPR operating limit to indirectly verify the TS SLMCPR is not violated.

Therefore, the NRC staff finds the use of the AREVA POWERPLEX system for BFN Unit 1 Cycle 12 to be acceptable.

3.7 Technical Evaluation Conclusion

The NRC staff finds the licensee's proposed BFN Unit 1 SLMCPR values of 1.06 for two-recirculation-loop operation and 1.08 for single-recirculation-loop operation acceptable for

BFN Unit 1 Cycle 12 because they bound SLMCPR values for Cycle 12 that were determined through the appropriate use of NRC-approved methodologies in accordance with NRC staff guidelines. The staff further finds that the licensee used methods consistent with the regulatory requirements and guidance identified in Section 2.0 above.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the FR on January 5, 2016 (81 FR 276). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from J. W. Shea (TVA) to U.S. NRC, "Browns Ferry Nuclear Plant (BFN), Unit 1 – Application to Modify Technical Specifications 2.1.1.2, Reactor Core Minimum Critical Power Ratio Safety Limits (TS-506)," September 25, 2015. (ADAMS Accession No. ML15268A566)

2. Letter from J. W. Shea (TVA) to U.S. NRC, "Response to NRC Request for Additional Information Browns Ferry Nuclear Plant (BFN), Unit 1 – Application to Modify Technical Specification 2.1.1.2, Reactor Core Minimum Critical Power Ratio Safety Limits (TS-506) (TAC No. MF6760)," December 28, 2015. (ADAMS Accession No. ML15363A426)

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 4.4, Revision 2, March 2007. (ADAMS Accession No. ML070550060)

4. Letter from Farideh E. Saba (U.S. NRC) to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 – Issuance of Amendments Regarding Technical Specification (TS) Change TS-478 Addition of Analytical Methodologies to TS 5.6.5 and Revision of TS 2.1.1.2 for Unit 2 (TAC Nos. MF0877, MF0878, and MF0879), July 31, 2014. (ADAMS Accession No. ML14113A286)

5. AREVA Nuclear Power Licensing Topical Report ANP-10307PA, Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," June 2011. (ADAMS Accession No. ML11259A021)

6. AREVA Nuclear Power Licensing Topical Report ANP-10298PA, Revision 0, "ACE/ATRIUM 10XM Critical Power Correlation," March 2010. (ADAMS Accession No. ML101190044)

7. AREVA Nuclear Power Licensing Topical Report ANP-3140(P), Revision 0, "Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation," August 2012. (ADAMS Accession No. ML13070A324)

8. AREVA Nuclear Power Licensing Topical Report EMF-2209(P)(A), Revision 3, "SPCB Critical Power Correlation," September 2009. (ADAMS Accession No. ML093650235)

9. AREVA Report ANP-3248P, Revision 1, Enclosure 2 to Letter from J. W. Shea (TVA) to U.S. NRC, "Response to Request for Additional Information on Technical Specification Change TS-478," September 30, 2013. (ADAMS Accession No. ML13276A062)

10. Letter from J. W. Shea (TVA) to U.S. NRC, "Browns Ferry, Unit 1 - Update to License Amendment Request, Application to Modify Technical Specification 2.1.1.2, Reactor Core Minimum Critical Power Ratio Safety Limits (TS-506)," March 28, 2016. (ADAMS Accession No. ML16088A230)

Principal Contributor: William MacFee

If you have any questions concerning this letter and the SE, contact me at 301-415-1447 or by E-mail at Farideh.Saba@nrc.gov.

Sincerely,

/RA/

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

Docket No. 50-259

Enclosures:
1. Amendment No. 295 to DPR-33
2. Safety Evaluation

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