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

January 13, 2023

Mr. James Barstow
Vice President, Nuclear Regulatory
Affairs and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 325, 348, AND 308 REGARDING APPLICATION OF ADVANCED FRAMATOME METHODOLOGIES, AND ADOPTION OF TSTF-564-A, REVISION 2, "SAFETY LIMIT MCPR," IN SUPPORT OF ATRIUM 11 FUEL USE (EPID L-2021-LLA-0132)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 325, 348, and 308 to Renewed Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant (Browns Ferry), Units 1, 2, and 3, respectively. These amendments are in response to your application dated July 23, 2021, as supplemented by letters dated April 8, May 27, July 18, July 28, September 13, September 29, and December 9, 2022.

The amendments revise Browns Ferry Technical Specification (TS) 5.6.5.b, "Core Operating Limits Report," to allow application of Advanced Framatome Methodologies for determining core operating limits in support of loading Framatome fuel type ATRIUM 11™ under the currently licensed Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain. The amendments also delete Note (f) from TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation," which no longer applies. The amendments also delete a plant-specific report previously required at the time ATRIUM 10XM fuel was approved for use that is now no longer needed. Additionally, the amendments revise the safety limit in TS 2.1.1.2 for the minimum critical power ratio (MCPR) based on Technical Specification Task Force (TSTF) Traveler TSTF-564-A, Revision 2, "Safety Limit MCPR" (SLMCPR). Lastly, the amendments revise TS 5.6.5 to require the SLMCPR value be included in the core operating limits report.

The NRC staff has completed its review of the information provided by TVA. Enclosure 4 provides the staff's safety evaluation (SE). The staff has determined that it contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Accordingly, the NRC staff has prepared a redacted, non-proprietary version (Enclosure 5). The NRC staff will delay placing the non-proprietary SE in ADAMS and the public document room for a period of 10 working days from the date of this letter to allow you to comment on any proprietary aspects. If you believe that any information in Enclosure 4 is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10 working days, the non-proprietary SE will be made publicly available.

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

Sincerely,

/RA/

Kimberly J. Green, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

Enclosures:
1. Amendment No. 325 to DPR-33
2. Amendment No. 348 to DPR-52
3. Amendment No. 308 to DPR-68
4. Proprietary Safety Evaluation
5. Non-Proprietary Safety Evaluation

cc: Listserv w/Enclosures 1, 2, 3, and 5 (10 working days after issuance)

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 325
Renewed License No. DPR-33

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

   A. The application for amendment by the Tennessee Valley Authority (the licensee) dated July 23, 2021, as supplemented by letters dated April 8, May 27, July 18, July 28, September 13, September 29, and December 9, 2022, 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 Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

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

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

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

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

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

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 325, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from the Unit 1 Refueling 15 outage in fall 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License
and Technical Specifications

Date of Issuance: January 13, 2023

ATTACHMENT TO LICENSE AMENDMENT NO. 325

BROWNS FERRY NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace page 3 of Renewed Facility Operating License No. DPR-33 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the areas of change.

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

Remove	Insert
2.0-1	2.0-1
3.3-7	3.3-7
5.0-24	5.0-24
5.0-24a	5.0-24a
5.0-24b	5.0-24b
5.0-24c	5.0-24c

(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) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3952 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 325, 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 ≤ 23% RTP.

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

MCPR shall be ≥ 1.05.

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.

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	≥ 18%	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	NA
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. High Drywell Pressure	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, he 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 documenta ion shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

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 MINIMUM CRITICAL POWER RATIO (MCPR) and $MCPR_{99.9\%}$ for Specification 3.2.2;

(4) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased Simulated Thermal Power-High Scram setpoints used in the Automated BSP Scram Region, and the BSP Boundary 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. (Deleted).

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

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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

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

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

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

10. (Deleted).

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

11. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.

12. (Deleted).

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

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

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

16. EMF-2361(P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP Inc., May 2001 as supplemented by the site-specific approval in NRC safety evaluation dated April 27, 2012, and July 31, 2014.

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

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

19. (Deleted).

20. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

21. ANP-10298P-A Revision 1, ACE/ATRIUM 10XM Critical Power Correlation, AREVA Inc., March 2014.

22. (Deleted).

23. NEDC-33075P-A, GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, Revision 8, November 2013.

24. BAW-10247P-A Supplement 2P-A Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods, Framatome Inc., August 2018.

25. ANP-10300P-A Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome Inc., January 2018.

26. ANP-10332P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios, Framatome Inc., March 2019

27. ANP-10333P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Framatome Inc., March 2018 (as supplemented by Section 6.4 of ANP-3908P Revision 4, Applicability of Framatome BWR Methods to Browns Ferry with ATRIUM 11 Fuel, Framatome Inc., June 2022).

28. ANP-10335P-A Revision 0, ACE/ATRIUM 11 Critical Power Correlation, Framatome Inc., May 2018.

29. ANP-10340P-A, Revsion 0, Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods, Framatome Inc., May 2018.

30. ANP-3907P Revision 0, Application of BEO-III Methodology with the Confirmation Density Algorithm at Browns Ferry, Framatome Inc., April 2021.

(continued)

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 348
Renewed License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 23, 2021, as supplemented by letters dated April 8, May 27, July 18, July 28, September 13, September 29, and December 9, 2022, 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 Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-52 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 348, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from the Unit 2 Refueling 22 outage in spring 2023.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License
and Technical Specifications

Date of Issuance: January 13, 2023

ATTACHMENT TO LICENSE AMENDMENT NO. 348

BROWNS FERRY NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace page 3 of Renewed Facility Operating License No. DPR-52 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the areas of change.

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

Remove	Insert
2.0-1	2.0-1
3.3-8	3.3-8
5.0-24	5.0-24
5.0-24a	5.0-24a
5.0-24b	5.0-24b
--	5.0-24c

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 3952 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 348, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

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

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 ≤ 23% RTP.

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

MCPR shall be ≥ 1.05.

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.

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	≥ 18%	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	NA
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.

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 MINIMUM CRITICAL POWER RATIO (MCPR) and $MCPR_{99.9\%}$ for Specification 3.2.2;

(4) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased Simulated Thermal Power-High Scram setpoints used in the Automated BSP Scram Region, and the BSP Boundary 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)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (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/MICROBURN-B2, 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. (Deleted.)

10. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.

11. (Deleted).

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.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

15. EMF-2361(P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP Inc., May 2001 as supplemented by the site-specific approval in NRC safety evaluation dated February 15, 2013 and July 31, 2014.

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. (Deleted.)

19. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

20. ANP-10298P-A Revision 1, ACE/ATRIUM 10XM Critical Power Correlation, AREVA Inc., March 2014.

21. (Deleted.)

22. NEDC-33075P-A, GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, Revision 8, November 2013.

23. BAW-10247P-A Supplement 2P-A Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods, Framatome Inc., August 2018.

24. ANP-10300P-A Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome Inc., January, 2018.

25. ANP-10332P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios, Framatome Inc., March 2019.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

26. ANP-10333P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Framatome Inc., March 2018 (as supplemented by Section 6.4 of ANP-3908P Revision 4, Applicability of Framatome BWR Methods to Browns Ferry with ATRIUM 11 Fuel, Framatome Inc., June 2022).

27. ANP-10335P-A Revision 0, ACE/ATRIUM 11 Critical Power Correlation, Framatome Inc., May 2018.

28. ANP-10340P-A Revision 0, Incorporation of Chromia Doped Fuel Properties in AREVA Approved Methods, Framatome Inc., May 2018.

29. ANP-3907P Revision 0, Application of BEO-III Methodology with the Confirmation Density Algorithm at Browns Ferry, Framatome Inc., April 2021.

(continued)

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 308
Renewed License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 23, 2021, as supplemented by letters dated April 8, May 27, July 18, July 28, September 13, September 29, and December 9, 2022, 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 Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-68 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 308, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from the Unit 3 Refueling 21 outage in spring 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License
and Technical Specifications

Date of Issuance: January 13, 2023

ATTACHMENT TO LICENSE AMENDMENT NO. 308

BROWNS FERRY NUCLEAR PLANT, UNIT 3

RENEWED FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace page 3 of Renewed Facility Operating License No. DPR-68 with the attached page 3. The revised page is identified by amendment number and contains a marginal line indicating the areas of change.

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

Remove	Insert
2.0-1	2.0-1
3.3-8	3.3-8
5.0-24	5.0-24
5.0-24a	5.0-24a
5.0-24b	5.0-24b
5.0-24c	5.0-24c

(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) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 3952 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 308, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

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 ≤ 23% RTP.

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

MCPR shall be ≥ 1.05.

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.

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	≥ 18%	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	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, he 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 documenta ion 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.

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 MINIMUM CRITICAL POWER RATIO (MCPR) and $MCPR_{99.9\%}$ for Specification 3.2.2;

(4) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased Simulated Thermal Power-High Scram setpoints used in the Automated BSP Scram Region, and the BSP Boundary 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)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (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/MICROBURN-B2, 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. (Deleted).

10. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.

11. (Deleted).

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

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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 evaluations dated February 15, 2013, and July 31, 2014.

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. (Deleted).

19 BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

20. ANP-10298P-A Revision 1, ACE/ATRIUM 10XM Critical Power Correlation, AREVA Inc., March 2014.

21. (Deleted).

22. NEDC-33075P-A, GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, Revision 8, November 2013.

23. BAW-10247P-A Supplement 2P-A Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods, Framatome Inc., August 2018.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

24. ANP-10300P-A Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome Inc., January, 2018.

25. ANP-10332P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios, Framatome Inc., March 2019.

26. ANP-10333P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Framatome Inc., March 2018 (as supplemented by Section 6.4 of ANP-3908P Revision 4, Applicability of Framatome BWR Methods to Browns Ferry with ATRIUM 11 Fuel, Framatome Inc., June 2022).

27. ANP-10335P-A Revision 0, ACE/ATRIUM 11 Critical Power Correlation, Framatome Inc., May 2018.

28. ANP-10340P-A Revision 0, Incorporation of Chromia Doped Fuel Properties in AREVA Approved Methods, Framatome Inc., May 2018.

29. ANP-3907P Revision 0, Application of BEO-III Methodology with the Confirmation Density Algorithm at Browns Ferry, Framatome Inc., April 2021.

(continued)

ENCLOSURE 5

NON-PROPRIETARY SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 325, 348, AND 308

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-33, DPR-52, AND DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

Table of Contents

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 325, 348, AND 308

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-33, DPR-52, AND DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter dated July 23, 2021 (Reference 1), as supplemented by letters dated April 8, (Reference 2), May 27 (Reference 3), July 18 (Reference 4), July 28 (Reference 5), September 13 (Reference 6), September 29 (Reference 7), and December 9, 2022 (Reference 8), the Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC, the Commission) for changes to the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Browns Ferry), Technical Specifications (TSs). The requested changes would revise Browns Ferry TS 5.6.5.b, “Core Operating Limits Report,” to allow application of Advanced Framatome Methodologies for determining core operating limits in support of loading Framatome fuel type ATRIUM 11™ under the currently licensed Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain. As a result, methodologies that no longer apply would be removed from TS 5.6.5, and new methodologies would be added, along with other conforming changes to the methodologies. The requested changes would also delete Note (f) from TS Table 3.3.1.1-1, “Reactor Protection System Instrumentation,” which would no longer apply due to the transition from the GE Hitachi TRACG04-based Detect and Suppress Solution – Confirmation Density (DSS-CD) stability methodology to the Framatome RAMONA5-FA-based Best Estimate Enhanced Option-III stability methodology that is coupled with the Confirmation Density Algorithm (BEO-III w/CDA). The requested changes would also delete a plant-specific report required at the time ATRIUM 10XM fuel was approved for use. Additionally, the requested changes would revise the TS safety limit (SL) for the minimum critical power ratio (MCPR) based on Technical Specification Task Force (TSTF) Traveler TSTF-564-A, Revision 2, “Safety Limit MCPR” (SLMCPR) (Reference 9). Lastly, the requested changes would revise TS 5.6.5 to require the SLMCPR value be included in the core operating limits report.

By letter dated August 6, 2021 (Reference 10), the licensee provided certain information that had been requested of prior licensees so that the NRC staff could perform confirmatory calculations of the anticipated transient without scram with instability (ATWS-I) transient using the TRACE computer code. Subsequent to the submittal of this information, the NRC staff determined that it was unnecessary to perform any confirmatory calculations for this LAR. Therefore, the NRC staff did not evaluate this information as part of its review.

In section 3.2 and Attachment 4 to the LAR, the licensee committed to provide reports that summarize the results of analyses performed to demonstrate Browns Ferry's compliance with the cycle specific criteria provided by Framatome to TVA as part of the normal reload licensing document package. The licensee stated this information would not be available until later in the reload licensing process. The licensee provided these reports (References 3, 5, 7, and 8) for information only to satisfy certain limitations and conditions of TRs cited in the LAR. The NRC staff confirmed the receipt of the reports but did not evaluate these "for information only" documents as part of its review.

The NRC staff approved Traveler TSTF-564, Revision 2, by letter dated November 16, 2018 (References 11 and 12). Traveler TSTF-564, Revision 2, revises the basis, calculational method, and the value of the TS 2.1.1.2 SL, which protects against boiling transition on the fuel rods in the core, in the current standard technical specifications (STS) for boiling-water reactor (BWR) designs (Reference 13). The STS provide guidance on the format and content of TSs for each of the light-water reactor (LWR) nuclear steam supply systems.

The supplements dated April 8, May 27, July 18, July 28, September 13, September 29, and December 9, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 8, 2022 (87 FR 13014).

2.0 REGULATORY EVALUATION

2.1 PROPOSED CHANGES

As described in Section 2.4 and Attachment 1 of the LAR, the licensee proposed the following changes to the TSs.

a. Revise Browns Ferry, Units 1, 2, and 3, TS 2.1.1.2 and TS 5.6.5.a as follows:

- In TS 2.1.1.2, replace the SLMCPR from ≥ 1.06 to ≥ 1.05 and delete "for two recirculation loop operation of ≥1.08 for single loop operation."
- Relocate the current SLMCPR value (referred to as $MCPR_{99.9\%}$) in TS 2.1.1.2 to the Core Operating Limits Report (COLR).
- Modify TS 5.6.5.a.(3) to require that the $MCPR_{99.9\%}$ value be included in the COLR.

b. Delete the following methodologies from TS 5.6.5.b and replace with "(Deleted).":

Browns Ferry, Unit 1

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

Browns Ferry, Units 1, 2, and 3

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

- 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.

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

- ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.

c. Delete the following duplicate reference from Browns Ferry, Unit 3, TS 5.6.5.b on TS page 5.0-24c:

    21. ANP-3140(P) Revision 0, Browns Ferry Units, 1, 2, and 3 Improved K factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.

d. Replace Reference 21 (Unit 1) and Reference 20 (Units 2 and 3) in TS 5.6.5.b as follows:

    ANP-10298PA Revision 0, "ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010."

with

    ANP-10298P-A Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation, AREVA Inc., March 2014."

e. Add the following methodologies to the Browns Ferry, Units 1 (numbered as References 24 through 30), 2, and 3 (numbered as References 23 through 29), TS 5.6.5.b:

    24. BAW-10247P-A, Supplement 2P-A, Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods, Framatome Inc., August 2018

    25. ANP-10300P-A Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome Inc., January 2018

    26. ANP-10332P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios, Framatome Inc., March 2019

27. ANP-10333P-A Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Framatome Inc., March 2018 (as supplemented by Section 6.4 of ANP-3908P Revision 4, Applicability of Framatome BWR Methods to Browns Ferry with ATRIUM 11 Fuel, Framatome Inc., June 2022)

28. ANP-10335P-A Revision 0, ACE/ATRIUM 11 Critical Power Correlation, Framatome Inc., May 2018

29. ANP-10340P-A Revision 0, Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods, Framatome Inc., May 2018

30. ANP-3907P Revision 0, Application of BEO-III Methodology with the Confirmation Density Algorithm at Browns Ferry, Framatome Inc., April 2021

f. Delete Note (f) (reproduced below) from TS Table 3.3.1.1-1 and Function 2.f in the table:

> Following Detect and Suppress Solution – Confirmation Density (DSS-CD) implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered OPERABLE and shall be maintained OPERABLE and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

g. Revise Browns Ferry, TS 5.6.5 to make the following editorial changes:

- Add the heading “5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)” to the top of TS pages 5.0-24a and 5.0-24b (Unit 2) and TS pages 5.0-24a, 5.0-24b, and 5.0-24c (Unit 3)
- Add a hyphen to Reference 8 (Unit 1) and Reference 7 (Units 2 and 3) between “MICROBURN” and “B2”

2.2 APPLICABLE REGULATIONS AND GUIDANCE

The NRC staff reviewed the licensee’s application to determine whether (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 the activities proposed will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or the health and safety of the public. The NRC staff considered the following regulatory requirements, guidance, and licensing and design-basis information during its review of the proposed changes.

Section 50.36, “Technical specifications,” of Title 10 of the *Code of Federal Regulations* (10 CFR) requires, in part, that TSs for operating reactors include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCO); (3) surveillance requirements (SR); (4) design features; and (5) administrative controls.

Section 50.36(c)(5) of 10 CFR states that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review, and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in [10 CFR] 50.4."

Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," of 10 CFR is not directly applicable to the ATWS-I event because it is intended to address postulated loss-of-coolant accidents (LOCAs) rather than ATWS events. However, this regulation does present a set of acceptance criteria for ensuring adequate cooling of fuel such that significant fuel failures do not occur.

Section 50.62, "Requirements for reduction of risk from anticipated transient without scram (ATWS) events for light-water-cooled nuclear power plants," of 10 CFR requires that the licensee provide an acceptable reduction of risk from ATWS events by inclusion of prescribed design features and demonstrating their adequacy in mitigation of the consequences of an ATWS event.

Section 50.67, "Accident source term," of 10 CFR provides requirements for licensees who seek to revise the accident source term used in the design basis radiological analysis.

Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 establishes the requirements for acceptable ECCS evaluation models. It also specifies the documentation requirements.

The Browns Ferry plants were designed and constructed based on the proposed General Design Criteria (GDC), published in the *Federal Register* (32 FR 10213) by the Atomic Energy Commission (AEC) on July 11, 1967 (hereafter "draft GDC"). The AEC stated these GDC would not add any new requirements but were intended to describe more clearly the Commission requirements at that time to assist applicants in preparing applications. The AEC published the final rule that added Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, as corrected, 36 FR 12733; July 7, 1971) (hereafter "adopted GDC" or "GDC").

As discussed in the NRC Staff Requirements Memorandum (SRM) for SECY-92-223, dated September 18, 1992 (Reference 14), the Commission decided not to apply the adopted GDC to plants with construction permits issued prior to May 21, 1971. The Commission stated in the SRM that the plants licensed before the GDC were formally adopted were evaluated on a plant-specific basis, determined to be safe, and that current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDC.

As stated in the LAR and discussed in Appendix A of the Browns Ferry Nuclear Plant Updated Final Safety Analysis Report (UFSAR) (Reference 15), the licensee has made changes to the facility over the life of the plant that may have invoked the adopted GDC. The extent to which the adopted GDC have been invoked can be found in specific sections of the UFSAR and in other design and licensing basis documentation. The NRC staff determined that the following GDC are applicable to this review:

- GDC 4, *Environmental and dynamic effects design bases*, which requires, in part, that structures, systems, and components (SSCs) important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer.

- GDC 10, *Reactor design*, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

- GDC 12, *Suppression of reactor power oscillations*, which requires unstable oscillations with the potential of exceeding specified acceptable fuel design limits (SAFDLs) either be impossible or reliably and readily detected and suppressed.

- GDC 13, *Instrumentation and control*, which requires, in part, that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges.

- GDC 20, *Protection system functions,* which requires, in part, that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions.

- GDC 25, *Protection system requirements for reactivity control malfunctions*, which requires the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

- GDC 26, *Reactivity control system redundancy and capability*, which requires, in part, two independent reactivity control systems of different designs be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes, including AOOs, so that SAFDLs are not exceeded.

- GDC 28, *Reactivity Limits*, which requires, in part, that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor sufficiently disturb the core, its support structures, or other reactor vessel internals to significantly impair the capability to cool the core.

- GDC 29, *Protection against anticipated operational occurrences*, which requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

- GDC 33, *Reactor coolant makeup*, which requires a system to supply reactor coolant makeup for protection against small breaks in the RCPB be provided. The system safety function must assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components that are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

- GDC 35, *Emergency core cooling*, which requires, in part, a system to provide abundant emergency core cooling to transfer heat from the reactor core following any LOCA.

Regulatory Guide (RG) 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” describes methods and procedures that the NRC staff considers acceptable when analyzing a postulated control rod drop accident (Reference 16). It is the approved version of draft guide (DG)-1327 of the same name (Reference 17). Regulatory Guide 1.236 also references RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” (Reference 18) and RG 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” (Reference 19) for evaluation of accident dose radiological consequence criteria.

The NRC staff’s guidance for the review of applications is generally contained in NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (Reference 20). Guidance for the review of fuel system design is contained in section 4.2 (Reference 21); guidance for the review of the design of the fuel assemblies and control systems is contained in section 4.3 (Reference 22); and guidance for the review of the thermal and hydraulic design of the core is contained in section 4.4 (Reference 23). Guidance for the review of TSs is contained in chapter 16.0, “Technical Specifications,” Revision 3, dated March 2010 (Reference 24). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared STS for each of the LWR nuclear designs. Accordingly, for General Electric boiling-water reactor (BWR) designs, the NRC staff’s review included consideration of whether the proposed changes are consistent with NUREG-1433, “Standard Technical Specifications, General Electric BWR/4 Plants,” Volumes 1 and 2, (Reference 13), as modified by NRC staff-approved TSTF travelers.

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the LAR, as supplemented, to determine if the proposed changes are consistent with the regulations, guidance, and plant-specific design and licensing basis information discussed in section 2.2 of this safety evaluation (SE).

The NRC staff also reviewed the LAR, as supplemented, to evaluate the applicability of the Advanced Framatome Methodologies to Browns Ferry to confirm that the use of the methodologies is within the NRC staff-approved ranges necessary to support the licensee’s planned transition to ATRIUM 11 fuel and to verify that the results of the analyses and methodologies meet the GDC and are in compliance with the regulatory requirements and guidance listed in section 2.2 of this SE. Each subsection of this SE includes a regulatory evaluation section specific to that portion of the staff’s review.

3.1 EVALUATION OF TECHNICAL SPECIFICATION CHANGES

The NRC staff evaluated the proposed TS changes described in section 2.1 of this SE to determine if the proposed changes meet the criteria set forth in 10 CFR 50.36(c).

The current SL on MCPR (i.e., $MCPR_{99.9\%}$) in TS 2.1.1.2, also referred to as SLMCPR, is affected by the cycle-specific design, such as core power distribution, fuel type, and operating power-flow domain. These factors generally vary from cycle-to-cycle that change the SLMCPR values. The licensee calculated the proposed SLMCPR (i.e., $MCPR_{95/95}$) as 1.05 for both the ATRIUM 10XM and ATRIUM 11 fuel types using the statistical analysis described in ANP-3857, Revision 2 (Attachment 5a to Reference 1). The NRC staff reviewed the licensee's calculation in ANP-3857 and finds the proposed change of SLMCPR to 1.05 in TS 2.1.1.2 acceptable for the reasons discussed below. The proposed SLMCPR (or $MCPR_{95/95}$) of 1.05 means that with the value of SLMCPR ≥ 1.05, there is a 95-percent probability at 95-percent confidence level that the hot rod in the core does not experience a departure from nucleate boiling (DNB) or boiling transition condition during normal operation or AOOs. The NRC staff also finds the licensee's proposed change to add the $MCPR_{99.9\%}$ to TS 5.6.5.a and include it in the COLR acceptable, as justified in section 3.13 of this SE.

The NRC staff finds the methodologies proposed to be deleted acceptable because they are no longer applicable and are being replaced with the Advanced Framatome Methodologies described in section 2.1 above.

The NRC staff finds the replacement of ANP-10298, Revision 0 with Revision 1 acceptable because Revision 0 was for ATRIUM 10XM ACE CPR correlation, supplemented by a plant-specific report ANP-3140P, which was required at the time of the NRC review of the Browns Ferry ATRIUM 10XM fuel transition LAR (approved in XN-NF-80-19(P)(A) Volume 3, Revision 2 [Reference 25]) due to an error in the K factor method described in XN-NF-80-19(P)(A), Volume 4, Revision 1 (Reference 26). The NRC subsequently approved ANP-10298, Revision 1, which includes the corrected K factor method. Therefore, the NRC staff also finds it acceptable to delete the reference to the plant-specific supplemental report, ANP-3140P, because it is no longer required.

All TRs proposed to be added in TS 5.6.5.b are NRC staff approved with the following exceptions:

- The NRC staff-approved control rod drop accident (CRDA) methodology described in ANP-10333P-A, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," (Reference 27) requires the addition of two plant-specific methodology elements, which are mentioned in section 6.4 of ANP-3908, Revision 4, (Attachment 12a to Reference 6). Review of these plant-specific elements is discussed in section 3.11 of this SE.
- ANP-3907 (Attachment 13a to Reference 1) references the following two NRC staff-approved reports: (a) ANP-10344P-A, Revision 0, "Framatome Best-Estimate Enhanced Option III Methodology," (Reference 28), and (b) NEDC-33075P-A, Revision 8, "GE Hitachi Boiling Water Reactor Detect and Suppress Solution Confirmation Density," (Reference 29). Therefore, the NRC staff finds the addition of plant-specific report ANP-3907 in TS 5.6.5.b acceptable, for the reasons discussed in section 3.9 of this SE.

The NRC staff reviewed Note (f) of TS Table 3.3.1.1-1 and finds that this note would be unnecessary with the implementation of the plant-specific BEO-III w/CDA of DSS-CD approach. Therefore, the NRC staff finds the deletion of this note acceptable.

Lastly, the NRC staff reviewed the proposed editorial changes to Browns Ferry, TS 5.6.5 and finds them acceptable because they correct the format of the TS and a title of one of the TRs and do not alter the technical content.

3.2 CRITICAL POWER CORRELATION (ANP-3857)

NUREG-0800, section 4.4, provides the specific criteria necessary for the evaluation of fuel design limit to meet the requirements of GDC 10 and GDC 12. It states that there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs. The NRC SE for Traveler TSTF-564, Revision 2 revises the basis, calculational method, and the value of TS SL 2.1.1.2, which protects against boiling transition on the fuel rods in the core. Prior to the adoption of Traveler TSTF-564, the TS basis ensured that 99.9 percent of the fuel rods in the core are not susceptible to boiling transition. The revised basis would ensure that there is a 95-percent probability at a 95-percent confidence level that no fuel rods will be susceptible to boiling transition using an SL based on CPR data statistics. The CPR is the ratio of the power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition to the actual assembly operating power.

ATRIUM 10XM and ATRIUM 11 fuel types are not identified in Traveler TSTF-564, Table 1. However, Traveler TSTF-564 allows fuel vendors for other fuels to determine the MCPR with a 95 percent probability at a 95 percent confidence level ($MCPR_{95/95}$) so that no fuel rods will be susceptible to boiling transition using the methodology described in Traveler TSTF-564. The licensee provided the required description of the derivation of the $MCPR_{95/95}$ for ATRIUM 10XM and ATRIUM 11 fuel types in ANP-3857 (Attachment 5a to the LAR) based on the information contained in NRC staff-approved TRs ANP-10298P-A (Reference 30) and ANP-10335P-A (Reference 31) respectively listed in Browns Ferry TS 5.6.5.b.

The NRC staff reviewed the licensee's calculation in ANP-3857, sections 3.0 and 4.0, and finds it is in accordance with the NRC staff-approved TRs ANP-10298P-A and ANP-10335P-A and, therefore, acceptable.

3.2.1 Compliance with NRC-Imposed Limitations and Conditions in Safety Evaluation for ANP-10335P-A, Revision 0

The ACE/ATRIUM 11 critical power correlation methodology TR has three limitations and conditions listed in section 4.0 of the NRC staff's SE for ANP-10335P-A. In a letter dated July 18, 2022 (Reference 3), the licensee provided compliance statements for these limitations and conditions. The NRC staff's evaluation of the compliance with the limitations and conditions is given below.

Limitation and Condition 1

> The ACE/ATRIUM 11 correlation shall not be applied outside of the parameter ranges presented in Table 2.1 of ANP-10335P.

> Because the testing did not include flow in the internal water canister, the limits on mass flow rate are imposed on the mass flow rate in the heated section of the bundle (i.e., they do not include bypass flow that would be included if the bundle inlet mass flow rate were to be used). Also note that while Framatome did not specify **[[**
>
> **]]**
>
> Additionally, the LPF (local peaking factor) limit of **[[ ]]** can be exceeded only for perturbed conditions in MCPR safety limit Monte Carlo calculations and for bundles that can be shown to be non-limiting (e.g., high burnup or controlled bundles).

<u>Evaluation</u>

In a letter dated July 18, 2022, the licensee provided the following evaluation to this limitation and condition:

> The ACE correlation is implemented in the code library ACELIB. The ACELIB library is used by all codes that require determination of critical power for ATRIUM 11. Limitations on parameter ranges of mass flow rate, inlet enthalpy, and pressure are enforced within ACELIB. **[[**
>
> **]]**

Based on the above explanation by the licensee, the NRC staff finds this limitation and condition is satisfied.

<u>Limitation and Condition 2</u>

> For bundles with LPFs greater than **[[**
>
> **]]**

> The following increased [[ ]] uncertainties shall be applied to the following listed rod positions [[ ]]:
>
> [[
>
>]]

Evaluation

In the letter dated July 18, 2022, the licensee stated that the higher uncertainty associated with peaking factors greater than [[ ]] is applied within the three dimensional safety limit (SAFLIM3D) computer program documented in ANP-10307PA (Reference 32) which is used to determine the SLMCPR also referred to as $MCPR_{99.9\%}$. The rod position dependent [[ ]] shown above are also applied in the SAFLIM3D computer program.

Based on the licensee's statement, and staff review of ANP-10307PA, the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 3

> Application of the ACE/ATRIUM 11 correlation in a transient analysis methodology requires verification that the correlation conservatively predicts CP [critical power] compared to test data and demonstrates similar behavior compared to other implementations of the correlation.
>
> Framatome shall not apply the ACE/ATRIUM 11 correlation in transient analysis methodologies other than XCOBRA-T and AURORA-B without first verifying the appropriate correlation behavior and conservatism.

Evaluation

The licensee used the ANP-10300P-A (Reference 33) methodology for transient analysis. In the letter dated July 18, 2022, the licensee stated that the application of the ACE/ATRIUM 11 correlation using this methodology is documented in ANP-10335P-A, Enclosure 1, "Proprietary copy of ANP-10335Q2P, Revision 1, 'ACE/ATRIUM 11 Critical Power Correlation – RAIs,'" Section 2.0, and Enclosure 2, "Non-Proprietary copy of ANP-10335Q2P, Revision 1, 'ACE/ATRIUM 11 Critical Power Correlation – RAIs,'" Section 2.0 (see Reference 31). The NRC staff reviewed the licensee's demonstration of the ACE/ATRIUM 11 correlation application in ANP-10335P-A, Enclosure 1, and finds this limitation and condition is satisfied.

The NRC staff finds the licensee has satisfied the limitations and conditions listed in section 4.0 of the NRC staff's SE for ANP-10335P-A.

3.3 EQUILIBRIUM CYCLE DESIGN (ANP-3854)

ANP-3854P, Revision 2 (Attachment 6a to Reference 6) summarizes the equilibrium core design and fuel management calculations for a representative full core of ATRIUM 11 fuel loaded at Browns Ferry. These analyses were performed using the Framatome neutronic methodology, which uses the CASMO-4 lattice depletion code for generation of nuclear cross-section data and the MICROBURN-B2 3-dimensional (3D) core simulator code for depletion, core physics calculations, and pin power reconstruction for thermal margin analysis (Reference 34).

The equilibrium core design is not intended to reflect the actual nuclear design of fuel assemblies or a loading pattern for use at Browns Ferry. Rather, it is a core design that is developed using the assumption that every cycle is operated identically, and the fresh fuel batches for every cycle consist of the same number of ATRIUM 11 fuel assemblies with the same enrichment and gadolinia distributions. As such, this core design does not directly support a demonstration that a full-core loading of ATRIUM 11 fuel can safely be operated at Browns Ferry. However, the equilibrium core design serves as a reference core design that is used in other analyses to either: (1) demonstrate how the licensee will perform cycle-specific safety analyses, or (2) perform a cycle-independent analysis intended to become a licensing analysis of record for future cycles.

As such, the NRC staff's review focused on the reasonableness of this equilibrium core design as a stand-in for future cycles. The primary design criteria include operating cycle length, coastdown assumptions, control rod operating strategy, thermal limit margins, and shutdown margin.

The operating cycle length, coastdown assumptions, and control rod operating strategy are consistent with current plant operations. The licensee would evaluate any change to the operating cycle length, coastdown assumptions, or control rod operating strategy under 10 CFR 50.59, "Changes, tests and experiments," or other change processes, as necessary, to ensure that any impact on the licensing basis analysis will be evaluated. The thermal limits are based on other analyses such as maximum linear heat generation rate (LHGR) values assumed in the LOCA and ATWS-I analyses. Finally, the shutdown margin and depletion target eigenvalues are developed based on historical data for Browns Ferry, which is consistent with standard industry practice. Based on these constraints, the fuel assembly batch sizes and nuclear compositions (including U-235 and gadolinia enrichments) are specified to ensure that the equilibrium fuel cycle design will meet all applicable design constraints. As such, this core design may be considered a representative core design for the purpose of the safety analysis demonstrations. When used directly in the licensing analyses such as ATWS-I and LOCA analyses, the NRC staff confirmed the applicability of use of this core design as reasonably representative or bounding of future cycles, as discussed later in this SE.

Based on the above, the NRC staff finds that the cycle design calculations and projected control rod patterns for the equilibrium core design are consistent with their intended uses.

3.4 ATRIUM 11 MECHANICAL DESIGN (ANP-3860)

ANP-3860P (Attachment 8a to Reference 6) provides the mechanical design details, fuel structural analysis results of the ATRIUM 11 fuel assemblies, and fuel channel designs, while ANP-3866P (Attachment 9a to Reference 6) provides the design parameters and design evaluation results of the ATRIUM 11 fuel rods to be used at Browns Ferry.

3.4.1 Summary of Mechanical Design of ATRIUM 11 Fuel Assemblies for Browns Ferry

ANP-3860P provides key fuel assembly design details for the Framatome ATRIUM 11 fuel assembly design planned for use at Browns Ferry. The ATRIUM 11 fuel assembly consists of a lower tie plate (LTP), an upper tie plate (UTP), 112 fuel rods, nine spacer grids, a central water channel, and miscellaneous assembly hardware. Of the 112 fuel rods, 20 are part-length fuel rods (PLFRs) of which 12 are short PLFRs and 8 are long PLFRs. The structural members of the fuel assembly include the tie plates, spacer grids, water channel, and connecting hardware. The structural connection between the LTP and the UTP is provided by the central water channel. The lowest of the nine spacer grids is located just above the LTP to laterally restrain the lower ends of the fuel rods. Table 2-1 of ANP-3860P lists the fuel assembly and component description of the ATRIUM 11 fuel assembly design. Further descriptions of the fuel assembly components are provided in ANP-3860P.

3.4.2 Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Assembly Mechanical Design

To perform specific evaluations for the ATRIUM 11 fuel assembly mechanical design, the licensee utilized specific NRC staff-approved methodologies. NRC approval of these methodologies is conditional on adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the TRs. A discussion of how these limitations and conditions are met for Browns Ferry is provided below for each of the TRs directly supporting the ATRIUM 11 fuel assembly mechanical design evaluations, as well as a discussion of the applicability of TRs already in use at Browns Ferry for analysis of the ATRIUM 10 assembly design that may not automatically apply to the ATRIUM 11 fuel assembly design.

ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, and Supplement 1, dated May 1995 (Reference 35).

ANF-89-98(P)(A) provides some generic mechanical design criteria that were approved by the NRC staff for use with evaluation of Framatome fuel designs. As discussed in section 3.4 of this SE, ANP-3860P describes how the design criteria presented in ANF-89-98(P)(A) apply to the ATRIUM 11 fuel assembly mechanical design.

EMF-93-177P-A, "Mechanical Design for BWR Fuel Channels," Revision 1, August 2005 (Reference 36), and Supplement 1P-A, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," Revision 0, September 2013 (Reference 37)

The NRC staff's SE for EMF-93-177P-A specified several limitations and conditions that have already been shown to be met at Browns Ferry for the channels associated with the ATRIUM 10 fuel. Since the ATRIUM 11 channels are very similar, the staff finds that the disposition of the limitations and conditions remain applicable. The two exceptions are the use of Z4B (i.e., zircaloy BWR material similar to Zircaloy-4 (Zry-4)) channels, as approved in EMF-93-177, Supplement 2P-A, Revision 1, and interior milling, which is addressed through use of the EMF-93-177, Supplement 1P-A methodology. The Supplement 1P-A methodology was approved with no limitations or conditions.

BAW-10247P-A, Supplement 2P-A, "Realistic Thermal Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," Revision 0, August 2018 (Reference 38)

The ATRIUM 11 fuel mechanical design evaluation, as discussed in section 3.4.3 of this SE below, confirms that the **[[** **]]** and that **[[** **]]** The remaining limitations and conditions are met for the ATRIUM 11 fuel assembly design, since the channels are constructed of either Zry-4 or Z4B and the fuel rod materials fall within the range of applicability for the database used to support the fuel rod growth correlations.

3.4.3 Fuel Assembly Mechanical Design Evaluation

The objectives of the fuel design are that (i) the fuel assembly (system) does not fail as a result of normal operation and AOOs, (ii) fuel system damage is never so severe as to prevent control rod insertion when it is required, (iii) the number of fuel rod failures is not underestimated for postulated accidents, (iv) fuel coolability is always maintained (Reference 36), (v) the mechanical design of the fuel assemblies shall be compatible with co-resident fuel and the reactor core internals, and (vi) fuel assemblies shall be designed to withstand the loads from handling and shipping. The first four objectives are from SRP Section 4.2 and the latter two are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel (co-resident fuel). This fuel assembly mechanical design evaluation contains only fuel assembly structural analyses.

3.4.4 Stress, Strain, or Loading Limits on Assembly Components

The licensee used the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Reference 39) as a guide to establish the acceptable stress, deformation, and load limits for standard assembly components. These limits are applied to the design and evaluation of the UTP, LTP, spacer grids, springs, and load chain components, as necessary and applicable. The fuel assembly structural component criteria under faulted conditions are based on Appendix F of the ASME B&PV Code Section III with some criteria derived from component tests. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. Analyses were performed to determine the mechanical performance of assembly components during accidents (e.g., seismic events or LOCA events), fuel handling events, or during normal and AOO conditions.

The licensee stated that all significant loads experienced during normal operation, AOOs, and under faulted conditions are evaluated to confirm the structural integrity of the fuel assembly components. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. Stress calculations use conventional, open-literature equations. A general-purpose, finite element stress analysis code, such as ANSYS, may be used to calculate component stresses. The NRC staff finds this acceptable because the structural design of the assembly and fuel channel were evaluated using the NRC staff-approved methodologies, specifically, ANF-89-98(P)(A) and BAW-10247P-A, Supplement 2P-A, described in section 3.4.2 of this SE.

3.4.5 Fretting Wear

The general acceptance criterion is that fuel rod failures due to grid to rod fretting shall not occur. Fretting wear is evaluated by tests using a full-size ATRIUM 11 test assembly. **[[**

]] Post-test inspections of the fuel assembly showed no significant wear on fuel rods. Although the testing period is short relative to the time that a fuel assembly will typically spend in the reactor core, this result is sufficient to provide reasonable assurance that structural flaws in the fuel rod cladding would not be expected to lead to widespread fuel rod failures.

The NRC staff finds that based on the results of the fretting wear testing, widespread rod failures would not be expected because of fretting effects. The staff notes that isolated rod failures due to localized mechanisms leading to excessive fretting are not explicitly required by regulatory acceptance criteria to be addressed; therefore, the staff finds the generic testing performed in support of this conclusion sufficient.

3.4.6 Rod Bow

A combination of differential expansion between the fuel rods and cage structure, thermal gradients, and flux gradients can result in lateral loads applied to the fuel rods. This load may result in rod bowing in the spans between spacer grids due to creep. Since a reduction in rod pitch may have a detrimental impact on power peaking and local heat transfer, the licensee must address the potential impact on thermal margins. The Framatome design criterion for fuel rod bowing is **[[**

]]

The licensee completed post-irradiation exams for fuel rod bow behavior for exposures up to **[[** **]]** of the lead test assembly for ATRIUM 11. This exposure is beyond the threshold where increasing rod bow had been observed on other designs. Therefore, the NRC staff finds that the ATRIUM 11 fuel design has been shown to have minimal rod bow.

3.4.7 Axial Irradiation Growth

Rod growth, assembly growth, and fuel channel growth are calculated using correlations that were reviewed and approved by the NRC staff in BAW 10247P-A Supplement 2P-A (Reference 38). In accordance with BAW 10247P-A Supplement 2P-A, **[[**

]]

The NRC staff finds the approach used to address axial irradiation growth to be acceptable based on the use of an NRC staff-approved methodology within the bounds of applicability of the approval and consistent with the limitations and conditions.

3.4.8 Assembly Lift-Off

The design criteria for assembly lift-off are no lift-off from fuel support during normal operations (including AOOs) and no disengagement from fuel support during postulated accidents. These criteria assure control blade insertion is not impaired. For normal operating conditions, the calculated net axial force acting on the assembly due to the addition of the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy will be in the downward direction, indicating no assembly lift-off. The licensee confirmed that the calculated net force will be in the downward direction, indicating no assembly lift-off. **[[**

]]

Mixed core conditions for assembly lift-off are considered on a cycle-specific basis, as determined by the plant operating conditions and other fuel types. Analyses to date indicate a large margin to assembly lift-off under normal operating conditions.

For faulted (postulated accident) conditions, **[[**

]] The fuel will not lift under normal or AOO conditions. It will not become disengaged from the fuel support under faulted conditions, nor block insertion of the control blade in all operating conditions.

The NRC staff finds the lift-off evaluation acceptable because the evaluation is complete and adequate to meet the required design criteria and satisfy the SRP objectives.

3.4.9 Fuel Assembly Handling and Fuel Assembly Axial Load Tests

The fuel assembly is required to withstand, without permanent deformation, all normal axial loads from shipping and fuel handling operations. The analyses or testing shall demonstrate that the fuel is capable of **[[**
]]

The licensee completed tests and analyses by loading a test assembly or the individual components of the load chain to an axial tensile force greater than **[[**
]] The results demonstrated no yielding after loading.

The fuel assembly axial load test is used in support of analytical or finite element analysis to demonstrate that no significant permanent deformation occurs for loads of **[[**
]] The test consists of the following:

- **[[**
-
-

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The NRC staff finds the fuel assembly handling and axial load analyses acceptable because the analyses and testing are complete and adequate and demonstrate that the fuel assembly meets the requirements.

3.4.10 Compression Spring Forces

The compression spring force shall support the weight of the UTP and channel throughout the design life of the fuel. The ATRIUM 11 has a single large compression spring mounted on the central water channel. The compression spring serves to provide support for the UTP and fuel channel. The licensee stated the spring force is calculated based on the installed deflection and specified spring force requirements to meet support criteria. Irradiation-induced relaxation is taken into account for EOL conditions. The minimum compression spring force at EOL is greater than the combined weight of the UTP assembly and fuel channel assembly. Since the compression spring design of the ATRIUM family of fuel assemblies load chain designs do not interact with the fuel rods, the staff determined that no consideration is required for fuel rod buckling loads. The NRC staff finds this acceptable because the licensee's analysis methodology and criteria are consistent with the approved topical report ANF-89-98(P)(A), Revision 1.

3.4.11 Lower Tie Plate Seal Spring

The LTP seal spring shall limit the bypass coolant leakage rate between the LTP and fuel channel. The seal spring shall accommodate expected channel deformation while remaining in contact with the fuel channel, shall have adequate corrosion resistance, and be able to withstand the operating stresses without yielding. Flow testing is used to confirm acceptable bypass flow characteristics. Seal spring stresses are analyzed using a finite element method. The NRC staff finds this method of LTP seal spring analysis acceptable because the licensee's analysis methodology and criteria are consistent with the NRC staff-approved TR ANF-89-98(P)(A), Revision 1.

3.4.12 Structural Deformations

Structural deformations or stresses from postulated accidents are limited according to the requirements contained in ASME B&PV Code, Section III, Division 1, Appendix F, and SRP section 4.2, appendix A (Reference 21).

[[

]]

The dynamic properties of the ATRIUM 11 fuel assembly are provided to Browns Ferry by the vendor in support of evaluations assessing the impact of the introduction of the ATRIUM 11 fuel to the reactor pressure vessel, internal reactor components and other applicable evaluations.

The NRC staff finds the structural deformation evaluation acceptable because it was evaluated consistent with NRC staff-approved TR EMF-93-177(P)(A), Revision 1.

3.4.13 Test Verification

In general, testing and analyses have shown the ATRIUM 11 dynamic response is similar to the ATRIUM-10 and ATRIUM 10XM fuel assemblies. Testing is performed to obtain the dynamic characteristics of the fuel assembly and spacer grids.

Fuel Assembly Static Lateral Deflection testing is performed to determine the fuel assembly stiffness, both with and without a fuel channel. The fuel assembly is supported at two ends in a vertical position during the test where a side displacement is applied at the central spacer location and the corresponding force is measured.

The lateral vibration testing consists of both a free vibration test and a forced vibration test **[[** **]]** The test setup for the free vibration test **[[** **]]** The forced vibration test **[[** **]]**

Spacer grid impact strength is determined by a **[[** **]]** The maximum force prior to the onset of buckling was determined from the tests. The results were adjusted to reactor operating temperature conditions to establish an allowable lateral load.

In order to establish an allowable lateral load, tie plate strength tests were completed. The tests were lateral load tests for both the upper and lower tie plates. To determine a limiting lateral load for accident conditions for the 3rd Generation FUELGUARD™ (3GFG) LTP, a lateral load test was conducted by attaching the grid of the tie plate to a rigid vertical plate **[[** **]]**

The NRC staff finds the licensee has acceptably addressed the testing process for the ATRIUM-11 fuel and has shown the fuel will perform as desired during operation.

3.4.14 Fuel Channel and Components

3.4.14.1 Normal Operations

During normal operation, the stress limits are obtained from the ASME B&PV Code, Section III, Division 1, Subsection NG for Service Level A (Reference 39). The calculated stress intensities are due to the differential pressure across the fuel channel wall. The pressure loading includes the normal operating pressure plus the increase during AOO. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation. The licensee obtained the properties from Huang et al., 1996 (Reference 40). As an alternative to the elastic analysis stress intensity limits, a plastic analysis may be performed as permitted by paragraph NB-3228.3 of the ASME B&PV Code (Reference 39).

The licensee stated, that in the case of AOOs, the amount of bulging is limited to that value which will permit control blade movement. During normal operation, any significant permanent deformation due to yielding is precluded by restricting the maximum stresses at the inner and outer faces of the channel to be less than the yield strength.

The licensee used a conservative fatigue life for the fuel channel based on the O'Donnell and Langer curve (Reference 41). A factor of 2 on stress amplitude or a factor of 20 on the number of cycles is added, whichever is more conservative. This method accounts for cyclic changes in power and flow during operation which impose a duty load on the fuel channel.

Fuel channels do not have limits for oxidation and hydriding. The BWR Water Chemistry Guidelines, alloy composition, impurity limits and heat treatments work in concert such that **[[**

]]

Long-term deformation due to creep and exposure are to be protected against in order to maintain normal control blade maneuvers and maintain control blade insertion times within the TS limits. Overall deformation of the fuel channel occurs from a combination of bulging and bowing. Bulging of the side walls occurs because of the differential pressure across the wall. Lateral bowing of the channel is caused primarily from the neutron flux and thermal gradients. The licensee stated that **[[**

]]

3.4.14.2 Accident Conditions

The fuel channel stresses, load limit, and vertical acceleration criteria are based on the ASME B&PV Code, Section III, Appendix F, for faulted conditions (Service Level D) (Reference 39). The unirradiated properties of the fuel channel material are used during

analyses since the yield and ultimate tensile strength increase during irradiation (Reference 40). **[[**
]]
Vertical acceleration produces a membrane stress in the axial direction due to a postulated impact of the channeled fuel assembly impacting the fuel support after liftoff. The amount of bulging remains limited to that value which will permit control blade insertion.

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]] The NRC staff finds this acceptable because it was evaluated consistent with the NRC staff-approved TR EMF-93-177(P)(A), Revision 1.

3.4.15 Technical Conclusion

Tables 3-1 through 3-2 of Attachment 8a to the LAR (Reference 1) provide a disposition of the specific design criteria evaluated for the ATRIUM 11 fuel assembly design, based on the aforementioned tests and analyses. The NRC staff's evaluation of the approach used to perform the dispositions are documented in the above subsections. For the reasons described in the above subsections, the NRC staff finds that these analyses and tests have been performed acceptably to ensure that the mechanical design criteria for the ATRIUM 11 fuel assembly design are met for use in the Browns Ferry reactor cores.

3.5 ATRIUM 11 FUEL ROD THERMAL MECHANICAL (ANP-3866)

ANP-3866P (Attachment 9a to Reference 6) provides key fuel rod design details for the Framatome ATRIUM 11 fuel planned for use at Browns Ferry. The ATRIUM 11 fuel rod is conventional in design configuration and very similar to past designs such as the ATRIUM 10XM and ATRIUM-10 fuel rods. The fuel rods are made with Zircaloy-2 cladding **[[**

]] plenum spring on the upper end of the fuel column assists in maintaining a compact fuel column during shipment and initial reactor operation.

There are two PLFR designs incorporated in the fuel assembly. **[[**

]]

Table 3-1 of ANP-3866P lists the key fuel rod design parameters for the ATRIUM 11 fuel. Further descriptions of the fuel assembly components are provided in ANP-3866P.

3.5.1 Applicability of Methodologies for Analysis of ATRIUM-11 Fuel Rod Design

To perform specific evaluations for the ATRIUM-11 fuel rod design, the licensee utilized specific NRC staff-approved methodologies. NRC approval of these methodologies is conditional on adequately meeting the limitations and conditions listed in the NRC staff's SE for each of these TRs. A discussion of how this requirement is met for Browns Ferry is provided below for each of the TRs directly supporting the ATRIUM 11 fuel rod design evaluations, as well as a discussion of the applicability of TRs that were already in use at Browns Ferry for analysis of the ATRIUM 10 fuel rod design that may not automatically apply to the ATRIUM 11 fuel rod design.

ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Revision 1, and Supplement 1, May 1995 (Reference 35)

ANF-89-98(P)(A) provides some generic fuel rod design criteria that were approved by the NRC staff for use with evaluation of Framatome fuel designs. The ATRIUM 11 fuel rod design describes how the design criteria presented in ANF-89-98(P)(A) apply to the ATRIUM 11 fuel rod design.

BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, February 2008 (Reference 42)

Section 3.5.2 of this SE includes a discussion under the "Oxidation, Hydriding, and Crud Buildup" subsection that describes how the crud effects are addressed. ANP-10340P-A (Reference 43) contains a similar limitation and condition on the **[[ ]]**, which is addressed through an automated software check. The remaining limitations and conditions are addressed by only utilizing the methodology within the bounds defined by the limitations and conditions.

ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA-Approved Methods," Revision 0, May 2018 (Reference 43)

The chromia-doped fuel properties and models described in this TR are directly applicable to the ATRIUM 11 fuel pellets. The limitations and conditions are met through a combination of automated software checks and administrative controls, as described in section 2-18 of the BWR compendium (Reference 44). The automated software checks are managed through the Framatome software quality assurance program, which is subject to normal NRC oversight activities as part of verifying compliance with Appendix B to 10 CFR Part 50. The NRC staff noted that the methodologies that will be used to evaluate the ATRIUM 11 fuel at Browns Ferry are approved for maximum fuel rod burnups of up to 62 GWD/MTU.

Section 3.6.2 of the NRC staff's SE for ANP-10340P-A, Revision 0, considered sample problems that reflect implementation of chromia-doped fuel properties into AURORA-B and impacts on thermal-hydraulic performance (see Reference 43). Calculated figures of merit (FoMs) for selected anticipated operating conditions were compared. In that SE, the NRC staff discussed the competing effects on energy stored in the fuel induced by chromia dopant and noted that impacts on FoMs were small. Additionally, the NRC staff noted in section 5.0 of that SE that the impact of the chromia dopant on in-reactor performance has been adequately analyzed. Based on these observations, and because trends in calculation results are consistent with expectations, the NRC staff finds this acceptable.

3.5.2 ATRIUM 11 Fuel Rod Design Evaluation

This section of the SE presents the results of the NRC staff's review of fuel rod thermal-mechanical analyses for the ATRIUM 11 fuel. The analyses were performed using acceptance criteria from ANP-89-98(P)(A), Revision 1, and Supplement 1 (Reference 35), and the RODEX4 analysis methodology described in BAW-10247PA (References 42 and 38). In addition, the methodology described in ANP-10340P-A (Reference 43) was used to address the impact of the chromia additive in the fuel pellets for ATRIUM 11 fuel assemblies. The RODEX4 fuel rod analysis code and methodology are used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue, and external oxidation.

3.5.2.1 Fuel Rod Design Evaluation

The ATRIUM 11 fuel assembly design contains multiple changes in geometry to accommodate the change from a 10x10 rod array to an 11x11 rod array within the same basic channel dimensions. The part-length rod specifications also differ from the ATRIUM 10 design. The ATRIUM 11 fuel also utilizes two relatively new materials in its overall composition—the chromia additive in the fuel pellets and the Z4B alloy used for some of the structural elements. Additional details regarding the fuel rod design are provided in section 3.1 of ANP-3866P (Attachment 9a to Reference 6). The fuel rod geometry and compositions generally fit within the applicability of the NRC staff-approved RODEX4 thermal-mechanical analysis methodology (References 42 and 38), with the addition of the chromia doped fuel properties and models reviewed and approved by the NRC staff (Reference 43). Therefore, the RODEX4 code was used to evaluate the fuel rod thermal-mechanical performance of the ATRIUM 11 fuel rod, as appropriate.

Table 2-1 of ANP-3866P provides a summary of the findings from the fuel rod design evaluations that demonstrate that the acceptance criteria are met. The key fuel rod design parameters used in the fuel rod design evaluations are provided in Table 3-1. Table 3-2 provides the specific results based on the equilibrium cycle for MELLLA+ conditions. The fuel rod analyses, such as those for fuel centerline temperature and cladding strain, cover normal operating conditions, and AOOs. More detail on the NRC staff's considerations in reviewing each acceptance criteria is provided below.

3.5.2.2 Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. As stated in section 3.3.1 of ANP-3866P, a fabrication limit is imposed **[[ ]]** and enforced via moisture controls. The NRC staff finds this to be an acceptable approach to ensure that the potential sources for hydrogen absorption inside the cladding are minimized, since the fabrication limit is based on NRC staff-approved mechanical design criteria (Reference 35).

3.5.2.3 Cladding Collapse

Fuel pellets undergo a densification process during irradiation, which can result in pellet shrinkage and generate axial gaps along the fuel column. The coolant system pressure causes the cladding to slowly creep inward and close the radial gap between the fuel pellet and the cladding. Since large axial gaps may cause the cladding to collapse into the space between fuel pellets and fail, Framatome imposes an upper limit on the size of the axial gaps. RODEX4 (References 42 and 38) is used to predict the size of the gaps that may form. Since RODEX4 is

a best estimate code, a statistical method is applied to confirm that the maximum size of the axial gaps due to densification is not exceeded for **[[**
]] This approach is consistent with the use of the RODEX4 code and the acceptance criterion in the NRC staff-approved fuel rod evaluation methodology (Reference 35), and therefore, is acceptable.

3.5.2.4 Overheating of Fuel Pellets

One of the limitations on use of the RODEX4 methodology is that it may not be used to model fuel above incipient fuel melting temperatures. In practice, this is avoided by ensuring that the fuel centerline temperatures remain below melting. As necessary, the licensee adjusted the melting point to account for **[[**
]] The RODEX4 methodology (References 42 and 38) is used to determine the fuel centerline temperature for normal operating conditions and AOOs to establish an upper limit on the LHGR that ensures that no centerline melting will occur. This approach is consistent with the use of the RODEX4 methodology, and therefore, is acceptable.

3.5.2.5 Stress and Strain Limits

Under transient conditions, the inner diameter of the cladding may shrink more rapidly than the outer diameter of the fuel pellet due to differences in their rates of change in temperature. If the cladding surface presses on the outside of the fuel pellet, this results in the pellet-clad interaction (PCI) phenomenon. The pressure of the fuel pellet resisting the shrinkage of the cladding can cause local deformation of the cladding or cladding strain. The RODEX4 methodology is used to calculate the predicted cladding strain **[[**
]] to confirm that the strain is no more than one percent. This is consistent with the RODEX4 methodology, and the one percent strain limit is consistent with the NRC staff-approved fuel rod evaluation methodology (Reference 35), and therefore, is acceptable.

Cladding stresses are calculated using solid mechanics elasticity solutions and finite element methods. Stresses are calculated for the primary and secondary loadings. **[[**
]] The results were determined for both beginning of life and end of life conditions to bound the spectrum of possible stresses and then compared against the design limits prescribed by Section III of the ASME B&PV Code (Reference 39). This is consistent with NRC staff-approved mechanical design criteria (Reference 35), and therefore, the NRC staff finds this acceptable.

3.5.2.6 Fuel Densification and Swelling

There are no specific acceptance criteria for fuel densification and swelling; however, these phenomena may affect other acceptance criteria. Consequently, their effects are explicitly included in the RODEX4 methodology (References 42 and 38). The NRC staff has reviewed and approved the models used in RODEX4 to address these phenomena; therefore, the NRC staff finds this is an acceptable disposition.

3.5.2.7 Fatigue

The fuel rod cladding experiences cyclic thermal loads due to power changes during normal operating maneuvers. The thermal cycling translates to cyclic stress, which can lead to fuel rod cladding fatigue. The stresses are calculated using the RODEX4 methodology, and **[[**

]] This information can be used to determine fatigue usage factors for each axial region of the fuel rod, which represents the ratio of the number of accumulated cycles to the maximum allowed number of cycles for a given set of loadings. The cumulative usage factor is determined for each fuel rod by combining the fatigue usage factors. The axial region with the highest cumulative usage factor is used in the subsequent **[[**

]] The results are confirmed to remain below the maximum cumulative usage factor specified as an acceptance criterion.

Since the acceptance criterion is consistent with the NRC staff-approved fuel rod evaluation methodology (References 42 and 38), and the evaluation is performed with a combination of an NRC staff-approved fuel rod analysis methodology with appropriately applicable data, the NRC staff finds this to be acceptable.

3.5.2.8 Oxidation, Hydriding, and Crud Buildup

The RODEX4 code and methodology are used to determine cladding external oxidation and its effect on the heat transfer coefficient from the cladding to the coolant. The acceptance criterion for oxidation is discussed within the NRC staff-approved RODEX4 fuel rod evaluation methodology (References 42 and 38), along with a discussion of how the impact of hydriding and crud buildup are to be addressed. The RODEX4 calculational methodology is calibrated to obtain an appropriate fit to measured oxide thickness data along with relevant uncertainties. The result is used to perform a **[[**

]] A brief discussion of the applicability of hydriding and crud buildup to Browns Ferry is discussed below.

- **[[**

  ]]

- BAW-10247PA (Reference 42) discusses what constitutes “abnormal crud” and how to capture the effect using the crud heat transfer coefficient. Since the corrosion model takes into consideration the effect of the thermal resistance of the crud on the corrosion rate, this is already incorporated into the RODEX4 code. A similar approach would be used to address abnormal corrosion. However, no such observations have been made at Browns Ferry. The cladding properties for the ATRIUM 11 fuel assembly design are not different from the ATRIUM 10 fuel assembly design, so no change is expected as a result of transitioning to ATRIUM 11 fuel.

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The effects of oxidation, crud buildup, and hydriding are addressed through use of the NRC staff-approved RODEX4 fuel rod evaluation methodology and its acceptance criteria, as appropriately applied to Browns Ferry and the ATRIUM 11 fuel assembly design; therefore, the NRC staff finds the disposition as discussed above to be acceptable.

3.5.2.9 Rod Internal Pressure

The fuel rod internal pressure is calculated using the RODEX4 code and methodology (References 42 and 38). The maximum rod pressure is limited to **[[**

]], under both steady-state and transient conditions, consistent with the acceptance criterion defined in ANF-89-98(P)(A) (Reference 35). The NRC staff finds this approach to be acceptable since it is based on a methodology and acceptance criteria that the NRC staff has previously reviewed and approved.

3.5.2.10 Summary of Sections 3.5.1 – 3.5.2

The NRC staff reviewed the licensee's application of the RODEX4 code, analysis methodologies, and acceptance criteria, as approved in ANF-89-98(P)(A) (Reference 35) and BAW-10247PA (References 42 and 38), in the fuel rod thermal-mechanical analyses for the Framatome ATRIUM 11 fuel design that will be loaded and used for operation at Browns Ferry. The staff determined that the fuel design criteria, as supported by the applicable regulations and sections of NUREG-0800, have been satisfied and provide reasonable assurance for safe operation at Browns Ferry.

3.5.3 <u>Conclusion of ATRIUM 11 Fuel Assembly/Rod Design</u>

For the evaluation of ATRIUM 11 Fuel Assembly/Rod Design (section 3.2 of this SE), the NRC staff concludes that the application of ATRIUM 11 fuel (fuel assembly and fuel rod) to Browns Ferry is acceptable by complying with the requirements of GDC 10, 27, and 35. This conclusion is based on the following:

1. The application meets the requirements of GDC 10 with respect to the SAFDLs not being exceeded during any condition of normal operation, including the effects of AOOs by:

    a. Developing and complying with fuel system damage criteria for all known damage mechanisms and operating conditions as evaluated in section 3.5.2.1 of this SE.

    b. Applying NRC staff-approved fuel system design methodologies and adequately meeting the limitations and conditions listed in the NRC staff SE for each of the applied TRs as evaluated in section 3.5.1 of this SE.

2. The application meets the requirements of GDC 27 with respect to the reactivity control system being designed with margin to have capability by assuring the fuel system damage will never be so severe as to prevent control rod insertion when it is required. For example, as evaluated in sections 3.4.13 and 3.4.5 of this SE, respectively, the fatigue and fretting wear of the fuel assembly components were tested to ensure not to interfere with control blade insertion. The fuel will not lift under normal or AOO conditions as analysis demonstrated. It will not become disengaged from the fuel support under faulted conditions, nor block insertion of the control blade in all operating conditions. The

fuel channel was specifically evaluated for changes due to exposure to the reactor environment that may lead to loss of strength or deformation to affect the control rod insertability.

3. The application meets the requirements of GDC 35 with respect to the fuel system being able to transfer heat from the reactor core following any loss of reactor coolant at an acceptable rate by assuring the fuel rod damage will not interfere with effective emergency core cooling, and the cladding temperatures will not reach a temperature high enough to allow a significant metal-water reaction to occur. These assurances are achieved by developing and complying with the fuel coolability related criteria for all severe fuel rod damage mechanisms as addressed in sections 3.5.2.2, 3.5.2.3, 3.5.2.4, 3.5.2.5, 3.5.2.6, and 3.5.2.8, respectively, i.e., internal hydriding, cladding collapse, overheating of fuel pellets, cladding stress and strain limits, fuel densification and swelling, and clad oxidation, hydriding, and crud buildup. The licensee applied NRC staff-approved RODEX4 fuel rod evaluation methodology and adequately met the limitations and conditions listed in the NRC staff's SE for the applied TR.

3.6 THERMAL-HYDRAULIC DESIGN OF ATRIUM 11 FUEL ASSEMBLIES (ANP-3859)

3.6.1 Regulatory Basis

The licensee's thermal-hydraulic analysis for the ATRIUM 11 fuel design is presented in ANP-3859, Revision 1 (Attachment 10a to Reference 6). The purpose of the analysis is to demonstrate that the ATRIUM 11 fuel is hydraulically compatible with the currently loaded ATRIUM 10XM fuel design at the extended power uprate (EPU) under the MELLLA+ conditions.

The applicable 10 CFR Part 50, Appendix A GDC are as follows:

- GDC 10, "Reactor design," requiring that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

- GDC 12, "Suppression of reactor power oscillations," requiring that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

NUREG-0800, SRP section 4.2, "Fuel System Design"; section 4.3, "Nuclear Design"; and section 4.4, "Thermal and Hydraulic Design," provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and the thermal and hydraulic design of the core. In addition, the SRP provides guidance for compliance with the applicable GDC in Appendix A to 10 CFR Part 50.

3.6.2 Technical Evaluation

This section describes the NRC staff's evaluation of the licensee's thermal-hydraulic analyses to demonstrate the following:

- hydraulic compatibility of ATRIUM 11 fuel with co-resident ATRIUM 10XM fuel

- thermal margin performance
- fuel centerline temperature
- rod bow
- bypass flow
- stability
- void fraction
- LOCA analysis
- ASME over-pressurization analysis
- seismic/LOCA liftoff

3.6.2.1 Hydraulic Characterization

The licensee summarized the basic dimensional parameter and component loss coefficients (including resistance of leakage paths) for ATRIUM 11 and the co-resident ATRIUM 10XM fuels in ANP-3859, Table 3.2 and Table 3.3, respectively. The licensee stated that the component loss coefficients for the fuels are based on testing. The primary resistance for the leakage flow through the LTP flow holes is **[[ ]]**

The licensee summarized testing and analysis for determining the hydraulic characteristics in ANP-3908, Revision 4, (Attachment 12a to Reference 6), section 5.3. The wall friction and component loss coefficients were determined based on single-phase testing of a prototypic ATRIUM 11 fuel assembly in the Portable Hydraulic Test Facility (PHTF). Reference 45 describes the PHTF and an overview of the licensee's process for determining the component loss coefficients. ANP-3908, Table 5-3 provides the component loss coefficients of the ATRIUM 11 fuel assembly. Using the tested values of the component loss coefficients, the licensee validated the analytically determined hydraulic characteristics from the MICROBURN-B2 code pressure drop model for the ATRIUM 11 fuel design code described in the NRC staff-approved TR EMF-2158(P)(A) (Reference 34). The graph in ANP-3908, Figure 5-1 shows consistency in the measured versus the MICROBURN-B2 predicted two phase pressure drop for a range of conditions which validates the MICROBURN-B2 models for the pressure drop.

The NRC staff finds the licensee's determination of the ATRIUM 11 fuel hydraulic characteristics acceptable because of the following:

- The component loss coefficients determined by test and analysis are consistent with the results obtained from the MICROBURN-B2 code models confirming the applicability of these models to predict pressure drop for the ATRIUM 11 fuel.

- The testing and analysis approaches used are similar to the approaches that have previously been used to characterize the ATRIUM 10 fuel assembly design which are reviewed by the NRC staff for applicability to other plants.

- ATRIUM 11 fuel assembly does not have any attributes that are expected to require special treatment relative to the ATRIUM 10 fuel assembly design.

- There are no attributes associated with the ATRIUM 11 fuel assembly design that would be expected to require special treatment relative to the ATRIUM 10 fuel assembly design.

3.6.2.2 Hydraulic Compatibility

The licensee analyzed the performance of ATRIUM 11 fuel with its coresident ATRIUM 10XM fuel to evaluate their hydraulic compatibility so that the thermal-hydraulic design criteria are satisfied for the transition cores. The analysis is based on **[[**

]]

Acceptance Criteria

The thermal-hydraulic design criteria are based on the NRC staff-approved TRs ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 35), and XN-NF-80-19(P)(A) Volume 4 Revision 1 (Reference 26).

Methodology

The licensee performed the thermal-hydraulic analysis in accordance with the Framatome methodology for BWRs, XN-NF-80-19(P)(A). The constitutive relationships used for the calculation of pressure drop in the fuel assemblies are calculated using XCOBRA code which is presented in NRC staff-approved XN-NF-79-59(P)(A) (Reference 46). The XCOBRA code predicts steady-state thermal-hydraulic performance of the fuel assemblies of BWR cores at various operating conditions and power distributions. It is used to evaluate pressure drops, channel and bypass flow distributions, and MCPRs, as well as the hydraulic compatibility of fuel designs. The NRC has approved XCOBRA code in XN-NF-80-19(P)(A). The NRC reviewed the information provided in licensee's letter dated January 9, 1990 (Reference 47) regarding inclusion of water rod models in XCOBRA code and accepted the inclusion in NRC letter dated February 1, 1990 (Reference 48).

Analysis

The evaluations consisted of analyzing bottom-, middle-, and top-peaked axial power distributions shown in ANP-3859, Figure 3.1. The licensee stated that **[[**

]] In a letter dated July 18, 2022 (Reference 3), the licensee explained that if radial power in the hot channels were to be increased, the increased voiding would increase the two phase pressure drop in those channels. However, in order to maintain the core average power, the power in other channels will need to be decreased which will lead to a decrease in the two-phase pressure drop in those channels. All channels communicate to a common channel inlet as well as a common channel outlet which forces all pressure drops in the core to be equal. To maintain that equal pressure drop, the flow in the hot channels will be reduced while the flow in the cooler assemblies will be increased. While changes in radial power distributions will have some impact on core pressure drop, the core flow redistribution will mean that any impacts are likely to be small.

Based on the above explanation, the NRC staff finds that the **[[**

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The four combinations of power / flow state points analyzed are: (a) 100 percent power / 100 percent flow, (b) 100 percent power / 85 percent flow, (c) 77.6 percent power / 55 percent flow, and (d) 54.3 percent power / 37.3 percent flow. The analysis inputs for these state points are given in ANP-3859, Table 3.4. The analyzed core configurations defined in this table are as follows:

- Core Loading 1: Core consisting of approximately one-third ATRIUM 11 fuel with the remainder ATRIUM 10XM fuel representing a core with a single reload of ATRIUM 11 fuel.
- Core Loading 2: Core consisting of approximately two-third ATRIUM 11 fuel with the remainder ATRIUM 10XM fuel representing a core with two reloads of ATRIUM 11 fuel.

Core Loading 1 – 1/3rd ATRIUM 11 and 2/3rd ATRIUM 10 XM:

- ANP-3859, Table 3.4 provides the analysis design conditions at rated and off-rated state-points of the mixed core representing a single reload of ATRIUM 11 fuel.
- ANP-3859, Table 3.5 shows the results **[[**

  ]]
- ANP-3859, Figures 3.2 to 3.5 shows differences in assembly flow between the fuel designs as a function of assembly power level.
- ANP-3859, Tables 3.7 to 3.10 show core pressure drop and core bypass flow fraction.

Core Loading 2 – 2/3rd ATRIUM 11 and 1/3rd ATRIUM 10 XM:

- ANP-3859, Table 3.4 provides the analysis design conditions at rated and off-rated state-points of the mixed core representing two reloads of ATRIUM 11 fuel.
- ANP-3859, Table 3.6 shows the results **[[**

  ]]
- ANP-3859, Figures 3.6 to 3.9 show differences in assembly flow between the fuel designs as a function of assembly power level.
- ANP-3859, Tables 3.7 to 3.10 show core pressure drop and core bypass flow fraction.

<u>Results</u>

The licensee's results are presented in ANP-3859, Tables 3.5 to 3.10 and Figures 3.2 to 3.9 for bottom peaked power distribution. The licensee stated that the results for the middle-peaked and top-peaked axial power distributions have similar trends. ANP-3859, Tables 3.5 and 3.6

provide a summary of the thermal-hydraulic results for the two core configurations provided in Table 3.4. ANP-3859, Tables 3.7 to 3.10 provide a summary of results for the core configurations evaluated.

Technical Conclusion

Based on the licensee's thermal-hydraulic analysis results for the changes in the pressure drop and assembly flow in the first (one-third ATRIUM 11) and the second (two-third ATRIUM 11) reloads of ATRIUM 11, the ATRIUM 11 design is hydraulically compatible with the coresident ATRIUM 10XM design for the MELLLA+ operation. The NRC staff finds it acceptable that the ATRIUM 11 fuel is hydraulically compatible with the co-resident ATRIUM 10XM fuel because the acceptance criteria established in NRC staff-approved TRs ANF-89-98(P)(A), Revision 1, Supplement 1, and XN-NF-80-19(P)(A), Volume 4, Revision 1 are satisfied.

3.6.2.3 Thermal Margin Performance

The purpose of the thermal margin performance analysis is to determine if the fuel assembly geometry, including spacer design and rod-to-rod local power peaking, should minimize the likelihood of boiling transition during normal reactor operation as well as during AOOs. With all constraints, the thermal margin should be within the bounds of the applicable approved empirically based boiling transition correlation for the reload fuel.

Analysis

The licensee analyzed thermal margin performance using the NRC staff-approved XCOBRA code and calculated the MCPR of the ATRIUM 10XM and ATRIUM 11 fuel assemblies with radial peaking factors (RPFs) between **[[ ]]** The CPR for ATRIUM 10XM fuel is calculated using the ACE/ATRIUM empirical correlation in ANP-10298P-A and the CPR for ATRIUM 11 fuel is calculated using ACE/ATRIUM empirical correlation in ANP-10335P-A.

Results

ANP-3859, Tables 3.5 and 3.6 show representative CPRs of the ATRIUM 11 and ATRIUM 10XM fuel designs. ANP-3859, Tables 3.7 through 3.10 show similar comparisons of CPR and assembly flow for the various mixed core configurations evaluated.

Technical Conclusion

The NRC staff considers the licensee's thermal margin performance analysis and results acceptable for the following reasons:

- The licensee used an NRC staff-approved method and NRC staff-approved design-specific fuel assembly CPR correlations for ATRIUM 10XM and ATRIUM 11 fuels.

- The results show that **[[**

  ]]

- Based on the results, the NRC staff finds that ATRIUM 11 fuel will not adversely affect the thermal margin performance of the coresident ATRIUM 10XM fuel and appropriate thermal margins will be maintained throughout the transition.

3.6.2.4 Fuel Centerline Temperature

The thermal-hydraulic design criterion is that the centerline temperature of the fuel shall not exceed the melting point of the fuel in its design, during normal operation and AOOs. The licensee used the RODEX4 code described in NRC staff-approved TR XN-NF-80-19(P)(A). RODEX4 is a thermal-mechanical code for analyzing the fuel centerline temperature for the fuel rods. To limit the fuel centerline temperature below its melting point, the licensee established a LHGR during steady state operation and during AOOs. The licensee stated that analyses show that the fuel centerline temperature design criteria are satisfied when the fuel is operated at or below the LHGR limit. The NRC staff's evaluation of this criterion is addressed in section 3.5.2.4 of this SE.

3.6.2.5 Rod Bow

The thermal-hydraulic criteria require that the anticipated bow for the fuel rod under irradiation should be addressed **[[** **]]** The licensee addressed the rod bow in ANP-3860, section 3.3.5. The NRC staff's evaluation for rod bow is provided in section 3.4.6 of this SE.

The NRC staff finds that the above disposition is acceptable because the licensee used an NRC staff-approved model to compute the impact of rod bow on thermal limits and concluded that **[[** **]]**

3.6.2.6 Bypass Flow

The total core bypass flow is defined as leakage flow through LTP flow holes, channel seal, core support plate, and LTP-fuel support interface. A change in the bypass flow in transitioning may impact the **[[** **]]**

The licensee provided the bypass flow results at 100 percent power/100 percent flow (or rated condition) and at 100 percent power/85 percent flow conditions respectively in ANP-3859, Tables 3.7 and 3.8. Both tables provide the bypass flow results at full and mixed core loadings of the ATRIUM 10XM and ATRIUM 11 fuel. At the rated condition, excluding the water rod flow, the total core bypass flow changes from **[[** **]]** of rated core flow during transition. At the 100 percent power/85 percent flow condition, excluding the water rod flow, the core bypass flow changes from **[[** **]]** of rated core flow during the transition. The licensee stated that **[[** **]]**

The NRC staff finds that the licensee's analysis and results for the impact on bypass flow is acceptable because the flow **[[** **]]** The NRC staff therefore finds that adequate bypass flow will be available with the introduction of the ATRIUM 11 fuel design and the applicable design criteria are met.

3.6.2.7 Stability

The thermal-hydraulic design criteria approved by the NRC staff in ANF-89-98(P)(A) include a requirement to confirm that the stability characteristics for a new fuel design are equivalent to or better than that of prior approved fuel designs. The stability performance is a function of the core power, core flow, core power distribution and, to a lesser extent the fuel design. The licensee performed a comparative stability analysis using the NRC staff-approved STAIF code (Reference 49). In the analysis, the licensee kept the **[[**

]] The licensee stated that the results of the comparative stability analysis showed that the ATRIUM 11 fuel design has decay ratios equivalent to or better than other Framatome fuel designs. Since the stability performance of a core is strongly dependent on the core power, core flow, and power distribution in the core, the licensee intends to perform this evaluation on a cycle-specific basis and address it in the reload licensing report.

The NRC staff finds the licensee's evaluation meets the requirements within the NRC staff-approved ANF-89-98(P)(A) generic fuel assembly mechanical design criteria used by Framatome to qualify new fuel designs.

3.6.2.8 Void Fraction

ANP-3908, section 5.1 addresses the use of the **[[ ]]** correlation for ATRIUM 11 fuel. The NRC staff's evaluation is given in section 3.7.3.1 of this SE.

3.7 APPLICABILITY OF FRAMATOME BWR METHODS TO BROWNS FERRY WITH ATRIUM 11 FUEL (ANP-3908)

Applicability of the methods are addressed in the BWR compendium (Reference 44). While the NRC staff did not separately review and approve this reference, the staff reviewed it for applicability to the use of ATRIUM 11 fuel at Browns Ferry. Much of the methodologies discussed in the compendium have previously been confirmed to be applicable to ATRIUM 10 fuel, and also apply to use of ATRIUM 11 fuel because it is fundamentally an evolutionary fuel design with similar geometry and composition characteristics. The applicability of methodologies to specific safety analyses are addressed.

3.7.1 ATRIUM 11 Fuel Assembly Design

ATRIUM 11 fuel is an evolutionary design. **[[**

]]

The LTP utilizes the 3GFG filter insert **[[**

]] The licensee described each fuel design characteristic in Tables 3-1 and 3-2 in ANP-3908.

3.7.2 Thermal-Mechanical Limits Methodology

The LHGR limit is established to support plant operation while satisfying the fuel thermal-mechanical design criteria. Fuel rod evaluation is described in the associated methodology reports. The licensee takes into account the uncertainties of important physical phenomena such as operating power uncertainties, code model parameter uncertainties, and fuel manufacturing tolerances. In addition, adjustments are made to the power history inputs for possible differences in planned versus actual operation. Upper limits on the analysis results are obtained for comparison to the design limits for fuel melt, cladding strain, rod internal pressure and other characteristics as described by the design criteria.

Since the power history inputs, which include LHGR, fast neutron flux, reactor coolant pressure and reactor coolant temperature, are used as input to the analysis, the results explicitly account for conditions representative of the ATRIUM 11 operation. The resulting LHGR limit is used to monitor the fuel, so it is maintained within the same maximum allowable steady-state power envelope as analyzed.

3.7.3 Thermal Hydraulics

3.7.3.1 ATRIUM 11 Void Fraction

The ATRIUM 11 fuel uses the same void-quality correlation **[[   ]]** as ATRIUM-10 and ATRIUM 10 XM. Framatome qualified the correlation through testing and determined that the standard deviation is applicable to the evolutionary design of ATRIUM 11 fuel. The NRC staff has previously accepted this conclusion in prior amendments and finds that the rationale for finding this conclusion acceptable in those amendments applicable to this review.

3.7.3.2 ACE/ATRIUM 11 Critical Power Ratio Correlation

The licensee described the critical power correlation used in MICROBURN-B2, SAFLIM3D, S-REPLAP5, RAMONA5-FA, and X-COBRA in the NRC staff-approved TR ANP-10335P-A. A bounds checking process was used to determine the applicability of the correlation to ATRIUM 11. The results were similar to the results of ATRIUM 10XM and the staff determined that the same correlation is acceptable to use for ATRIUM 11 fuel. The ranges of applicability of the ACE/ATRIUM 11 and ACE/ATRIUM 10XM are compared in Table 5-2 of ANP-3908 (Attachment 12a to Reference 6).

3.7.3.3 Loss Coefficients

Framatome determined the wall friction and component loss coefficients for Browns Ferry based on single-phase testing of a prototype ATRIUM 11 fuel assembly at the PHTF. The PHTF tests form the basis for the single-phase loss coefficients used in the ATRIUM 11 design. The wall friction and component loss coefficients determined from the PHTF and utilized in the validation of the MICROBURN-B2 pressure drop model for the ATRIUM 11 fuel design are provided in Table 5-3 of ANP-3908.

The modeling of the two-phase spacer pressure drop multiplier for the ATRIUM 11 fuel design has been confirmed with two-phase pressure drop measurements taken in the KATHY facility. Figure 5-1 of ANP-3908 shows measured versus the MICROBURN-B2 predicted two phase pressure drop for a range of conditions. This figure confirms the applicability of the thermal-hydraulic models to predict pressure drop for the ATRIUM 11 design. The NRC staff's evaluation conclusions are provided in section 3.6.2.1 of this SE.

3.7.4 Transients and Accidents

3.7.4.1 Void Calculation Uncertainties

The Framatome analyses methods and the correlations used are applicable for all Framatome designs in EPU conditions. The approach for addressing bias and uncertainties in the void calculation remains unchanged and is applicable for Browns Ferry operation with the ATRIUM 11 fuel design.

The impact of void prediction uncertainty is inherently incorporated in the methodologies. The operating limit minimum critical power ratio (OLMCPR) limit is determined based on the SLMCPR methodology and the transient analysis methodology. The methodologies have been found acceptable to use with ATRIUM 11 fuel in ANP-10344P-A (Reference 28). Additionally, to add conservatism, the licensee biases the input parameters in licensing calculations to establish the OLMCPR.

3.7.4.2 [[

]] The following sections provide the discussion.

3.7.4.3 Transient Mixing Determination

For Browns Ferry, the mixing is evaluated using **[[**

]]

3.7.4.4 Implementation in AURORA-B AOO

The AURORA-B licensing model is constructed after the amount of mixing has been determined. The AURORA-B SE discusses how a conservative model is ensured. **[[**

]]

3.7.5 AURORA-B AOO Time Step Size

The time step size was reviewed by the NRC staff in the AURORA-B methodology. Browns Ferry will comply with the methodology. **[[**

]]

3.7.6 ATWS

3.7.6.1 ATWS General

The AURORA-B methodology is used for the ATWS overpressurization analysis. Dryout might occur in the limiting (high power) channels of the core during the ATWS event. For the ATWS overpressurization analysis, ignoring dryout for the hot channels is conservative in that it maximizes the heat transferred to the coolant and results in a higher calculated pressure.

The ATWS event is not limiting relative to acceptance criteria identified in 10 CFR 50.46 and the core remains covered and adequately cooled during the event. The ATWS event is significantly less limiting than the loss of coolant accident relative to 10 CFR 50.46 acceptance criteria.

3.7.6.2 Void Prediction

Framatome performs cycle-specific ATWS analyses of the short-term reactor vessel peak pressure using the AURORA-B methodology. The ATWS peak pressure calculation is a core-wide pressurization event that is sensitive to similar phenomenon as other pressurization transients. Bundle design is included in the development of input for the coupled neutronic and thermal-hydraulic S-RELAP5 core model. Important inputs to the S-RELAP5 system model are biased in a conservative direction. The void prediction is robust for past and present fuel designs, including ATRIUM 11, as discussed in NRC staff-approved TR ANP-10300P-A (Reference 33).

3.7.6.3 ATWS Containment Heatup

Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This in turn may impact the amount of steam discharged to the suppression pool and containment.

[[

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3.7.7 Neutronics

The ATRIUM 11 fuel design is different from ATRIUM 10XM primarily in the fuel rod diameter and pitch. The CASMO-4 code is designed to model a wide range of fuel rod diameters and pitches. The neutronics models have demonstrated accuracy for the ATRIUM 11 fuel design.

3.7.7.1 Shutdown Margin

No change in the predicted hot operating or cold critical eigenvalue is anticipated with the ATRIUM 11 fuel design. The shutdown margin performance is improved by the part-length rod in the corner of the assembly because a flux trap is created in the cold condition.

3.7.7.2 Monitoring

The part-length rod in the corner of the assembly has an impact on the corner flux that influences the detector response. For the Browns Ferry analyses, the plena have been explicitly modeled with the heterogeneous CASMO-4 model, thus providing the most accurate model available. The heterogeneous solution of CASMO-4 accurately calculates this corner flux depression and this characterization is used directly in the MICROBURN-B2 determination of the predicted detector response.

3.7.7.3 Bypass Modeling

The bypass behavior of the ATRIUM 11 fuel design is identical to the ATRIUM 10XM fuel design, thus there is no difference in the modeling.

3.7.7.4 Vessel Fluence

Framatome compared the ATRIUM 11 and ATRIUM 10XM peripheral fast fluence using equilibrium cycle designs. The comparison showed that the ATRIUM 11 design is bound by the ATRIUM 10XM design. The implementation of ATRIUM 11 fuel will not have an adverse impact on the existing vessel fluence evaluation or to the lower and upper core components. The licensee provided results of these comparisons in Figures 8-1 and 8-2 in ANP-3908 (Attachment 12a to Reference 6).

3.7.8 Mixed Cores

The mixed core analyses are performed using a generically approved methodology in a manner consistent with NRC approval of the methodology. Based on results from the analyses, operating limits are established for each fuel type present in the core. During operation, each fuel type is monitored against the appropriate operating limits.

For Browns Ferry, the entire core will be composed of Framatome fuel. Thermal hydraulic characteristics are determined for each fuel type that will be present in the core and used in the core design, safety analysis and core monitoring.

For neutronics, each fuel assembly is explicitly modeled in MICROBURN-B2 using cross-section data from CASMO-4 and geometric data appropriate for the fuel design.

3.7.9 Conclusion

The NRC staff's considerations of the approach used to perform the applicability of Framatome BWR Methods to Browns Ferry are documented in the above subsections. As a result, the NRC staff finds that evaluations have been performed acceptably to ensure that the methods used for the ATRIUM 11 fuel design are acceptable for use in the Browns Ferry reactor cores.

3.8 LOCA ANALYSIS FOR ATRIUM 11 FUEL (ANP-3905)

NRC regulations require that licensees of operating LWRs analyze a spectrum of accidents involving the loss of reactor coolant to assure adequate core cooling under the most limiting set of postulated design-basis conditions. The postulated spectrum of LOCAs range from scenarios with leakage rates just exceeding the capacity of normal makeup systems to those involving rapid coolant loss from the complete severance of the largest pipe in the reactor coolant system.

3.8.1 Applicable Regulatory Requirements

The following regulatory requirements described below are pertinent to the analysis of the spectrum of LOCA events postulated to occur:

- 10 CFR 50.46
- 10 CFR Part 50; Appendix A, GDC 35 (1967 AEC Draft GDC 37, 41, and 44)
- 10 CFR Part 50, Appendix K

10 CFR 50.46

Key regulatory requirements specified in 10 CFR 50.46 that are relevant to the proposed license amendments include the following:

- Each boiling or pressurized light-water reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must perform analysis of core cooling performance under postulated LOCA conditions using an acceptable evaluation model.

- An acceptable LOCA evaluation model must be used that either applies realistic methods with an explicit accounting for uncertainties or follows the prescriptive, conservative requirements of Appendix K to 10 CFR Part 50.

- Core cooling performance must be analyzed for several postulated LOCAs of different sizes, locations, and other characteristics to ensure that the most severe event is calculated.

10 CFR 50.46(b) Acceptance Criteria:

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The licensee referred to acceptance criteria (1) through (5) as the peak clad temperature (PCT) criterion, the maximum local oxidation (MLO) criterion, the hydrogen generation (or core wide oxidation (CWO)) criterion, the coolable geometry criterion, and the long-term cooling criterion respectively. A maximum average planar LHGR (MAPLHGR) limit is established for each fuel type to ensure these criteria are met.

In accordance with limitation and condition 4 from the NRC staff's final SE on TR ANP-10332P-A, Revision 0 (Reference 50), the AURORA-B LOCA evaluation model may not be referenced as a basis for demonstrating adequate long-term core cooling in satisfaction of 10 CFR 50.46(b)(5). To demonstrate continued adherence to this requirement, the licensee cited existing licensing basis analysis performed on a generic basis by the nuclear reactor vendor (i.e., General Electric), which is documented in approved TR NEDO-20566A (Reference 51). Accordingly, the proposed license amendments would not modify the licensing basis method for demonstrating satisfaction of the requirement in 10 CFR 50.46(b)(5) for adequate long-term core cooling.

The AURORA-B LOCA evaluation model uses models and computer codes approved for reactor licensing analyses by the NRC staff, including those in the approved Exxon Nuclear Company TR XN-NF-82-07(P)(A), Revision 1 (Reference 52). The letter accompanying the SE for this TR states:

> Should Nuclear Regulatory Commission criteria or regulations change, such that our conclusions as to the acceptability of the report are invalidated, Exxon Nuclear Company, Inc., and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Consistent with this statement, site-specific implementation of the TR must ensure that subsequent changes to the NRC regulations identified in the TR (i.e., 10 CFR 50.46 and 10 CFR Part 50, Appendix K) do not affect the acceptability of the TR.

Because the NRC criteria or regulations have not changed such that prior NRC staff conclusions as to the acceptability of XN-NF-82-07(P)(A) are invalidated, the NRC staff finds that the TR is applicable.

The NRC staff notes that the applicability of XN-NF-82-07(P)(A) models to modern fuel designs was considered during the review and approval of ANP-10332P-A, Revision 0, as documented in Sections 3.3.4.1.2, 3.6.2.4, and 3.6.2.7 of that report's SE (Reference 50). As a result of this consideration, the NRC staff imposed limitation and condition 11. The NRC staff's evaluation of the licensee's compliance with limitation and condition 11 is provided in Section 3.8.6 of this SE.

10 CFR Part 50, Appendix A, GDC 35

Criterion 35—Emergency core cooling states:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

10 CFR Part 50, Appendix K

Appendix K to 10 CFR Part 50 consists of two parts:

- required and acceptable features of LOCA evaluation models and
- documentation required for LOCA evaluation models.

The first part specifies modeling requirements and acceptable methods for simulating significant physical phenomena throughout all phases of a design-basis LOCA event, including relevant heat sources, fuel rod performance, and thermal-hydraulic behavior.

The second part specifies requirements for the documentation of LOCA evaluation models, including a complete description, a code listing, sensitivity studies, and comparisons against experimental data.

The NRC staff's basis for concluding that the AURORA-B LOCA evaluation model used to perform the LOCA analysis for Browns Ferry conforms to the requirements of Appendix K to 10 CFR Part 50 is discussed in section 6.2.1 of the NRC staff's SE on ANP-10332P-A (see Reference 50).

3.8.2 LOCA Analysis

The LOCA event response is divided into 3 phases: the blowdown phase, the refill phase, and the reflood phase. The licensee described these LOCA phases in ANP-3905, Revision 2

(Attachment 12a to Reference 6), section 3.1. To support the planned transition to ATRIUM 11 fuel, the licensee analyzed the spectrum of postulated LOCA events to verify the satisfaction of applicable regulatory requirements following the transition to ATRIUM 11 fuel. The licensee used the AURORA-B LOCA Evaluation Model (EM) to demonstrate compliance with the acceptance criteria in 10 CFR 50.46(b)(1) through (b)(4) that apply to the short-term LOCA analysis.

3.8.3 Methodology

The AURORA-B LOCA methodology described in NRC staff-approved TR ANP-10332P-A is an Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 analysis methodology. The methodology is based on S-RELAP5 that incorporates a kernel of transient fuel rod thermal-mechanical subroutines from the RODEX4 code. The fuel parameters are specified using RODEX4 code which is documented in BAW-10247P-A (References 42, 53, and 38). RODEX4 is used to determine the **[[**

]] The initial stored energy used in S-RELAP5 is **[[**
]]

As documented in the SE on TR ANP-10332P-A, the AURORA-B LOCA EM is acceptable for application to LOCA analysis for BWR/3-BWR/6 plants, and therefore is applicable because the Browns Ferry units are BWR/4 plants.

3.8.4 Break Spectrum Analysis

The purpose of the break spectrum analysis is to identify the break characteristics that result in the highest calculated PCT **[[** **]]** during a postulated LOCA. The results of the analysis provide the MAPLHGR limit for ATRIUM 11 fuel as a function of exposure for normal, i.e., two-loop operation (TLO).

The licensee performed the break spectrum analysis over a range of break locations, break sizes, break types (double-ended guillotine (DEG), split), initial state points, axial power shapes (top-peaked or mid-peaked) and assumed single-failures to determine the break that yields the highest PCT **[[**
]]

The following is a summary of the assumptions and attributes of the licensee's break spectrum analysis:

- The analysis is based on a full core of ATRIUM 11 fuel.
- Reactor thermal power is assumed to be 102 percent of the rated thermal power to address the maximum measurement uncertainty.
- **[[** **]]**
- **[[**
  ]]

- **[[** **]]** were assumed to be at the MAPLHGR limit shown in Figure 2.1 of ANP-3905.

- **[[** **]]** of ANP-3905.

- With operation in the MELLLA+ domain shown in ANP-3905, Figure 1.1, the licensee **[[** **]]:**

  (a) **[[** **]]**
  (b) **[[** **]]**
  (c) **[[** **]]**

- With operation in the MELLLA+ domain shown in ANP-3905, Figure 1.1, the licensee **[[** **]]**

- **[[** **]]**

- Used S-RELAP5 code which incorporates the clad swelling and rupture models from NUREG-0630 (Reference 54) to calculate the thermal-hydraulic response during all phases of LOCA.

- The following tables in ANP-3905 provide inputs used in the break spectrum analysis:

  - Table 4.1 shows reactor initial conditions
  - Table 4.2 shows reactor system parameters
  - Table 4.3 shows ATRIUM 11 fuel assembly parameters
  - Table 4.4 shows HPCI system parameters
  - Table 4.5 shows low pressure coolant injection (LPCI) system parameters
  - Table 4.6 shows low pressure core spray (LPCS) parameters
  - Table 4.7 shows automatic depressurization system (ADS) parameters
  - Table 4.8 shows recirculation discharge isolation valve parameters

- ANP-3905, Figure 4.1 shows the reactor vessel nodalization, Figure 4.2 shows the core nodalization used in the analysis which are consistent with those in NRC approved TR ANP-10332P-A, and Figure 4.3 shows the ECCS schematic.

- **[[** **]]**

- ECCS initiation is assumed to occur when the water level drops to the applicable water level setpoint (ANP-3905, Tables 4.4 and 4.5) and conservatively the analysis does not credit ECCS flow until the ECCS injection valves open and the ECCS pumps reach rated speed.

- Conservatively HPCI, LPCS, and LPCI are not initiated based on the drywell pressure exceeding its high pressure setpoint.

The licensee evaluated the recirculation line breaks and non-recirculation break LOCAs. The NRC staff's evaluation of the consequence of the breaks is given below.

<u>Reactor Recirculation Line Breaks During Two-Loop Operation</u>

For evaluating the ECCS performance during LOCAs, consistent with 10 CFR 50.46 and 10 CFR Part 50, Appendix K, the licensee analyzed a spectrum of possible pipe breaks up to and including the instantaneous DEG break and longitudinal splits in the recirculation system pipes with the split area equal to the cross-sectional area of the pipe. The break types and sizes along with consideration of single failures are analyzed for both suction and discharge recirculation pipe breaks. The largest diameter recirculation pipes are the suction line between the reactor vessel and the recirculation pump and the discharge line between the recirculation pump and the riser manifold ring. The LOCA analyses are performed for breaks in both locations for DEG and split breaks. The break areas analyzed range between full pipe area and **[[ ]]** square feet (ft^2) with discharge coefficients from 1.0 to 0.4. The licensee stated that the range of discharge coefficients is used to cover uncertainty in the actual geometry at the break. **[[**
]] The licensee determined the most limiting DEG break is by varying the discharge coefficient.

ANP-3905, Table 5.1 identifies the following single failures (SFs) considered for LOCA analysis: (a) failure of one train of direct current (DC) power or battery (BATT) (i.e., SF-BATT|BA, or SF-BATT|BB, or SF-BATT|BC), (b) opposite unit false LOCA signal (i.e., SF-LOCA), (c) failure of a LPCI system injection valve (i.e., SF-LPCI), (d) failure of a diesel generator (DGEN) (i.e., SF-DGEN), (e) failure of a HPCI system (i.e., SF-HPCI), (f) failure of ADS valve initiation logic (IL) (i.e., SF-ADS|IL), and (g) failure of a single valve (SV) of ADS (i.e., SF-ADS|SV).

In ANP-3905, Tables 5.1 and 5.2, for the recirculation and non-recirculation line breaks respectively, after considering each of the above SFs, the licensee listed the remaining ECCS that would be available for LOCA mitigation. The ECCS resources available for SF-BATT|BA and SF-BATT|BB are the least and, therefore, the licensee determined that analysis is needed, whereas for the remaining SF cases, the licensee determined that the analysis is not needed as their PCTs would be equal or bounded because of equivalent or higher availability of the ECCS. The NRC staff's review found that this determination was appropriate, and that the licensee considered all postulated single failures defined in the Browns Ferry UFSAR Table 6.5-3.

The SF cases SF-BATT, SF-DGEN, SF-HPCI, and SF-ADS use either one LPCI (2 pumps) loop or two LPCI (4 pumps) loops. The following is a description of a scenario of concern noted by the NRC staff which could occur during normal plant operation

When the residual heat removal (RHR) system is placed in suppression pool cooling/mixing mode or flow test mode is not placed in an LCO, the system test line isolation valve through which water returns to the suppression pool is open. While in these modes, a LOCA signal opens the LPCI injection valve fully in 40 seconds and closes the test line isolation valve in 90 seconds (UFSAR section 7.4.3.5.4). During the 50 seconds time difference between the closing time of the test line isolation valve and the opening time of the LPCI injection valve, some of the LPCI flow will be diverted to the suppression pool and therefore the reactor will not receive the fully rated LPCI flow.

To address the above concern, in a letter dated July 18, 2022 (Reference 3), the licensee stated that there is another valve downstream of the 90 seconds stroke time valve that is partially closed during the test and is used to throttle the flow from the LPCI pump during flow testing. This valve is either FCV-74-59 or FCV-74-73 (UFSAR Figure 7.4-6a, Sheet 1), depending on which pair of LPCI pumps are being tested. During testing, this valve is positioned such that it will stroke from the test position to fully closed in a maximum time of 49 seconds, which is significantly shorter than the 90 second closing stroke time of the upstream valve. Therefore, valve FCV-74-59 or FCV-74-73 determines the time at which the affected return line is fully isolated on a LOCA signal. These two valves have an associated surveillance test procedure that ensures their 49 second closing time criterion is met. In the analysis presented in ANP-3905, the licensee did not consider the LPCI flow diversion for the small time during a LOCA in the presence of a loss of offsite power, which initiates when the suppression pool return line isolation valves are open. To determine the impact of the delayed fully rated LPCI flow for the pump suction breaks cases, the licensee performed bounding sensitivity calculations using a 49 second LPCI valve opening time without taking any credit for LPCI flow until the valve is fully open. This delays the LPCI injection until after the test line isolation valve closes and the LPCI valve fully opens. The pump suction cases with the highest PCTs are the 1.0 DEG breaks, which depressurize below the LPCI pressure permissive before power is available to the valves and maximize the potential impact of a LPCI delay. The sensitivity calculation results are presented in Table 2.2 of Reference 3. The results for a 1.0 DEG pump suction break for both axial power shapes (mid-peaked and top-peaked) and the two SF cases (SF-BATT|BA and SF-BATT|BB) show that the PCTs remains substantially non-limiting and do not affect the limiting PCT or oxidation results reported in ANP-3905 and are given below.

To determine an exposure-dependent MAPLHGR limit, and **[[**
]], the licensee analyzed the **[[**

]] from beginning-of-life to end-of-life **[[** **]]** increments as shown in ANP-3905, Table 9.1. To confirm the acceptability of the LOCA analysis with respect to 10 CFR 50.46 criteria, the licensee calculated PCT, MLO, and CWO over the range of exposures. The licensee also **[[**

]] The cases in ANP-3905, **[[**
]] using the S-RELAP5 model. The analysis is performed at **[[**
]] The MAPLHGR input is consistent with the data in ANP-3905, Figure 2.1. **[[** **]]** Exposure-dependent fuel rod data is provided from RODEX4 results. The impact of thermal conductivity degradation (TCD) is addressed using RODEX4.

The NRC staff finds that the licensee's exposure-dependent analysis generally conforms to the approved EM documented in ANP-10332P-A. In particular, the exposure study analyzed **[[**
]] accounting for exposure-dependent limiting values of the

[[**]]** and MAPLHGR. The exposure study deviated from the methodology approved in the NRC staff's SE on ANP-10332P-A in that at **[[**

]] However, because these **[[** **]]** do not produce limiting PCT and MLO results in the analysis under review, the NRC staff finds this deviation from the approved EM acceptable for the LOCA analysis.

The NRC staff finds the licensee's TLO break spectrum analysis acceptable based on the following:

- The NRC staff-approved S-RELAP5 methodology is used for without exposure and exposure-dependent analysis from beginning-of-life to end-of-life in appropriate increments.

- The analysis is performed for a spectrum of pipe breaks up to and including the instantaneous DEG break and longitudinal split breaks consistent with 10 CFR 50.46, and 10 CFR Part 50, Appendix K.

- SFs are appropriately assumed in each analysis.

- **[[** **]]**, which ensures appropriate limits are applied up to the monitored maximum assembly average and rod average exposure limits.

- Exposure-dependent fuel rod data is provided by the NRC staff-approved RODEX4 code while including the effect of TCD of fuel.

<u>Reactor Recirculation Line Breaks During Single-Loop Operation</u>

In a single-loop operation (SLO), the loop in which the recirculation pump is not operating is the inactive loop while the one in which the pump is operating is the active loop. The licensee stated that the PCT results for a break in the inactive loop would be like those from a similar TLO break because flow to the reactor vessel would continue during the active loop pump coastdown period and would provide core cooling. A break in the active loop causes a more rapid loss of core flow and causes fuel rod surface temperatures to increase faster as compared to a break in an inactive loop. Therefore, the licensee only analyzed breaks in the active loop. The NRC staff finds this acceptable because of the similar results for a TLO break with a SLO break in the inactive loop, the SLO LOCA in the active loop would be more limiting compared to a TLO LOCA.

[[

]] kW/ft which is equal to applying a 0.85 multiplier as a reduction factor to the two-loop MAPLHGR of **[[** **]]** **[[** **]]** The SLO breaks are not analyzed in the MELLLA+ region because SLO is not permitted in this domain.

The NRC staff finds the licensee's SLO break spectrum analysis acceptable based on the following:

- The NRC staff-approved S-RELAP5 methodology is used without exposure and exposure-dependent analysis from beginning-of-life to end-of-life in appropriate increments.
- An appropriate multiplier (or reduction factor) (i.e., 0.85) is applied to the TLO MAPLHGR. In a letter dated July 18, 2022 (Reference 3), the licensee stated that the reduction factor is defined so that the SLO PCT is bounded by the TLO PCT.

Main Steam Line Breaks

The licensee described the large main steam line break inside the containment in ANP-3905, section 5.3.1 and stated that **[[**

]]

The NRC staff finds it acceptable that **[[** **]]** because during the blowdown period from the break, **[[** **]]** In this condition, **[[** **]]**

Feedwater Line Breaks

The licensee described the feedwater line break inside the containment in ANP-3905, section 5.3.2 and stated that **[[**

]]

The NRC staff finds it acceptable **[[**

]]

HPCI Line Breaks

The licensee described the HPCI line break inside the containment in ANP-3905, section 5.3.3, and stated that this line is connected to the feedwater line outside containment and, **[[** **]]** The HPCI steam supply line is connected to the main steam line inside containment and, **[[** **]]**

The NRC staff finds it acceptable that [[

]]

LPCS Line Breaks

The licensee described the LPCS line break inside the containment in ANP-3905, section 5.3.4. The break is assumed to occur just outside the reactor vessel. The licensee [[

]]

Based on the licensee's [[

]]

LPCI Line Breaks

The licensee described the LPCI line break inside the containment in ANP-3905, section 5.3.5. The LPCI injection lines are connected to the larger recirculation discharge lines. [[

]]

The NRC staff finds it acceptable that the LPCI line break would be nonlimiting relative to the acceptance criteria because it is [[

]]

Reactor Core Isolation Cooling (RCIC) Line Breaks

The licensee described the RCIC line break inside the containment in ANP-3905, section 5.3.6, and stated that the RCIC discharges to the feedwater line; [[

]]

The NRC staff finds it acceptable that the RCIC steam or liquid line breaks would be nonlimiting relative to the acceptance criteria because [[

]]

Reactor Water Cleanup (RWCU) Line Breaks

The licensee described the RWCU line break inside the containment in ANP-3905, section 5.3.7 and stated that the [[

]]

The NRC staff finds it acceptable that the RWCU extraction and return line breaks would be nonlimiting relative to the acceptance criteria because [[

]]

Shutdown Cooling Line Breaks

The licensee described the shutdown cooling line break inside the containment in ANP-3905, section 5.3.8, and stated that the shutdown cooling suction piping is connected to a recirculation line and the shutdown cooling line is connected to a recirculation discharge line; **[[**

]]

The NRC staff finds it acceptable that the shutdown cooling piping break PCT would be nonlimiting relative to the acceptance criteria because **[[**
]]

Instrument Line Breaks

The licensee described the instrument line break inside the containment in ANP-3905, section 5.3.9 and stated that **[[**

]]

The NRC staff finds it acceptable that **[[** **]]** instrument line breaks PCT would be nonlimiting relative to the acceptance criteria because **[[**
]]

Transition Cores

The licensee stated that **[[**

]]

3.8.5 Results

ANP-3905, Table 7.1 summarizes the limiting case PCT results for SLO and TLO. The PCTs for SLO and TLO are **[[** **]]** °F and **[[** **]]** °F respectively both based on a pump discharge break area of 3.5 ft^2, SF-BATT│BB, and top-peaked axial power shape.

ANP-3905, Table 9.1 shows the **[[**
]]

ANP-3905, Table 6.1 describes the following parameters for the TLO LOCA analysis case that produced the **[[** **]]** result:

- Break Area 3.5 ft^2 (square feet)
- Break Location Pump Discharge
- Single Failure SF-BATT│BB

- Power 102% Rated Core Power
- **[[ ]]** **[[ ]]**
- Power Shape Top-Peaked Axial

ANP-3905, Table 6.1 describes the following parameters for the TLO LOCA analysis case that produced the **[[ ]]** result:

- **[[**
-
-
-
-
- **]]**

ANP-3905, Table 9.1 provides the exposure-dependent LOCA analysis results. Table 1 below summarizes the maximum PCT and MLO based on the PCT and MLO case parameters listed above.

Table 1: TLO Exposure-Dependent LOCA Analysis Results

[[				**]]**	10 CFR 50.46(b) Acceptance Criterion
[[				**]]**	≤ 2,200 °F
[[				**]]**	≤ 17%
CWO < 0.73% at all exposures meets 10 CFR 50.46(b)(3) acceptance criterion of ≤ 1%					

The results show that the 10 CFR 50.46(b)(1) through (b)(3) acceptance criteria for PCT ≤ 2200ºF, local oxidation ≤ 17 percent, and hydrogen generation (core wide oxidation) ≤ 1 percent respectively are satisfied. The results also demonstrate the acceptability of the ATRIUM 11 MAPLHGR limit shown in ANP-3905, Figure 2.1.

For 10 CFR 50.46(b)(4), the licensee stated:

- Compliance with 10 CFR 50.46(b)(1) through (b)(3) ensures that the core coolable geometry is maintained.

- As described in ANP-3860, Table 3-1, for a combination of seismic and LOCA loads, the licensee used NRC guidance in NUREG-0800, SRP 4.2, Appendix A.

The NRC staff finds that compliance with 10 CFR 50.46(b)(4) is achieved based on the satisfaction of 10 CFR 50.46(b)(1) through (b)(3) acceptance criteria and maintaining of coolable core geometry under a combination of seismic and LOCA loads.

For compliance with 10 CFR 50.46(b)(5) regarding long-term coolability, the licensee stated:

- For non-recirculation line breaks, the core can be reflooded to the top of the active fuel and be adequately cooled indefinitely.

- For the recirculation line breaks, the core will initially remain covered following reflood due to the static head provided by the water filling the jet pumps to a level of approximately two-thirds core height. Eventually, the heat flux in the core will not be adequate to maintain a two-phase water level over the entire length of the core. Beyond this time, the upper third of the core will remain wetted and adequately cooled by core spray. However, since the fuel temperatures during long-term cooling are low relative to the PCT, the long-term temperatures are not significantly affected by fuel design. Therefore, as demonstrated in NEDO-20566A, the conclusion of maintaining water level at two-thirds core height with one core spray system operating is sufficient to maintain long-term coolability would also be applicable to the ATRIUM 11 core.

The NRC staff finds the licensee's explanation for long-term coolability acceptable because the fuel temperatures in the long-term are low relative to the PCT and are not significantly affected by the fuel design and therefore the licensee's conclusions in NEDO-20566A described above would be applicable to the ATRIUM 11 core.

3.8.6 Compliance with NRC Imposed Limitations and Conditions in Safety Evaluation for ANP-10322P-A, Revision 1

The AURORA-B LOCA methodology has 27 limitations and conditions listed in section 5.0 of the NRC staff's SE for ANP-10322P-A, Revision 1. ANP-3905, Appendix A provides the licensee's disposition on how the requirements in these limitations and conditions are met. The following is the NRC staff's evaluation to confirm these limitations and conditions are satisfied:

Limitation and Condition 1

> The AURORA-B LOCA evaluation model shall be supported by an approved nodal core simulator and lattice physics methodology. Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall identify the nodal core simulator and lattice physics methods supporting the AURORA-B LOCA analysis and reference an NRC-approved TR confirming their acceptability for the intended application.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee used nodal core simulator and lattice physics methodology from the NRC staff-approved TR EMF-2158(P)(A) Revision 0 (Reference 34).

Limitation and Condition 2

> The full, stand-alone version of the RODEX4 code shall be used in accordance with an approved methodology to supply steady-state fuel thermal-mechanical inputs to the AURORA-B LOCA evaluation model.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee used the stand-alone version of RODEX4 to provide the steady-state fuel thermal-mechanical input in accordance with the NRC staff-approved methodology BAW-10247PA, Revision 0 (References 42, 53, and 38).

Limitation and Condition 3

> The AURORA-B LOCA evaluation model may not be used to perform analyses that result in any of its constituent components or supporting codes (i.e., S-RELAP5, RODEX4 kernel, RODEX4, core simulator and lattice physics methods) being operated outside approved limits documented in their respective TRs, SEs, code manuals, and plant-specific licensing applications.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee's analyses are within the limits of the TRs, SEs, code manuals, and plant-specific licensing applications.

Limitation and Condition 4

> TR ANP-10332P does not provide a technical basis to support satisfaction of the requirement in 10 CFR 50.46(b)(5) for long-term core cooling, and, as such, has not been approved for this purpose.

Evaluation

The NRC staff's evaluation is provided in section 3.8.5 of this SER.

Limitation and Condition 5

> As discussed above in Section 2.1, the conclusions of this SE apply only to the use of the AURORA-B LOCA evaluation model for the purpose of demonstrating compliance with relevant regulatory requirements in effect at the time the NRC staff's technical review of ANP-10332P was completed (i.e., as of December 31, 2018).

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee's analyses apply and meet the regulatory requirements in effect at the time the NRC staff's review for ANP-10332P-A was completed.

Limitation and Condition 6

> This SE does not constitute approval of the PIRT [phenomena identification and ranking table] rankings in Table 4-1 of ANP-10332P. Framatome's PIRT rankings represent an informed opinion of phenomenon importance that the NRC staff referred to as supporting information in its review of the AURORA-B LOCA evaluation model; beyond this, they remain subjective judgments that are not integral to the acceptability of the evaluation model.

Evaluation

The licensee stated that the AURORA-B EM **[[**
]] The NRC staff, therefore, finds this limitation and condition is

satisfied because, as noted in the limitation and condition, the PIRT rankings were considered to be supporting information for the AURORA-B EM and were not reviewed and approved by the NRC staff. The PIRT rankings were provided for information.

Limitation and Condition 7

> **[[**
>
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied because conservatively the licensee **[[** **]]**

Limitation and Condition 8

> **[[**
>
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied because **[[** **]]** in the analyses.

Limitation and Condition 9

> Safety analyses performed with the AURORA-B LOCA evaluation model may not credit a limit on **[[** **]]**, absent a plant-specific determination from the NRC staff that such credit is consistent with the requirements of 10 CFR 50.36. Absent such a determination, **[[**
>
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee **[[**

]]

Limitation and Condition 10

> To ensure adequate conservatism in future plant-specific safety analyses, absent specific NRC staff approval for higher values, this SE limits credit for gamma energy deposition outside of a fuel rod to no more than **[[**
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's statement in ANP-3905, Appendix A, in disposition of this limitation and condition, that a **[[**
]]

Limitation and Condition 11

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall adequately justify the averaging method for determining the temperature ramp rate used in the calculation of cladding swelling and rupture.

Evaluation

The BWR fuel rods have a **[[**

]] Therefore, NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 12

> The Appendix K lockout preventing the return to nucleate boiling shall be **[[**
> **]]**

Evaluation

The analyses **[[**
]] Therefore, the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 13

> **[[**
> **]]** shall be taken into account when determining the start of the refill and reflood phases and the release of Appendix K heat transfer lockouts. (Section 3.3.4.1.4)

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee conservatively **[[**
]]

Limitation and Condition 14

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[[**
>
> **]]** When figures of merit are reported in licensing submittals to the NRC, they shall show results for **[[**
>
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's analyses which **[[**

]]

Limitation and Condition 15

> **[[**
>
> **]]** (Section 3.3.4.2.1)

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's statement in its disposition that in the analysis the **[[**

]]

Limitation and Condition 16

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall justify that the input conditions assumed in the analysis are bounding across the entire approved operating domain, which may include, for example, extended power uprates, extended flow windows, equipment out of service (e.g., automatic depressurization system valves, feedwater heaters, single-loop operation), and feedwater temperature reduction. If necessary, analysis of multiple initial operating states shall be performed to ensure that the most limiting conditions with respect to the acceptance criteria of 10 CFR 50.46 have been calculated.

Evaluation

Considering Browns Ferry is licensed to the MELLLA+ operating domain, sufficient initial state points were considered in the break spectrum analysis to provide confidence that the most limiting conditions have been analyzed. **[[**

]]
Therefore, the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 17

> To assure satisfaction of GDC 35 (or similar plant-specific design criterion), **[[**
>
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied because the results of **[[**
]]

Limitation and Condition 18

> Safety analyses performed with the AURORA-B LOCA evaluation model shall include justification for any credit taken for the drywell high pressure trip signal.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee **[[**
]]

Limitation and Condition 19

> Safety analyses for mixed-core configurations shall appropriately justify application of the AURORA-B LOCA evaluation model and any supporting methodologies (e.g., nodal core simulator and lattice physics methods, fuel thermal-mechanical performance methods) to legacy fuel assemblies designed by other vendors. Furthermore, **[[**
>
> **]]**
> (Section 3.3.5.5)

Evaluation

The NRC staff finds the response to a LOCA and the resulting 10 CFR 50.46(b) FoMs using the AURORA-B LOCA methodology are not significantly different for transition cycles to a full ATRIUM 11 core because of the following:

- **[[** **]]**
- The licensee's **[[**

-
-

]]

Based on the above, the NRC staff finds the limitation and condition is satisfied

Limitation and Condition 20

> Simulations supporting plant safety analyses should be run to completion of quenching on all potentially limiting fuel rods. If premature termination occurs, **[[**
>
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's statement in ANP-3905, Appendix A, in the disposition of this limitation and condition, that simulations **[[ ]]**

Limitation and Condition 21

> As discussed in Section 3.3.5.7 [of ANP-10332P-A SER], Framatome used a non-representative modeling practice of **[[**
>
> **]]** Prior to implementing this practice in future plant safety analyses, the practice must be adequately defined in the AURORA-B LOCA modeling guidelines. Furthermore, this practice may not be implemented in the safety analysis for any given plant without explicit plant-specific approval by the NRC staff (e.g., in conjunction with a license amendment request to implement the AURORA-B LOCA evaluation model). Licensees requesting credit for this non-representative modeling practice must adequately describe the extent of its intended use and justify its conservatism. The justification must address the potential for **[[** **]]** excessive sensitivity to timestep and nodalization variations, as discussed further in Section 3.5.

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's statement in ANP-3905, Appendix A, in the disposition of this limitation and condition that this **[[**
]]

Limitation and Condition 22

> The NRC staff has not specifically reviewed any plant parameters in ANP-10332P or deemed them acceptable for use in plant safety analyses. Therefore, each licensee using the AURORA-B LOCA evaluation model is responsible for confirming that all plant-specific design parameters are consistent with the assumptions made in the analysis. This includes, for example,
> **[[**
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's statement in ANP-3905, Appendix A, in the disposition of this limitation and condition, that the licensee
[[
]]

Limitation and Condition 23

> Safety analyses performed with the AURORA-B LOCA evaluation model shall include justification that **[[**
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's statement in ANP-3905, Appendix A, in the disposition of this limitation and condition, that **[[**
]]

Limitation and Condition 24

> **[[**

]]

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the licensee's statement in ANP-3905, Appendix A, in the disposition of this limitation and condition, that **[[**

]]

Limitation and Condition 25

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall justify the acceptability of the following evaluation model changes Framatome implemented during the NRC staff's review of ANP-10332P:
>
> - **[[**
> -
> -
> - **]]**
>
> (Section 3.6.3)

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the following justification provided by the licensee in ANP-3905, Appendix A, in the disposition of this limitation and condition, for the changes in the AURORA-B LOCA EM:

- The **[[**
-
-
-

]]

Limitation and Condition 26

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall confirm that **[[**

]]

Evaluation

The licensee stated that the **[[**

]] The NRC staff finds the licensee's statement is acceptable and the limitation and condition is satisfied because the **[[**

]]

Limitation and Condition 27

> As discussed in Section 4.3 of this SE [for ANP-10332P-A], new or modified Framatome **[[**
>
> **]]**

Evaluation

The NRC staff finds the limitation and condition is satisfied because the analysis **[[**
]]

3.8.7 Technical Conclusions

The NRC staff reviewed the information in ANP-3905 and in responses to the RAIs in the letter dated July 18, 2022 (Reference 3) and concludes that the LOCA analysis with ATRIUM 11 fuel is acceptable because it complies with the relevant requirements of 10 CFR 50.46, 10 CFR Part 50 Appendix K, and GDC 35. This conclusion is based on the following:

- The licensee analyzed the performance of the ECCS with ATRIUM 11 fuel in accordance with 10 CFR 50.46.

- The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model that follows Appendix K to 10 CFR Part 50 and meets the requirements of 10 CFR 50.46.

- The results of the analyses show that the 10 CFR 50.46(b)(1) through (b)(3) acceptance criteria for PCT ≤ 2200ºF, MLO ≤ 17 percent, and CWO ≤ 1 percent, respectively, are satisfied.

- Compliance with 10 CFR 50.46(b)(1), (b)(2), and (b)(3) criteria plus the licensee's analysis on seismic and LOCA load combination for the fuel assembly ensures that the 10 CFR 50.46(b)(4) acceptance criterion on maintaining a coolable geometry is satisfied.

- The 10 CFR 50.46(b)(5) acceptance criterion on long-term coolability is satisfied because the conclusions in NEDO-20566A, Section III, "General Electric Boiling Water Reactor Conformance to 10 CFR 50.46 Acceptance Criteria," on long-term cooling are applicable to the ATRUM 11 core.

- Having shown compliance with applicable acceptance criteria, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10 CFR 50.46 acceptance criteria, given operation at or below the MAPLHGR limits specified in the COLR. The applicability of the LOCA analysis is confirmed on a cycle specific basis.

- Applying the NRC staff-approved LOCA EM and methodology for the LOCA analysis with ATRIUM 11 fuel and adequately meeting the limitations and conditions listed in the NRC staff's SE for the applied TRs (section 3.8.6 of this SE).

- The evaluation meets the requirements of GDC 35 by demonstrating with the LOCA analysis performed that abundant emergency core cooling is provided to transfer heat from the reactor core filled with ATRIUM 11 fuel in the event of a LOCA and showing that suitable redundancy of components and features is provided so that the safety function can be accomplished assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources (sections 3.8.4 and 3.8.5 of this SE).

- The break spectrum analysis considered a spectrum of postulated DEG and split breaks in the recirculation system suction and discharge piping which generally conforms to the approved EM documented in ANP-10332P-A.

- Consistent with ANP-10332P-A, break spectra were calculated for both mid- and top-peaked axial power shapes at the time of maximum fuel stored energy (i.e., near the beginning of the operating cycle).

- Considering Browns Ferry is licensed to the MELLLA+ domain, sufficient initial state points were considered in the break spectrum analysis to provide confidence that the most limiting conditions have been analyzed. The NRC staff finds the selected analysis state points acceptable because the licensee followed the appropriate regulatory guidance with respect to analyzing the MELLLA+ operating domain.

- The LOCA break spectrum analysis based on a future equilibrium cycle of ATRIUM 11 fuel would bound transition cycles containing some co-resident legacy fuel bundles of the ATRIUM 10XM design.

- The licensee provided adequate qualitative evidence that the impacts of transition cycles containing co-resident ATRIUM 10XM fuel on the LOCA evaluation would be small and within the conservative bounds established by the existing analysis so that the evaluation results meet the required design criteria.

- The thermal-hydraulic compatibility analysis demonstrates that the thermal-hydraulic characteristics of the ATRIUM 11 and the coexistent ATRIUM 10XM fuel **[[**

  ]]

- The licensee satisfactorily addressed the NRC staff concern regarding a partial LPCI flow diversion to the suppression pool during a postulated LOCA occurring when the RHR system is in flow test mode or suppression pool cooling/mixing mode during normal operation.

- LOCA analysis **[[**

  ]]

- The LOCA break spectrum analysis results demonstrate the acceptability of the ATRIUM 11 MAPLHGR limit shown in ANP-3905, Figure 2.1.

- Operating below the MAPLHGR will ensure that the 10 CFR 50.46(b)(1), (b)(2), and (b)(3) acceptance criteria are met.

3.9 BEO-III (ANP-3907)

The BEO-III methodology was approved by the NRC staff. Browns Ferry introduced a change in the calculation procedure. Browns Ferry does not rely on the period-based detection algorithm (PBDA) as the detect and suppress algorithm. Instead, Browns Ferry maintains their licensing basis CDA.

3.9.1 Regulatory Evaluation

The plant-specific BEO-III LTSS and related licensing basis were developed to comply with the requirements of GDC 10 and 12 in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

Criterion 10, "Reactor design," requires that, "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences."

Criterion 12, "Suppression of reactor power oscillations," requires that, "The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed."

Consistent with GDC 10 and GDC 12 from Appendix A to 10 CFR Part 50, the NRC staff will confirm that the licensee performs the plant-specific trip setpoint calculations for long-term stability using acceptable methodologies as prescribed in the SRP (NUREG-0800) chapters 4.4 and 15.9.

3.9.2 Technical Evaluation

Browns Ferry used a plant-specific BEO-III methodology that maintains the current Browns Ferry licensing basis CDA. The method establishes the cycle-specific OLMCPR based on statistical analyses of pump trip scenarios and evaluation of the time dependent local power range monitors (LPRMs) and core MCPR to determine the most limiting event based on the Browns Ferry CDA detect and suppress (D&S) hardware response.

The licensee's plant-specific implementation requests the use of the CDA as the licensing basis stability algorithm and the PBDA as defense-in-depth in its oscillating power range monitoring (OPRM) system. The calculation procedure remains unchanged in the BEO-III methodology with the exception of using the CDA trip times.

3.9.2.1 BEO-III Evaluation Model Revision

In order to support the plant specific methodology for BEO-III, the CDA algorithm adds post processing steps **[[**

]] RAMONA5-FA was modified to write both a time-dependent instrument data file and a file that contains the time dependent core MCPR **[[**

]]

The Browns Ferry application embeds the CDA post-processing calculations within the automated BEO-III statistical evaluations, allowing the Browns Ferry application to post-process all statistical trials with the CDA rather than just select statistical trials. The post-processing executable (CDAST) was developed with a contract between GE Hitachi (GEH) Nuclear Energy and TVA. The post-processor executable reads the time dependent power, flow and simulated LPRM signals from the instrument data file written by RAMONA5-FA.

<u>CDAST Installation</u>

The licensee confirmed that the post-processor, CDAST, was installed correctly by comparing results provided by GEH. The licensee confirmed that the installation was successful.

<u>CDAST Verification</u>

Verification that CDAST would run successfully on Framatome computers was completed by running a test suite. The tests cases were generated by Framatome by using the RAMONA5-FA code and transmitted to GEH for use in the CDAST code development. This ensured that the results were consistent. GEH ran CDAST for each test case and defined the reference CDA scram initiation signal time based on the Browns Ferry CDA and plant specific OPRM and RPS hardware delays. Execution of the CDAST code on the Framatome network reproduced the GEH results for all test cases exactly.

CDAST Automation within BEO-III Analysis

The CDAST calculations have been integrated into the BEO-III statistical analysis as a post-processor. The automation supporting the CDAST calculations navigate to each of the RAMONA5-FA directories and run the CDAST code to determine the CDA response.

Oscillations Suppressed by Scram

If the original RAMONA5-FA calculations **[[**

]] and an MCPR FoM based on the CDA suppression time and associated independent channel oscillations (ICO) FoM have been successfully computed.

If the **[[**

]]

Oscillations Not Suppressed by Scram

Oscillations not suppressed by scram are the result of RAMONA5-FA calculations which result in stable terminal conditions **[[**

]] ICO could invalidate Stage I MCPR results, the automation **[[**

]] If the trial indicates that the Stage I MCPR results are invalid due to potential ICO, and **[[**

]]

If the CDA does not result in a scram, **[[**

]]

BEO-III Calculation Procedure

The Browns Ferry BEO-III methodology is performed on a cycle-specific basis to confirm that any pump trip scenarios which lead to instability are non-limiting when compared to the events which determine the OLMCPR. The calculation methodology for Browns Ferry is the same as the methodology described in the approved TR, ANP-10344P-A (Reference 28) with the exception of using the CDA algorithm instead of the PBDA algorithm. **[[**

]]

The application procedure includes the following elements:

1. Definition of the state points to be analyzed.
2. The RAMONA5-FA base deck for the statistical analyses is prepared as a best estimate model **[[**

   ]]
3. Once the exposure points are identified **[[**

   ]]
4. For each of the selected exposure steps, an ensemble of RAMONA5-FA multi-stage analyses are run (in accordance with section 7.1.6 of ANP-10344P-A [Reference 28]). For each successful calculation **[[**

   ]] RAMONA5-FA also writes the instrumentation data file read by the CDAST post-processor.
5. For each of the RAMONA5-FA statistical trials, the CDAST calculation determines the CDA scram initiation signal time. **[[**

   ]] suppression time.
6. For each exposure step, the 95/95 simultaneous tolerance limit values associated with the coupled reactor response and the limiting fuel assembly response are determined by the FIND9595 post-processing software.
7. **[[**

   ]]
8. Review the **[[**
   ]] If it is determined that the pump coastdown behavior would have a significant impact on the final MCPR, the uncertainties in the recirculation pump coastdown response should be included in the statistical analyses or otherwise accounted for.

3.9.3 Backup Stability Protection

The preferred method is the automated backup stability protection (ABSP) scram region since it requires no operator action to protect against fast developing oscillations following a 2RPT from the MELLLA+ operating domain. If this feature is unavailable, manual BSP exclusion regions on the power/flow map will be employed and operation will be restricted to the MELLLA operating domain. This is enforced in the Browns Ferry Technical Specifications which requires operation to be lowered below the BSP Boundary (MELLLA Boundary) as specified in the cycle-specific COLR in the event the ABSP is not available.

The calculation process for the Manual BSP regions will be identical to that currently used at Browns Ferry to support MELLLA+.

3.9.3.1 Automate BSP Scram Region

The NUMAC APRM/OPRM hardware installed at Browns Ferry to support MELLLA+ implementation contains a feature to produce an automatic scram if potentially unstable regions of the power flow map are entered. This is done with adjustments to the ABSP flow-biased scram setpoints to encompass the nominal feedwater temperature manual scram region determined on a cycle specific basis. Due to the automatic scram upon entry into this region, operation in MELLLA+ is allowed since the automatic scram does not rely on operator actions in the event of an inadvertent flow reduction event that challenges the stability of the reactor core. The ABSP does not need setpoint adjustment for reduced feedwater temperature (RFWT) because ABSP protects against high growth rate oscillations resulting from a 2RPT from the MELLLA+ region which does not allow RFWT. There is no change to the process of determining ABSP scram setpoints with the introduction of BEO-III methodology.

3.9.4 RAMONA5-FA Qualification for Browns Ferry

Browns Ferry collected OPRM data during the MELLLA+ implementation which they used to compare to the RAMONA5-FA statistical calculation of the OPRM amplitudes and maximum amplitude. The data was taken from all three Browns Ferry units during the period of February 8, 2020, through May 13, 2020. The testing started at 100 percent power / 85 percent flow and followed the rod line to the minimum core flow for the MELLLA+ domain. **[[**

]]

3.9.5 ATRIUM 11 Equilibrium Cycle Sample Application

The licensee completed an equilibrium cycle sample application of the Browns Ferry-specific BEO-III methodology which demonstrated that the CDA hardware installed can detect and suppress oscillations with a high confidence level for the ATRIUM 11 fuel design. The licensee assessed the full power cycle exposures in the equilibrium cycle design depletion.

Browns Ferry used sample parameters consistent with those defined in ANP-10344P-A for the statistical analysis. Per ANP-10344P-A, there are **[[**

]]

The RAMONA5-FA simulation provides the time dependent LPRM and core MCPR as a function of time to allow the determination of the MCPR response for every statistical trial. To minimize the probability of **[[**

]]

3.9.5.1 Limiting MELLLA+ 2RPT Scenario

The limiting MELLLA+ 2RPT scenario analyzed was at 100 percent power and 85 percent rated core flow. Browns Ferry used a sample size of **[[ ]]** trials, consistent with the values in ANP-10334P-A. The analysis showed that the 95/95 MCPR value is well above the SLMCPR value.

The **[[**

]]
Therefore, ICO do not invalidate the assumption that the reactor protection system can detect and suppress the oscillations prior to violation of the specified acceptable fuel design limits.

3.9.5.2 Evaluation of Pump Coastdown

The licensee assessed the impact of pump coastdown on the FoM by comparing the trip timings and the pump coastdown time. The licensee concluded **[[**

]]

3.9.5.3 Potentially Limiting Scenarios

There are two other potentially limiting scenarios which the licensee analyzed. The scenarios are lowest core flow at rated core power within the MELLLA domain with the minimum feedwater temperature allowed by the licensee for the feedwater heater out-of-service or final feedwater temperature reduction scenario and the highest core power under SLO. They used the same statistical trials consistent with ANP-10344P-A.

The limiting single pump trip under SLO was analyzed at 67.95 percent of rated power and 54.4 percent of rated flow. Based on a conservatively assumed **[[ ]]** power-dependent OLMCPR, the core MCPR FoM of **[[**

]]

3.9.5.4 T_{min} Confirmation

The licensee stated that the Browns Ferry plant-specific D&S hardware specifies a minimum oscillation period of **[[ ]]** seconds. The MELLLA+ 2RPT scenario yielded a minimum period, T_{min}, of **[[ ]]** seconds. The 1PT from the MELLLA+ corner produced stable results. A conservative analysis was used for the sample application to minimize the iterative process to identify the initial condition that would result in credible oscillations.

The statistical analysis for the **[[ ]]** statepoint was repeated **[[**

]] The 95/95 minimum period was **[[ ]]** seconds. Based on these analyses, the T_{min} value of **[[ ]]** seconds is confirmed to bound the minimum oscillation period for any credible oscillations.

3.9.6 <u>Compliance with Limitations and Conditions</u>

The licensee discussed how they meet the limitations and conditions of ANP-10344P-A.

<u>Limitation and Condition 1</u>

> MICROBURN-B2 is an integral component in the BEO-III methodology. Application of a new core simulator requires review and approval by the NRC.

<u>Evaluation</u>

MICROBURN-B2 is the specified core simulator for the Browns Ferry Application Methodology. The NRC staff finds the limitation and condition is satisfied.

<u>Limitation and Condition 2</u>

> Selected settings and modeling options, including core and vessel nodalization and time step control parameters, shall be defined consistently with the validation basis presented in Section 6.0.

<u>Evaluation</u>

All settings and modeling options, including core and vessel nodalization and time step control parameters are consistent with the validation basis. The NRC staff finds the limitation and condition is satisfied.

<u>Limitation and Condition 3</u>

> **[[**
>
> **]]**

<u>Evaluation</u>

[[

]] The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 4

> **[[**
>
> **]]**

Evaluation

The **[[** **]]** for protecting the SAFDL. The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 5

> Framatome must continue to use existing regulatory processes for any code modifications made to the RAMONA-5 FA code. The existing regulatory processes do not allow changes to the RAMONA5 -FA code that would substantively alter the BEO-III methodology, as described in ANP-10344P and supporting RAI responses, which the NRC staff relied upon as the basis for the finding of acceptability in this SE, without prior NRC review and approval.

Evaluation

The RAMONA5-FA code was modified for the Browns Ferry Application Methodology as described in section 2.0 of ANP-3907 (Attachment 13a to Reference 1) and was evaluated as per existing regulatory processes. There are no changes beyond those in section 2.0 that would alter the basis of the methodology. The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 6

> Plant-specific applications shall justify whether the recirculation pump coastdown behavior will have a significant impact on the final MCPR for the specific plant and conditions being analyzed. If so, the uncertainties in the recirculation pump coastdown response should be included in the statistical analyses or otherwise accounted for.

Evaluation

The concern associated with this limitation and condition is that variations in the pump coastdown rate could lead to system resets that would cause the reactor trip to be delayed and allow the amplitude to grow to larger magnitudes. The assessment of **[[** **]]** on the final MCPR for the ATRIUM 11 equilibrium cycle. The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 7

> If the 1RPT EFW [extended flow window] event remains stable, additional analyses are required using **[[** **]]** to

ensure that the lowest oscillation period remains above T_{min} under any anticipated conditions.

Evaluation

The 1RPT from the minimum core flow at rated power in the MELLLA+ domain remained stable. **[[**

]] The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 8

After applying the **[[**

]] If trends are observed which indicate that the most limiting exposure point(s) may be outside the analyzed range of exposures, additional exposure points should be analyzed until reasonable assurance is attained that the limiting exposure point is analyzed.

Evaluation

The **[[** **]]** Review of the FoM trend as a function of exposure indicates that there is a reasonable assurance that the limiting exposure points were analyzed. The NRC staff finds the limitation and condition is satisfied.

3.9.7 TECHNICAL CONCLUSION

The NRC staff has reviewed the responses to the limitations and conditions and the supporting information in the relevant sections of ANP-3907 and find the licensee has acceptably addressed the limitations and conditions of ANP-10344P-A. Therefore, the NRC staff finds the methodology acceptable to use with Browns Ferry's plant-specific application.

3.10 ATWS-I (ANP-3906)

Browns Ferry used the RAMONA5-FA code to simulate ATWS-I events. The two scenarios that have been identified as potentially limiting are the turbine trip with bypass and the two-recirculation pump trip.

3.10.1 Regulatory Evaluation

Including GDC described in Section 2.2 of this SE, the following regulatory requirements apply to the ATWS I evaluation:

Section 50.62, "Requirements for reduction of risk from anticipate transients without scram (ATWS) events for light-water-cooled nuclear power plants," of 10 CFR requires that the licensee provide an acceptable reduction of risk from ATWS events by inclusion of prescribed design features and demonstrating their adequacy in mitigation of the consequences of an ATWS event. Within the context of review of the submittal, the ATWS-I analyses are intended to demonstrate that the combination of automated plant functions and prescribed operator actions will be sufficient to preclude fuel failure.

Section 50.46, "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors," of 10 CFR is not directly applicable to the ATWS-I event because it is intended to address postulated LOCAs rather than ATWS events. However, this regulation does present a set of acceptance criteria for ensuring adequate cooling of fuel such that significant fuel failures do not occur.

The SRP (NUREG-0800) is the primary regulatory guidance document used by the NRC staff to support review of this LAR. In particular, SRP chapter 15.8, "Anticipated Transients Without Scram" (Reference 55), establishes acceptance criteria for ATWS events. Chapter 15.8 does include additional GDC beyond those listed above; however, they define vessel, ECCS, and containment performance requirements. These GDC were not considered as part of this review because these items are not of any significance for the evaluation of ATWS-I event related to this amendment review.

3.10.2 Technical Evaluation

The ATWS-I transients are simulated using the RAMONA5-FA computer code. This code was approved by the NRC in October 2019 (Reference 56). The scope of the analysis covers the Browns Ferry MELLLA+ operating domain with an equilibrium ATRIUM 11 core. The turbine trip with bypass and two recirculation pump trips are evaluated.

3.10.2.1 RAMONA5-FA ATWS-I Methodology

ATWS-I calculations completed using RAMONA5-FA are to be in compliance with the calculation procedure defined in the approved TR for RAMONA5-FA. The licensee stated that all calculations are in compliance with the conditions of the TR. RAMONA5-FA is used to evaluate the fuel specific portion of the event which confirms the limiting PCT is below 2200 degrees Fahrenheit.

3.10.2.2 Critical Power Reduced Order Model Correlation

The RAMONA5-FA transient model uses the critical power reduced order model correlation for dryout. This correlation was presented in the RAMONA5-FA TR (Reference 56). The advantage of this correlation is that it is well-suited for transient models that include cyclical dryout and rewet with possible failure to rewet.

3.10.2.3 Turbine Trip With Bypass

The turbine trip with bypass scenario can lead to an ATWS-I situation because the feedwater temperature cools rapidly as the core settles at natural circulation due to the loss of extraction steam when turbine trips and the recirculation pumps trip on high pressure. The scenario is terminated with operator action to reduce water level.

The analysis for the ATRIUM 11 fuel transition was completed with the same input as the ATWS-I analysis of record. The feedwater temperature decrease is assumed to start at 14 seconds after the valve closure and the temperature is assumed to decrease at a rate of 1.1 degrees Fahrenheit per second for the first minute and then 1.4 degrees Fahrenheit per second until the final temperature is reached. The time critical operator action for water level reduction is assumed to begin at 90 seconds. Sensitivity calculations were completed at [[ ]] The worst case scenario for peak cladding temperature occurs at [[ ]] The PCT remains below the limit in all calculated scenarios.

3.10.2.4 Two Recirculation Pump Trip

During the two recirculation pump trip scenario, the turbine remains online and the extraction steam to the feedwater heaters is maintained. The feedwater temperature remains higher than during the turbine trip with bypass scenario. As a result, the power excursion in the two recirculation pump trip event is a less severe event when compared to the turbine trip with bypass event for the same operator intervention times. The two recirculation pump trip does not automatically signal a scram at event initiation. One scenario when the two recirculation pump trip can be more severe than the turbine trip with bypass is one where operator action is delayed and the ATWS unidentified.

The base statepoints analyzed for the two recirculation pump trip event were consistent with those chosen and analyzed for the turbine trip with bypass events. The results show [[ ]]

3.10.2.5 Compliance with Limitations and Conditions

Limitation and Condition 1

> The gap conductance sensitivity shall be repeated or otherwise justified for transitions to new fuel designs.

Evaluation

The licensee completed the gap conductance sensitivity study for the limiting event, [[ ]] The overall impact was shown to be within the available margin for ATRIUM 11 fuel with a gap conductance variation of [[ ]] The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 2

> If the acceptance criteria for the first paragraph in Step 3 of Section 8.0 of the TR are met, additional justification must still be provided to demonstrate adequate margin in operator action timing for variations in neutron kinetics response from specific core designs. This justification may be provided by following Steps 3.a through 3.c, as amended by the response to RAI 15, or providing an alternative justification on a plant-specific basis.

Evaluation

The licensee took steps to ensure the calculation will bound future core designs**. [[**

]] The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 3

> Plant-specific evaluations that are intended to be bounding of all core designs must be confirmed to provide reasonable assurance that neutron kinetics characteristics such as possible differences in dominant oscillation modes or the potential for multiple oscillation modes to be active simultaneously are bounded by the analysis of record.

Evaluation

As discussed in the evaluation for Limitation and Condition 2, a bounding analysis was completed. The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 4

> Due to the unique neutron kinetics characteristics associated with transition cycles, all transition cycles must be dispositioned in a manner consistent with Limitations and Conditions #2 and #3.

Evaluation

As discussed in Limitation and Condition 2, a bounding analysis was completed. The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 5

> The ATWS-I analysis must be performed for both the TTWB and 2RPT events during the initial implementation of this methodology, to confirm which event is limiting. Subsequent evaluations may only consider the event determined to be limiting, except when changes are made to the plant design or operation that may affect stability behavior during ATWS, such as: turbine bypass capability, fraction of steam-driven feedwater pumps, and changes expected to significantly increase core inlet subcooling during ATWS events.

Evaluation

Analyses were performed for both events and the licensee determined **[[**
]] The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 6

> The steam line and valve modeling options shall be confirmed to accurately capture the expected plant-specific system performance during ATWS-I events.

Evaluation

The steam line and valve models were completed using plant geometry and setpoints, allowing for an accurate representation. The NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 7

> Plant-specific applications must justify that the selected settings and modeling options are appropriate, including core and vessel nodalization, time step control parameters, and noise parameters. In particular, the modeling should be reasonably consistent with both the characteristics of the plant in question and the validation basis for the RAMONA5-FA ATWS-I methodology as discussed in this SE.

Evaluation

The nodalization was consistent with the nodalization of the benchmarks and sample problems in the original TR. A study was performed to demonstrate the nodalization reasonably represents the plant. The NRC staff finds the limitation and condition is satisfied.

3.10.3 Technical Conclusion

The licensee evaluated the ATWS-I event following all limitations and conditions and is, therefore, acceptable.

3.11 CONTROL ROD DROP ACCIDENT (ANP-3874)

3.11.1 Regulatory Evaluation

GDC 13 and 28 are the applicable criteria (see section 2.2 of this SE) along with 10 CFR 50.67 for the evaluation of CRDA events. GDC 13 addresses the provision of instrumentation to monitor systems and variables that can affect the fission process and controls to keep these systems and variables within prescribed operating ranges. General Design Criterion 13 primarily applies to the CRDA event by ensuring that the limiting system operating parameters and other controls in place (i.e., rod withdrawal limitations) are sufficient to ensure that the CRDA acceptance criteria are not exceeded during the CRDA event. This is satisfied by ensuring that the initial conditions represented in the CRDA analyses are sufficiently representative of the most conservative conditions allowed by the aforementioned controls. Application of GDC 28 requires that postulated reactivity accidents neither impart sufficient damage to impair core cooling capability nor damage the reactor coolant pressure boundary greater than limited local yielding. General Design Criterion 28 requires that these postulated reactivity accidents include rod dropout. In addition, Browns Ferry is licensed under 10 CFR 50.67 to establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.

Regulatory Guide 1.236 describes methods and procedures that the NRC staff considers acceptable when analyzing a postulated control rod drop accident (Reference 16). It is the approved version of DG-1327 (Reference 17). The licensee used the RG 1.236 regulatory guidance criteria for the CRDA event. The guidance criteria presented in RG 1.236 supersedes the acceptance criteria presented in SRP section 4.2 and DG-1327.

Regulatory Guide 1.236 also references RG 1.183 (Reference 18) and RG 1.195 (Reference 19) for evaluation of accident dose radiological consequence criteria.

The licensee used the methodology in ANP-10333P-A, Revision 0 (Reference 27), to evaluate CRDA. The NRC staff's SE for ANP-10333P-A (see Reference 27) was issued prior to the publication of RG 1.236. Section 3.0 of that SE notes that the NRC staff considered the applicability of the TR methodology to DG-1327. It states that prior to the use of the TR methodology with the final approved reactivity initiated accident acceptance criteria, any changes to the draft guidance must be evaluated to verify that there have been no changes beyond clarifications or editorial changes consistent with the discussion of accident acceptance criteria in that SE or adjustments to the numeric thresholds for specified limits that do not go outside the bounds of the values used to validate the methodology and uncertainties discussed in the TR. Section 4.2.2.2.4 of that SE (see Reference 27) discusses the basis for acceptability of the TR methodology in satisfying DG-1327 acceptance criteria.

The NRC staff modified DG-1327 in response to public comments. As discussed in section 3.11.2 of this SE, some DG-1327 guidance related to the evaluation of radiological consequences was removed from RG 1.236 to be published in other RGs. Threshold curves for evaluating pellet-clad mechanical interaction (PCMI) failure were also modified in RG 1.236. Additionally, clarification on the appropriate quantity to be used in the evaluation of PCMI failure threshold curves was provided in RG 1.236. These items are discussed in further detail in section 3.11.2 of this SE.

3.11.2 Technical Evaluation

The applicability of AURORA-B CRDA methodology to ATRIUM 11 fuel is discussed in section 6.4 of ANP-3908P, Revision 4 (Attachment 11 to Reference 6). A summary of the application of the AURORA-B CRDA methodology (Reference 27) to the Browns Ferry ATRIUM 11 equilibrium cycle is provided in ANP-3874P, Revision 3 (Attachment 15 to Reference 6). This demonstrates how the Framatome CRDA analysis methodology will be applied at Browns Ferry each cycle. The sample calculations were based on the equilibrium core design, but cycle-specific calculations will be performed to support each reload. The methodology presented in ANP-3874P includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic and fuel temperature feedback to show compliance with regulatory guidance criteria as presented in RG 1.236. The methodology demonstration involves sensitivity studies and determination of an evaluation boundary for ATRIUM 11 fuel, with similar process to be followed for the transition cores.

The NRC staff compared the information provided by the licensee against ANP-10333P-A and finds that the licensee demonstrated an acceptable application of the methodology to evaluate the CRDA event for the Browns Ferry equilibrium core design. Section 8.0, "Limitations and Conditions," of ANP-3874P contains a discussion of limitations and conditions for ANP-10333P-A. The NRC staff reviewed this information and finds that all of the limitations and conditions for ANP-10333P-A were met.

The NRC staff makes the following additional findings and observations specific to Browns Ferry:

- Appendix B to ANP-3874P provides information on the CHF correlation used for the CRDA calculations. **[[**

- **]]** Instead, the licensee used the **[[** **]]** CHF correlation, which is **[[** **]]** The **[[** **]]** CHF correlation was also **[[** **]]** Therefore, the NRC staff finds the **[[** **]]** CHF correlation to be acceptable for use for this purpose.

- Regulatory Position 2.3.4.2 of RG 1.236 states that total hydrogen content (including hydrogen present in the oxide layer) should be used to implement threshold curves for evaluation of pellet-cladding mechanical interaction failure, because these curves were developed using total hydrogen content. **[[** **]]** Because of this, the NRC staff finds **[[** **]]** obtained with the methodology described in ANP-10333P-A for implementation of RG 1.236 threshold curves for evaluating pellet-cladding mechanical interaction failure acceptable.

- To assess the contribution of transient fission gas release to radiological consequences of the postulated CRDA, the licensee used the transient fission gas release model from DG-1327. This model included a correlation for transient fission gas release and weighting-factors for different radionuclide groups. Based on public comments on DG-1327, the NRC staff elected to move transient fission gas release models to future revisions of RG 1.183 and RG 1.195 but retained the DG-1327 transient fission gas release correlation in Appendix B of RG 1.236 until RG 1.183 and RG 1.195 can be updated. Isotope-group weighting factors were not reproduced in Appendix B of RG 1.236 but were included in DG-1389 (Reference 57), which was released for public comment at the time of this writing. Because the isotope-group weighting factors are conservative with respect to draft guidance (DG-1389) (Reference 57), the NRC staff finds their use acceptable.

- Appendix A of ANP-3874P describes the process used to establish an evaluation boundary curve to simplify the calculations. This process was approved as part of the ANP-10333P-A methodology, with Limitation and Condition 31 requiring the licensee to confirm the applicability of the curve to several local characteristics that may be present in the core being analyzed. The licensee presented this information for an equilibrium core which contains only ATRIUM 11 fuel. The licensee will confirm that the evaluation boundary curve is also applicable to ATRIUM 10XM fuel prior to use as part of the reload analysis for the transition cores.

- The NRC revised DG-1327 curves for evaluating PCMI failure in RG 1.236. Section 4.2 of ANP-3874 states that Browns Ferry will use RG 1.236 curves when evaluating PCMI failure, and RG 1.236 curves are depicted in the sample analysis. The NRC staff finds this acceptable because it is consistent with NRC guidance.

For CRDA analysis, the NRC staff confirmed that: the licensee applied NRC-approved analytical methods to perform a demonstration CRDA analysis; derived the acceptance criteria from the TR approved for CRDA analysis; showed how it would determine whether fuel failures would occur; considered a scenario where fuel failures occur so they could show how the radiological consequences would be evaluated; performed calculations and evaluations in a manner consistent with the bases for the NRC staff approval of the methods; and demonstrated acceptance criteria are met. The NRC staff also evaluated the analysis methodology with respect to differences between DG-1327 and RG 1.236 and finds that the licensee's implementation of ANP-10333P-A, Revision 0 is consistent with current NRC staff guidance in RG 1.236 and does not invalidate the basis for approval of ANP-10333P-A, Revision 0. Based on the above evaluation, the NRC staff finds that the proposed adoption of the CRDA analysis methods as part of the transition to ATRIUM 11 fuel is acceptable.

3.11.3 Technical Conclusion

The NRC staff reviewed the information in the licensee's submittal pertaining to the analysis of the CRDA event for Browns Ferry, Units 1, 2, and 3. Based upon its review, as discussed above, the NRC staff finds that the licensee has proposed to implement the CRDA analysis methodology using the AURORA-B evaluation model in an acceptable manner, satisfying all limitations and conditions, and compliance with the applicable regulatory requirements has been demonstrated.

3.12 TRANSIENT DEMONSTRATION (ANP-3904)

3.12.1 Applicable Regulatory Requirements

In addition to the GDC listed in section 2.2 of this SE, the following regulatory requirement applies to the AOO/ATWS evaluation:

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which requires licensees to provide the means to address an ATWS, which for the purposes of 10 CFR 50.62 means an AOO as defined in Appendix A to 10 CFR Part 50 followed by the failure of the reactor trip portion of the protection system specified in GDC 20.

3.12.2 Anticipated Operational Occurrences

The NRC staff reviewed the following items submitted by the licensee in ANP-3904, Revision 5 (Attachment 16a to Reference 6):

- Demonstration of the applicability of the AURORA-B AOO methodology ANP-10300P-A (Reference 33) for Browns Ferry.

- Implementation of AURORA-B AOO methodology to Browns Ferry UFSAR chapter 14 events.

- Analyses of select licensing basis AOOs using the AURORA-B methodology to demonstrate that the results meet the applicable acceptance criteria.

- Compliance with the NRC imposed limitations and conditions for the application of the AURORA-B methodology.

3.12.3 Applicability of AURORA-B AOO Methodology

The AURORA-B AOO methodology and the NRC staff's SE of the methodology is documented in ANP-10300P-A. The AURORA-B AOO EM predicts the BWR response to transient and postulated accidents. The methodology is built upon three computers codes:

- S-RELAP5 – This code provides the transient thermal-hydraulic, thermal conduction, control systems, and special process capabilities (i.e., valves, jet pumps, steam separator, critical power correlations, etc.) necessary to simulate a BWR plant.

- MB2-K – This code provides the core neutronic response.

- RODEX4 – This code provides the thermal-mechanical response of the individual fuel rods. A subset of routines from this code are used to evaluate the transient thermal-mechanical fuel rod (including fuel/clad gap) properties as a function of temperature and rod internal pressure. The fuel rod properties are used by S-RELAP5 when solving the transient thermal conduction equations in lieu of standard S-RELAP5 material property tables.

The above three codes combined together provides a model to predict the results of AOOs. The methodology is based on **[[**
]] In the AURORA-B AOO methodology, the uncertainty analysis is used to bound the 95-percent worst case result at 95-percent confidence. Table 3.6 of the SE for the AURORA-B AOO methodology contains the parameters used for the uncertainty analysis.
[[
]] Deterministic analysis using conservative initial and boundary conditions is performed for operational transients and TS requirements.

The NRC staff reviewed the AURORA-B AOO methodology to ensure that it is applicable for the analysis of AOOs for Browns Ferry. As described in section 3.1 (Applicability of Framatome BWR Methods to Browns Ferry with ATRIUM 11 Fuel) of the SE for the AURORA-B AOO ANP-10300P-A, the methodology is applicable, in part, to BWR/3 through BWR/6 plants with conditions extending to EPU with EFW or MELLLA+ operating domain. Since the Browns Ferry units are BWR/4 plants, the methodology is applicable to Browns Ferry in this respect. In addition, as discussed in section 4.3 of the SE for ANP-10332P-A, Revision 0 (Reference 50), and in limitation and condition number 27 of the SE, ATRIUM 11 is identified as one of the existing fuel designs that was considered in the review. Therefore, the NRC staff finds that the AURORA-B AOO methodology is applicable to the ATRIUM 11 fuel design for Browns Ferry.

3.12.4 Implementation of AURORA-B AOO Methodology to Browns Ferry UFSAR Chapter 14 Events

Table 2 below provides the licensee's disposition of UFSAR chapter 14 events and accidents of whether the event or accident should be analyzed at the initial reload of ATRIUM 11 fuel only, or at each reload cycle using the AURORA-B AOO methodology, or no further analysis of the event or accident is necessary. The licensee provided reasons for each disposition in ANP-3904, Table 3.1. The NRC staff's evaluation of the licensee's dispositions is included in Table 2 below.

Table 2 - NRC Staff's Evaluation of Licensee's Disposition of UFSAR Chapter 14 Events

UFSAR Section	Event/Analysis	Licensee's Disposition	NRC Staff's Evaluation
14.5.2.1	Generator trip (TCV fast closure)	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the event is bounded by the generator trip with turbine bypass failure.
14.5.2.2	Generator trip (TCV fast closure) with turbine bypass valve failure	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable because of being an abnormal operational transient (AOT).
14.5.2.2.4	Load Rejection No Bypass (LRNB) with EOC-RPT-OOS	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable because of being an AOT.
14.5.2.3	Loss of condenser vacuum	No further analysis required	The NRC staff finds the licensee's disposition acceptable because this transient is equivalent to the turbine trip with bypass operable and is therefore bounded by the turbine trip with turbine bypass valve failure. The current UFSAR analysis therefore applies.
14.5.2.4	Turbine trip (TSV closure)	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the event is bounded by the turbine trip with turbine bypass valve failure.
14.5.2.5	Turbine bypass valves failure following turbine trip (TTNB), high power	Address for initial reload	In a letter dated July 18, 2022 (Reference 4), the licensee stated that this event will be addressed in the calculation plan mentioned in ANP-3904, section 3.2. The NRC staff finds the licensee's disposition acceptable because the calculation plan will identify the licensing campaign in which the analysis was performed and state that for the upcoming cycle, the analysis (a) is not needed if this event is determined to be non-limiting in its previous analysis, and (b) is needed if this event is determined to be limiting in its previous analysis.

UFSAR Section	Event/Analysis	Licensee's Disposition	NRC Staff's Evaluation
14.5.2.6	Turbine bypass valves failure following turbine trip (TTNB), low power	No further analysis required	In a letter dated July 18, 2022, the licensee stated that this event will be addressed in the calculation plan mentioned in ANP-3904, section 3.2. The NRC staff finds the licensee's disposition acceptable because the calculation plan will identify the licensing campaign in which the analysis was performed and state that for the upcoming cycle, the analysis (a) is not needed if this event is determined to be non-limiting in its previous analysis, and (b) is needed if this event is determined to be limiting in its previous analysis.
14.5.2.7	Main steam isolation valve (MSIV) closure	No further analysis required	In a letter dated July 18, 2022, the licensee stated that this event credits the scram signal on the MSIV position which greatly reduces its severity and is the basis for its disposition. The event is bounded by the LRNB with EOC-RPT-OOS of UFSAR section 14.5.2.2.4 with respect to thermal limit response. The NRC staff finds the licensee's disposition acceptable and the current UFSAR analysis therefore applies.
14.5.2.8	Pressure regulator failure (downscale)	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the licensee eliminated this as an AOT by the installation of a digital fault-tolerant main turbine electrohydraulic control system. The current UFSAR analysis therefore applies.
14.5.3.1	Loss of feedwater heater (LFWH)	Address each reload	The NRC staff finds the licensee's disposition acceptable because conservatively the licensee intends to analyze at each reload cycle to determine if it remains bounded by other events specifically by FWCF event.
14.5.3.2	Shutdown cooling (RHR) malfunction – decreasing temperature	No further analysis required	The NRC staff finds the licensee's disposition acceptable because it is does not depend on the fuel design.
14.5.3.3	Inadvertent HPCI pump start (IHPS)	Address for initial reload	The licensee stated that the IHPS event is similar to the limiting LFWH event and has a considerable CPR margin from the LFWH event. The LFWH event would be analyzed at every reload cycle. Based on this, the NRC finds the licensee disposition acceptable.
14.5.4.1	Continuous rod withdrawal during power range operation	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable because of being a potential AOT.

UFSAR Section	Event/Analysis	Licensee's Disposition	NRC Staff's Evaluation
14.5.4.2	Continuous rod withdrawal during reactor startup	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore applies.
14.5.4.3	Control rod removal error during refueling	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore applies.
14.5.4.4	Fuel assembly insertion error during refueling	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore applies.
	Mislocated or misoriented fuel assembly	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable.
14.5.5.1	Pressure regulator failure open (PRFO)	Address each reload	The NRC staff finds the licensee's disposition acceptable because this is a potentially limiting ATWS overpressurization event to be addressed at each reload cycle. However relative to AOT thermal operating limits, the PRFO event does not depend on the fuel design and therefore the UFSAR analysis remans applicable
14.5.5.2	Inadvertent opening of a MSRV (IORV)	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore remains applicable.
14.5.5.3	Loss of feedwater flow (LOFW)	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore remains applicable.
14.5.5.4	Loss of auxiliary power	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore remains applicable.
14.5.6.1	Recirculation flow control failure – decreasing flow	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are bounded by the recirculation pump seizure event.

UFSAR Section	Event/Analysis	Licensee's Disposition	NRC Staff's Evaluation
14.5.6.2	Trip of one recirculation pump	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are bounded by the TTNB event.
14.5.6.3	Trip of two recirculation pumps	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are bounded the TTNB event.
14.5.6.4	Recirculation pump seizure	Address initial reload	The NRC staff finds the licensee's disposition acceptable because the licensee will perform plant specific analysis at the initial reload to determine its impact on the thermal limits.
14.5.7.1	Recirculation flow control failure-increasing flow	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable because the consequences of the slow flow run-up event determine the flow-dependent MCPR and LHGR operating limits.
14.5.7.2	Startup of idle recirculation loop	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore remains applicable.
14.5.8.1	Feedwater controller failure (FWCF) – maximum demand	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable because of being a potential AOT.
14.5.8.2	Feedwater controller failure (FWCF) – maximum demand with EOC-RPT-OOS	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable because of being a potential AOT.
14.5.8.3	Feedwater controller failure (FWCF) – maximum demand with TBVOOS	Address each reload	The NRC staff finds the licensee's disposition is conservative and acceptable because of being a potential AOT.
14.5.9	Loss of habitability of the control room	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design. The current UFSAR analysis therefore remains applicable.

UFSAR Section	Event/Analysis	Licensee's Disposition	NRC Staff's Evaluation
14.6.2	Control rod drop accident (CRDA)	Address each reload	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are evaluated to confirm the acceptance criteria stated in ANP-3874, Revision 3, section 2.2 are satisfied.
14.6.3	Loss-of-coolant accident (LOCA)	Address initial reload	The NRC staff finds the licensee's disposition acceptable because the consequences of the LOCA are evaluated to determine appropriate fuel-specific MAPLHGR limits which are independent of cycle specific assembly designs. For the subsequent reloads, the licensee will perform checks of the limiting power history, gadolinia LHGR confirmation, and MAPLHGR.
14.6.4	Refueling accident	Address each reload	The NRC staff finds the licensee's disposition acceptable because the consequences of the refueling accident will be evaluated for each reload to confirm the acceptance criteria as described in UFSAR Section 14.6.4 is satisfied.
14.6.5	Main steam line break accident	No further analysis required	The NRC staff finds the licensee's disposition acceptable because the consequences of this event are independent of the fuel design and are far from limiting with respect to the 10 CFR 50.46 acceptance criteria. The current UFSAR analysis therefore remains applicable.
10.11	Fire protection systems	Address initial reload	The NRC staff finds the licensee's disposition acceptable because the licensee stated that the reload safety analysis report will address the plant specific analysis for the initial ATRIUM 11 reload and confirm that the NFPA acceptance is met.

3.12.5 Demonstration of Applicability of the AURORA-B Methodology for Selected AOOs

The licensee performed a demonstration transient analysis using the AURORA-B AOO methodology in ANP-3904 (Attachment 16a to Reference 6) for a subset of power and flow conditions. The analyses were performed using the plant-specific input parameters for a full core of ATRIUM 11 fuel. The transient events chosen for demonstration are typical limiting events for Browns Ferry as determined from previous cycle analyses and a review of chapter 14 of the UFSAR, as shown in ANP-3904, section 3.1. Since the licensee's analysis is a demonstration analysis, the NRC staff's review is to ensure that the licensee can adequately evaluate AOOs with the AURORA-B AOO methodology and ATRIUM 11 fuel.

The licensee analyzed the following transient events:

- Load Rejection No Bypass (LRNB)
- Turbine Trip No Bypass (TTNB)
- Feedwater Controller Failure (FWCF)
- ASME Overpressurization Analysis
- ATWS Overpressurization Analysis
- EOC Recirculation Pump Trip Out-of-Service (EOC-RPT-OOS)
- Turbine Bypass Valves Out-of-Service (TBVOOS)
- Feedwater Heaters Out-of-Service (FHOOS)
- Power Load Unbalance Out-of-Service (PLUOOS)
- Reduced Feedwater Temperature at Startup

3.12.5.1 Load Rejection No Bypass (LRNB)

The LRNB event is described in section 14.5.2.2.4 of the UFSAR. The licensee analyzed this event for the following conditions within the MELLLA+ power/flow map at the end of full power (EOFP) cycle exposure using the nominal scram speed (NSS) insertion times.

- 100 percent core power, with 105 percent, 95 percent, and 85 percent core flow
- 77.6 percent core power, with 109 percent and 95 percent core flow
- 50 percent core power, with 113 percent core flow
- 26 percent core power above P_{bypass}, with 113 percent core flow
- 26 percent core power above P_{bypass}, with 100 percent core flow

ANP-3904, Table 4.2 shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3904, Table 4.1 shows the ΔMCPR (change in MCPR) at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (percent of rated) = (26/100 below P_{bypass} [power below which direct scram on TSV/TCV closure is bypassed]).

ANP-3904, Figures 4.1, 4.2, and 4.3 show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant operating limits MCPR.

3.12.5.2 Turbine Trip No Bypass (TTNB)

The TTNB event is described in sections 14.5.2.5 and 14.5.2.6 of the UFSAR. The licensee analyzed this event for the same power/flow conditions as for the LRNB event within the EFW power/flow map at the EOFP cycle exposure, using NSS insertion times.

ANP-3904, Table 4.3 shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3904, Table 4.1 shows the ΔMCPR at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (percent of rated) = (26/100 below P_{bypass}).

ANP-3904, Figures 4.4, 4.5, and 4.6 show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant operating limits MCPR.

3.12.5.3 Feedwater Controller Failure (FWCF)

The FWCF event is described in sections 14.5.8.1, 14.5.8.2, and 14.5.8.3 of the UFSAR. The licensee analyzed this event for the same power/flow conditions as for the LRNB event within the EFW power/flow map at the EOFP cycle exposure, using NSS insertion times.

ANP-3904, Table 4.4 shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3904, Table 4.1 shows the ΔMCPR at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (percent of rated) = 26/100 below P_{bypass}.

ANP-3904, Figures 4.7, 4.8, and 4.9 show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant operating limits MCPR.

3.12.5.4 ASME Overpressurization Analysis

The ASME overpressurization event can be initiated by the MSIV closure event at power similar to other steam line closure events resulting in a rapid pressurization of the reactor initiating a reactor scram. The rapid pressurization of the core results in a decrease in void fraction which in turn causes a rapid increase in power. To demonstrate the applicability of the AURORA-B AOO methodology for ASME overpressurization analysis, the licensee analyzed the event initiated by MSIV closure at 102 percent power with 105 percent flow, and 102 percent power with 85 percent flow at the latest exposure in the cycle design. The licensee stated the following assumptions for the analysis:

- The most critical active component (direct scram on valve position) was assumed to fail. However, scram on high neutron flux and high dome pressure is available.
- The plant configuration analyzed assumed that one of the lowest setpoint SRVs was inoperable.
- TS scram speed insertion times were used.
- The initial dome pressure was set at the maximum allowed by the TSs plus an additional 5 psi bias, i.e., 1070 pounds per square inch absolute (psia) (1055 pounds per square inch gauge (psig)).

- A fast MSIV closure time of 3.0 seconds was used.
- The analytical limit ATWS-RPT setpoint and function were assumed.
- The SRV opening setpoints used in the analysis were set to the TS values increased by 3 percent, plus an additional 5 psi.

The acceptance criteria are to demonstrate compliance with the ASME Code by showing that the pressure in the reactor coolant and main steam systems remain below 110 percent of the design values.

ANP-3904, Table 4.5 presents the analysis results and Table 4.6 presents the sequence of event timing for the event at 102 percent power with 105 percent core flow. The maximum reactor lower plenum pressure is 1346 psig at 102 percent power and 105 percent core flow, and the maximum dome pressure is 1313 psig and are less than the acceptance criteria of maximum reactor pressure limit of 1375 psig and dome pressure limit of 1325 psig.

ANP-3904, Figures 4.10, 4.11, 4.12, and 4.13 show the response of various reactor plant parameters during this event.

The NRC staff finds the licensee's analysis for the ASME overpressurization event using the AURORA-B AOO methodology acceptable because based on conservative assumptions, the results show that the maximum reactor pressure during the limiting MSSV closure event does not exceed the ASME safety limit 110 percent (1.1 x 1250 = 1375 psig) of the reactor design pressure (1250 psig). The SRVs have sufficient capacity and performance to prevent the reactor vessel pressure from reaching the safety limit of 110 percent of the reactor design pressure.

3.12.5.5 ATWS Overpressurization Analysis

The licensee analyzed the ATWS overpressurization using the AURORA-B AOO methodology for MSIV closure and PRFO transients at 100 percent power with 85 percent flow and 100 percent power with 105 percent flow at the BOC exposure. The PRFO transient causes the TCV and TBV to open such that steam flow increases until the maximum combined steam flow limit is attained. The system pressure decreases until the low main steam line pressure setpoint is reached, resulting in the closure of all MSIVs. The resulting pressurization wave causes a decrease in core voids and an increase in core pressure thereby increasing the core power. The licensee stated the following assumptions for the analysis:

- The analytical limit ATWS-RPT setpoint and function were assumed.
- To support the operation with one SRVOOS, the plant configuration analyzed assumed that one of the lowest setpoint SRVs was inoperable.
- All scram functions were disabled.
- Initial dome pressure was set at the nominal pressure of 1050 psia.
- The MSIV closure is based on a nominal closure time of 4.0 seconds for both events.
- The SRV opening setpoints used in the analysis were set to the TS values increased by 3 percent, plus an additional 5 psi.

The acceptance criteria for this transient are based on GDC 10, 15, and 26. The main criteria is to demonstrate compliance with the ASME Code by showing that the pressure in the reactor coolant and main steam systems remain below the ASME Service Level C of 120 percent of the design values.

ANP-3904, Table 4.5 presents the results of the MSIV closure overpressurization analysis and Table 4.7 presents the sequence of event timing at 100 percent power with 85 percent core flow. The maximum reactor lower plenum pressure is 1454 psig, and the maximum dome pressure is 1438 psig both at 100 percent power and 85 percent core flow and are less than the acceptance criteria of maximum reactor pressure limit of 1375 psig and dome pressure limit of 1325 psig.

ANP-3904, Figures 4.14, 4.15, 4.16, and 4.17 show the response of various reactor plant parameters during the ATWS MSIV closure event which results in the maximum reactor vessel pressure.

The NRC staff finds the licensee's analysis for the ATWS overpressurization event using the AURORA-B AOO methodology acceptable because based on conservative assumptions, the results show that the maximum reactor pressure during the limiting MSIV closure transient does not exceed the acceptance criteria of ASME Service Level C limit of 120 percent (1.2 x 1250 = 1500 psig) of the reactor design pressure (1250 psig).

3.12.5.6 EOC Recirculation Pump Trip Out-of-Service (EOC-RPT-OOS)

The licensee analyzed LRNB and FWCF events assuming EOC-RPT-OOS at the EOFP cycle exposure, using NSS insertion times. These events are described in UFSAR sections 14.5.2.2.4 and 14.5.8.2, respectively. During these events, EOC-RPT-OOS (or inoperable) means that no credit is taken for RPT on TSV position or TCV fast closure. The licensee stated that the function of the EOC-RPT feature is to reduce the severity of the core power increase caused by the reactor pressurization transient. The RPT accomplishes this by increasing the void fraction in the core, thereby reducing the reactivity increase resulting from the pressurization transient.

ANP-3904, Table 4.8 shows the ΔMCPR results for the LRNB and FWCF events with EOC-RPT-OOS condition at 100 percent power with 105 percent core flow. For both events the ΔMCPR is **[[ ]]**

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant OLMCPR.

3.12.5.7 Turbine Bypass Valves Out-of-Service (TBVOOS)

The licensee analyzed FWCF event with TBVOOS at the EOFP cycle exposure, using NSS insertion times. The licensee stated that the effect of operation with TBVOOS is a reduction in the system pressure relief capacity, which makes the reactor pressurization more severe. The licensee determined that the FWCF event with TBVOOS is more limiting with respect to ΔMCPR than the LRNB and TTNB events analyzed assuming the TBVOOS condition.

ANP-3904, Table 4.8 shows the ΔMCPR results for the FWCF event with TBVOOS at 100 percent power with 105 percent core flow. The maximum ΔMCPR is **[[ ]]** under the EOC-RPT-OOS condition.

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant OLMCPR.

3.12.5.8 Feedwater Heaters Out-of-Service (FHOOS)

The licensee analyzed LRNB and FWCF events assuming FHOOS at the EOFP cycle exposure, using NSS insertion times assuming a feedwater temperature reduction of 70 degrees Fahrenheit (55 degrees Fahrenheit + 15 degrees Fahrenheit bias). The licensee stated that the effect of reduced feedwater temperature is an increase in core inlet subcooling which changes the axial power shape and core void fraction. In addition, it results in decreasing the steam flow for a given power level because more power is required to increase the reactor coolant enthalpy to saturated conditions. The licensee stated that FWCF events with FHOOS are more severe due to a larger increase in inlet subcooling and core power prior to the pressurization phase of the event compared to LRNB and TTNB events with FHOOS due to the decrease in steam flow relative to nominal conditions.

ANP-3904, Table 4.8 shows the ΔMCPR results for the FWCF event at 100 percent power with 105 percent core flow. The ΔMCPR is **[[ ]]** for FH00S and for FHOOS and TBVOOS is **[[ ]]** both with the EOC-RPT-OOS condition.

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant OLMCPR.

3.12.5.9 Power Load Unbalance Out-of-Service (PLUOOS)

The licensee analyzed the LRNB event assuming PLUOOS at the EOFP cycle exposure, using NSS insertion times. The licensee stated that during normal operation, the power load unbalance (PLU) device is assumed to not function below 50 percent core power. Assuming PLUOOS implies that the PLU device does not function for any power level and does not initiate fast TCV closure. The licensee described the PLUOOS scenario for the LRNB event as follows:

- Initially, the TCVs remain in pressure/speed control mode. There is no direct scram or EOC-RPT on valve motion.
- Loss of load results in increasing turbine speed. Depending on initial power, a turbine overspeed condition may be reached to initiate a turbine trip resulting in scram and EOC-RPT.
- Without a turbine trip signal, scram occurs on either high flux or high dome pressure to terminate the event.

ANP-3904, Table 4.8 presents the ΔMCPR results for the LRNB event at 100 percent power with 105 percent core flow. The maximum ΔMCPR is **[[ ]]** for the PLUOOS with the EOC-RPT-OOS condition.

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant OLMCPR.

3.12.5.10 Reduced Feedwater Temperature at Startup

The licensee analyzed the FWCF event assuming reduced feedwater temperature both with and without TBVOOS at the BOC exposure using NSS insertion times. The licensee stated that during reactor startup, it is beneficial to reduce feedwater temperature to avoid excessive wear on reactor equipment. The desired feedwater temperature is less than the temperature assumed in the FHOOS licensing analyses performed for each cycle. Therefore, the reduced feedwater temperatures are only applicable at 50 percent of rated power, and below.

ANP-3904, Table 4.8 presents the ΔMCPR results for the FWCF event at 50 percent power with 113 percent core flow. The maximum ΔMCPR is **[[ ]]** for the FWCF with startup FHOOS and TBVOOS condition.

The NRC staff finds the calculated ΔMCPR values are reasonable which should be combined with the safety limit MCPR to establish or confirm the plant OLMCPR.

3.12.6 Compliance with NRC Imposed Limitations and Conditions in Safety Evaluation for ANP-10300P-A Revision 1

The AURORA-B AOO methodology has 26 limitations and conditions listed in section 5.0 of the NRC staff's SE for ANP-10300P-A, Revision 1. The licensee's compliance statements for these limitations and condition are provided in ANP-3904, Appendix A. The NRC staff's evaluation on their compliance is given below.

Limitation and Condition 1

> AURORA-B may not be used to perform analyses that result in one or more of its CCDs [component calculational devices] (S-RELAP5, MB2-K, MICROBURN-B2, RODEX4) operating outside the limits of approval specified in their respective TRs, SEs, and plant-specific LARs. In the case of MB2-K, MB2-K is subject to the same limitations and conditions as MICROBURN-B2.

Evaluation

The NRC staff confirmed that these methods are used within their approved ranges and, therefore, finds this limitation and condition is satisfied.

Limitation and Condition 2

> The regulatory limit contained in 10 CFR 50.46(b)(2), requiring cladding oxidation not to exceed 17 percent of the initial cladding thickness prior to oxidation, is based on the use of the Baker-Just oxidation correlation. Because AURORA-B makes use of the Cathcart-Pawel oxidation correlation, this limit shall be reduced to 13 percent, inclusive of pre-transient oxide layer thickness.
>
> Should the NRC staff position regarding the appropriate acceptance criterion for the Cathcart-Pawel correlation change, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.

Evaluation

The licensee stated that **[[ ]]** The NRC staff confirmed that the AURORA-B AOO results meet this limit and, therefore, finds this limitation and condition is satisfied.

Limitation and Condition 3

> Parameter uncertainty distributions and their characterizing upper and lower 2σ levels are presented in Table 3.6 and discussed in Section 3.6 of this SE. The distribution types will not be changed and the characterizing upper and lower 2σ uncertainties will not be reduced without prior NRC approval. In the cases of the parameters **[[ ]]**, the respective methodologies discussed in Section 3.6.4.10 and Section 3.6.4.17 shall be used when determining the associated upper and lower 2σ levels. The **[[ ]]** is subject to Limitation and Condition No. 4, below.

Evaluation

The NRC staff confirmed that the generic uncertainty distributions presented in Table 2.2 of ANP-3904 are consistent with those in Table 3.6 of the SE for the AURORA-B methodology. For the **[[ ]]**, the licensee stated that the range was developed based on the approved process in section 3.6.4.10 of the methodology. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 4

> As discussed in Section 3.3.1.2, before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are modeled in licensing analyses using AURORA-B, AREVA must justify that the AURORA-B EM can acceptably predict void fraction results for the new fuel designs within the **[[ ]]** prediction uncertainty bands. Otherwise, the prediction uncertainty bands should be appropriately expanded, and the **[[ ]]** should be appropriately updated using the methodology discussed in Section O of this SE.

[Note by NRC staff: Section O does not exist in the AURORA-B AOO methodology SE. This is a typographical error as reported by one of the contributors and should be replaced with section 3.6.4.15.]

Evaluation

The NRC staff reviewed the void fraction prediction in section 3.7.3.1 of this SE and found that it was acceptable. The NRC staff finds that the licensee adequately addressed this limitation and condition because the void fraction prediction as evaluated in section 3.7.3.1 of this SE is acceptable.

Limitation and Condition 5

> As discussed in Section 3.3.2.4.4, before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are included in licensing analyses performed using the AURORA-B EF, AREVA must justify that the **[[ ]]** void-quality correlation within MICROBURN-B2 is valid for the new fuel designs at EPU and EFW conditions.

Evaluation

The licensee discussed the void-quality correlation in section 5.1 of ANP-3908. The NRC staff reviewed this in section 3.7.3.1 of this SE and found that it was acceptable. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 6

> The 2σ ranges **[[** **]]** until AREVA supplies additional justification (e.g., as part of a first-time application analysis) demonstrating an acceptable alternative for NRC review and approval. For **[[** **]]** will be utilized when performing licensing analyses to determine peak cladding temperature and maximum local oxidation.
>
> Should the NRC staff position regarding these uncertainties change as a result of additional justification, the NRC will notify AREVA with a letter either revising this limitation or stating that it is removed.

Evaluation

The licensee stated that **[[** **]]** Based on the licensee's commitment, the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 7

> As discussed in Section 3.6 of this AURORA-B methodology SE, licensees should provide justification for the key plant parameters and initial conditions selected for performing sensitivity analyses on an event-specific basis. Licensees should further justify that the input values ultimately chosen for these key plant parameters and initial conditions will result in a conservative prediction of FoMs when performing calculations according to the AURORA-B EM described in ANP-10300P.

Evaluation

In Section 5.3 of ANP-3904, the licensee states:

> As part of the initial preparations for licensing Browns Ferry, Framatome will review the plant parameters document for the key parameters associated with the potentially limiting events. Framatome will also look for parameters that have a range of values that may be allowed for operational flexibility. Likewise, for initial conditions, Framatome will examine the range allowed during normal operation. This will include initial conditions such as power, flow, pressure, and inlet subcooling. Sensitivity studies will be performed for all of these key parameters/conditions for all FoM (MCPR, LHGR, and overpressure) and **[[** **]]**

Based on the above statement, the NRC staff finds the licensee has satisfied this limitation and condition for the first ATRIUM 11 fuel application cycle by committing to review the plant parameters that have a range of values during normal operation which includes initial conditions such as power, flow, pressure, and inlet subcooling. The licensee has also committed to perform sensitivity studies for all of these key parameters/conditions for all FoM to **[[**

]]

<u>Limitation and Condition 8</u>

> The sampling ranges for uncertainty distributions used in the non-parametric order statistics analyses will be truncated at no less than ±6σ **[[**
>
> **]]**

[Note by NRC staff: Section O does not exist in the AURORA-B AOO methodology SE. This is a typographical error as reported by one of the contributors and should be replaced with section 3.6.4.15.]

<u>Evaluation</u>

ANP-3904, Table 2.2 shows the sampling ranges for the uncertainty distributions used in the analysis. The licensee stated that **[[**

]] Based on the licensee's statement, the NRC staff finds the limitation and condition is satisfied.

<u>Limitation and Condition 9</u>

> For any highly ranked PIRT phenomena whose uncertainties are not addressed in a given non-parametric order statistics analysis via sampling, AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.2 of this SE. For any pertinent medium ranked PIRT phenomena whose uncertainties are not addressed in a given non-parametric order statistics analysis via sampling, AREVA will address the associated uncertainties by modeling the phenomena as described in Table 3.4 of this SE.

<u>Evaluation</u>

The licensee stated that it complied with the requirements of Tables 3.2 and 3.4 of the SE for AURORA-B AOO methodology as they relate to this limitation. The NRC staff confirmed that the licensee modeled the phenomena as described in Tables 3.2 and 3.4 of the SE for the AURORA-B AOO methodology and, therefore, has adequately addressed this limitation and condition.

Limitation and Condition 10

> The assumptions of **[[**
> **]]** will be used in the AURORA-B EM to ensure the uncertainty in SL03: **[[**
> **]]** is conservatively accounted for.

Evaluation

The licensee stated that it complied with the requirements of Tables 3.2 and 3.4 of the SE for AURORA-B AOO methodology as they relate to this limitation. The NRC staff confirmed that the licensee modeled the phenomena as described in Tables 3.2 and 3.4 of SE of the AURORA-B AOO methodology and, therefore, has adequately addressed this limitation and condition.

Limitation and Condition 11

> AREVA will provide justification for the uncertainties used for the highly ranked plant-specific PIRT parameters C12, R01, R02, and SL02 on a plant-specific basis, as described in Table 3.2 of this SE [for AURORA-B methodology].

Evaluation

In ANP-3904, section 5.3, the licensee provided the following commitments for the uncertainties of the highly ranked plant specific PIRT parameters. The licensee stated that **[[**

]]

For the parameter C12, the licensee stated:

> The parameter C12 is the **[[**
>
> **]]**

For the parameter R01, the licensee stated:

> **[[**

]]

For the parameter R02, the licensee stated:

> [[
>
>]]

For the parameter SL02, the licensee stated:

> [[
>
>]]

Based on the licensee's commitments described above, the NRC staff finds that the uncertainties of the plant-specific parameters C12, R01, R02, and SL02 will be adequately justified and, therefore, this limitation and condition is satisfied.

Limitation and Condition 12

> When applying the AURORA-B EM to the **[[** **]]**, any changes to AURORA-B to enhance **[[** **]]** on a plant-specific basis without prior NRC review and approval are not approved as part of this SE, as described in Table 3.2 of this SE.

Evaluation

The licensee has committed to follow the conservative method described in ANP-3908, Revision 4, section 6.3.1 for transient mixing determination in the **[[** **]]** for the analysis of transients **[[** **]]** The licensee will provide the results and conclusions of this analysis as a part of the initial cycle reload safety analysis report (RSAR) of ATRIUM 11 fuel. Based on this commitment, the NRC finds this limitation and condition is satisfied.

Limitation and Condition 13

> The AURORA-B uncertainty methodology discussed in Section 3.6 of this SE may be used in licensing applications for the events listed in Section 3.1 of this SE, with the exception of three specific events identified in Section 3.6.2 of this SE: **[[**

> **]]** These events are generally expected to be benign and hence non-limiting. While the NRC staff's review concluded that the AURORA-B EM contains code models and correlations sufficient for simulating these events, the uncertainty methodology developed in the TR did not address certain important phenomena or conditions associated therewith. Therefore, while licensing applications may rely on nominal calculations with the AURORA-B EM for these events in the course of demonstrating that all regulatory limits are satisfied with significant margin, the existing uncertainty methodology may not be applied directly to these specific events.

<u>Evaluation</u>

The licensee stated that the dispositions of these events are non-limiting in the UFSAR; therefore, no additional evaluation is required. Regarding the **[[ ]]** event, the NRC staff notes that the existing uncertainty methodology applies to HPCI and not to **[[ ]]** Since Browns Ferry does not have an **[[ ]]** system, the NRC staff finds that the licensee adequately addressed this limitation and condition.

<u>Limitation and Condition 14</u>

> The scope of the NRC staff's approval for AURORA-B does not include the ABWR design.

<u>Evaluation</u>

This limitation and condition is not applicable because the Browns Ferry units are not an ABWR design.

<u>Limitation and Condition 15</u>

> For application to BWR/2s at EPU or EFW conditions, plant-specific justification should be provided for the applicability of AURORA-B, as discussed in Section 3.1 of this SE.

<u>Evaluation</u>

This limitation and condition is not applicable because the Browns Ferry units are not a BWR/2 design.

<u>Limitation and Condition 16</u>

> **[[ ]]** is not sampled as part of the methodology, justification should be provided on a plant-specific basis that a conservative flow rate has been assumed **[[ ]]**

Evaluation

The licensee stated that the plant parameter document contains the **[[ ]]** However, for the transient analysis, the AURORA-B model **[[ ]]** The NRC staff therefore finds that this limitation and condition is satisfied.

Limitation and Condition 17

> If the AURORA-B EM calculates that the film boiling regime is entered during a transient or accident, AREVA must justify that the uncertainty associated with heat transfer predictions in the film boiling regime is adequately addressed.

Evaluation

The licensee stated that **[[ ]]** for Browns Ferry. Therefore, the NRC staff finds limitation and condition is satisfied.

Limitation and Condition 18

> As discussed in Section 3.6.5 of this SE regarding conservative measures:
>
> a. Plant-specific licensing applications shall describe and provide justification for the method for determining and applying conservative measures in future deterministic analyses for each FoM (e.g., biasing calculational inputs, post-processing adjustments to calculated nominal results), and
>
> b. If the 95/95 FoM for a given parameter calculated according to the defined conservative measures during a deterministic analysis shows a difference in magnitude exceeding 1σ from the corresponding value calculated in the most recent baseline full statistical analysis, AREVA must re-perform the full statistical analysis for the affected scenario and determine new conservative measures.

Evaluation:

a. The FoM considered are **[[ ]]**

For the **[[ ]]**, the licensee stated that **[[ ]]** The NRC staff, therefore, finds it acceptable that **[[ ]]**

For the LHGRFACp evaluations, the licensee stated that **[[ ]]** The NRC staff reviewed the description **[[in section 5.3 of ANP-3904]]** and finds it acceptable because the **[[**

]]

b. The licensee stated that [[
]]

Based on these commitments, the NRC staff finds that this limitation and condition is satisfied.

Limitation and Condition 19

As discussed in Section 3.6.5 of this SE [for ANP-10300P-A], the following stipulations are necessary to ensure that the FoMs calculated by AREVA in accordance with ANP-10300P would satisfy the 95/95 criterion:

a. AREVA will use multivariate order statistics when multiple FoMs are drawn from a single set of statistical calculations,

b. AREVA will choose the sample size prior to initiating statistical calculations,

c. AREVA will not arbitrarily discard undesirable statistical results, and

d. AREVA will maintain an auditable record to demonstrate that its process for performing statistical licensing calculations has been executed in an unbiased manner.

Evaluation

To satisfy this limitation and condition the licensee made the following commitments:

a. Framatome calculations will utilize the multivariate order statistics when a single transient is used to determine multiple FoMs.

b. Framatome will choose the sample size prior to initiating statistical calculations.

c. Framatome will not arbitrarily discard undesirable statistical results.

d. Framatome will maintain an auditable record to demonstrate the process for performing statistical licensing calculations is being executed in an unbiased manner.

Based on these commitments, the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 20

The implementation of any new methodology within the AURORA-B EM (i.e., replacement of an existing CCD) is not acceptable unless the AURORA-B EM with the new methodology incorporated into it has received NRC review and approval. An existing NRC-approved methodology cannot be implemented within the AURORA-B EM without NRC review of the updated EM.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the licensee used the AURORA-B EM as described in ANP-10300P-A and CCDs as described in ANP-10300P-A are not replaced.

Limitation and Condition 21

> NRC-approved changes that revise or extend the capabilities of the individual CCDs comprising the AURORA-B EM may not be incorporated into the EM without prior NRC approval.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the Framatome regulatory procedures require use of NRC-approved methodologies within the applicability defined for that methodology.

Limitation and Condition 22

> As discussed in Section 3.3.1.5 and Section 4.0 of this SE, the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EF. Other CPR correlations (existing and new) that would be used with the AURORA-B EM must be reviewed and approved by the NRC or must be developed with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel" [Reference 58]. Furthermore, if transient thermal-hydraulic simulations are performed in the process of applying AREVA CPR correlations to co-resident fuel, these calculations should use the AURORA-B methodology.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the Framatome regulatory procedures require use of NRC staff-approved methodologies. The applicability of the ACE/ATRIUM 11 correlation for use in the AURORA-B AOO methodology is described in NRC staff-approved TR ANP-10335P-A.

Limitation and Condition 23

> Except when prohibited elsewhere, the AURORA-B EM may be used with new or revised fuel designs without prior NRC approval provided that the new or revised fuel designs are substantially similar to those fuel designs already approved for use in the AURORA-B EM (i.e., thermal energy is conducted through a cylindrical ceramic fuel pellet surrounded by metal cladding, flow in the fuel channels develops into a predominantly vertical annular flow regime, etc.). New fuel designs exhibiting a large deviation from these behaviors will require NRC review and approval prior to their implementation in AURORA-B.

Evaluation

The NRC staff finds this limitation and condition is satisfied because, as stated by the licensee, Framatome has no fuel designs that exhibit a large deviation from the behaviors described in this limitation and condition. Framatome committed that if a fuel design is developed that is significantly different, this fuel design will be submitted to the NRC for approval.

Limitation and Condition 24

> Changes may be made to the AURORA-B EM in the **[[**
>
> **]]** areas discussed in Section 4.0 of this SE without prior NRC approval.

Evaluation

The licensee stated that **[[**
]] The NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 25

> The parallelization of individual CCDs may be performed without prior NRC approval as discussed in Section 4.0 of this SER.

Evaluation:

The licensee stated that **[[**
]] The NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 26

> AREVA must continue to use existing regulatory processes for any code modifications made in the **[[**
> **]]**
> areas discussed in Section 4.0 of this SE.

Evaluation

The licensee stated that **[[**
]]
The NRC staff finds the licensee's evaluation acceptable because the **[[**

]]

3.12.7 Technical Conclusion

The following is a summary of NRC staff technical conclusions for the ATRIUM 11 transient demonstration:

- The licensee appropriately justified the use of AURORA-B AOO methodology for analyzing transient events for Browns Ferry.

- For the UFSAR chapter 14 events, the licensee identified (a) events that should be analyzed at each reload, (b) events that should be analyzed at the first reload only, and (c) events for which no further analysis is necessary. The NRC staff finds the licensee's dispositions acceptable.

- The NRC staff reviewed ATRIUM 11 transient demonstration analysis and finds that the licensee analyzed the potentially limiting events and their results are realistic and meet the specified acceptance criteria.

- All limitations and conditions for using the AURORA-B AOO methodology documented in TR ANP-10300P-A, Revision 1 are satisfied.

- Compliance with the applicable regulatory requirements has been demonstrated.

The NRC staff reviewed all limitations and conditions and finds that each was adequately addressed by the licensee for the demonstration case and will be supported by the RSAR when it will be submitted.

3.13 TRAVELER TSTF-564 SLMCPR

3.13.1 Introduction and Background

3.13.1.1 Background on Boiling Transition

During steady-state operation in a BWR, most of the coolant in the core is in a flow regime known as annular flow. In this flow regime, a thin liquid film is pushed up the surface of the fuel rod cladding by the bulk coolant flow, which is mostly water vapor with some liquid water droplets. This provides effective heat removal from the cladding surface; however, under certain conditions, the annular film may dissipate, which reduces the heat transfer and results in an increase in fuel cladding surface temperature. This phenomenon is known as boiling transition or dryout. The elevated surface temperatures resulting from dryout may cause fuel cladding damage or failure.

3.13.1.2 Background on Critical Power Correlations

For a given set of reactor operating conditions (pressure, flow, etc.), dryout will occur on a fuel assembly at a certain power, known as the critical power. Because the phenomena associated with boiling transition are complex and difficult to model purely mechanistically, thermal-hydraulic test campaigns are undertaken using electrically heated prototypical fuel bundles to establish a comprehensive database of critical power measurements for each BWR fuel product. These data are then used to develop a critical power correlation that can be used to predict the critical power for assemblies in operating reactors. This prediction is usually expressed as the ratio of the actual assembly power to the critical power predicted using the correlation, known as the CPR.

One measure of the correlation's predictive capability is based on its validation relative to the test data. For each point *j* in a correlation's test database, the experimental critical power ratio

(ECPR) is defined[1] as the ratio of the predicted critical power to the measured critical power (References 59 and Attachment 5a to Reference 1) or:

$$\mathrm{ECPR_j} = \frac{\text{Predicted Critical Power}_\mathrm{j}}{\text{Measured Critical Power}_\mathrm{j}}$$

For ECPR values greater than or equal to 1, the calculated critical power is greater than the measured critical power and the prediction is considered to be non-conservative. Because the measured critical power includes random variations due to various uncertainties, evaluating the ECPR for all of the points in the dataset (or, ideally, a subset of points that were not used in the correlation's development) results in a probability distribution. This ECPR distribution allows the predictive uncertainty of the correlation to be determined. This uncertainty can then be used to establish a limit above which there can be assumed that boiling transition will not occur (with a certain probability and confidence level).

3.13.1.3 Background on Thermal-Hydraulic Safety Limits

To protect against boiling transition, BWRs have implemented an SL on the CPR, known as the MCPR SL. As discussed in NUREG-1433 and NUREG-1434 for General Electric BWR designs, the current basis of the MCPR SL is to prevent 99.9 percent of the fuel in the core from being susceptible to boiling transition. This limit is typically developed by considering various cycle-specific power distributions and uncertainties and is highly dependent on the cycle-specific radial power distribution in the core. As such, the limit may need to be updated as frequently as every cycle.

The fuel cladding SL for pressurized-water reactor (PWR) designs, described in the STS for Babcock & Wilcox, Westinghouse, and Combustion Engineering plants in NUREG-1430[2], NUREG-1431[3], and NUREG-1432[4], respectively, correspond to a 95 percent probability at a 95 percent confidence level that departure from nucleate boiling will not occur. As a result of the overall approach taken in developing the PWR limits, they are only dependent on the fuel type(s) in the reactor and the corresponding departure from nucleate boiling ratio (DNBR) correlations. The limits are not cycle-dependent and are typically only updated when new fuel types are inserted in the reactor.

BWRs also have an LCO that governs MCPR, known as the MCPR operating limit (OL). The OL on MCPR is an LCO which must be met to ensure that AOOs do not result in fuel damage. The current MCPR OL is calculated by combining the largest change in CPR from all analyzed transients, also known as the ΔCPR, with the MCPR SL.

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1 Equation 1.1 of TSTF-564, Revisions 1 and 2, and Equation 1 of the corresponding model safety evaluation defined the ECPR as the measured critical power divided by the critical power from correlation. The TSTF published a notification regarding this error in Reference 59.
2 NUREG-1430, Revision 5.0, "Standard Technical Specifications: Babcock and Wilcox Plants," Volume 1, "Specifications," and Volume 2, "Bases," September 2021 (ML21272A363 and ML21272A370)
3 NUREG-1431, Revision 5.0, "Standard Technical Specifications: Westinghouse Plants," Volume 1, "Specifications," and Volume 2, "Bases," September 2021 (ML21259A155 and ML21259A159)
4 NUREG-1432, Revision 5.0, "Standard Technical Specifications: Combustion Engineering Plants," Volume 1, "Specifications," and Volume 2, "Bases," September 2021 (ML21258A421 and ML21258A424)

3.13.2 Regulatory Evaluation

3.13.2.1 Description of TS Sections

Traveler TSTF-564 modifies STS 2.1.1, “Reactor Core SLs.” Safety limits ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and AOOs.

Browns Ferry TS 2.1.1.2 currently requires that with the reactor steam dome pressure greater than or equal to (≥) 585 psig and core flow ≥ 10 percent rated core flow, MCPR shall be ≥ 1.06 for two recirculation loop operation or ≥ 1.08 for single recirculation loop operation. The MCPR SL ensures that 99.9 percent of the fuel in the core is not susceptible to boiling transition.

Traveler TSTF-564 also modifies STS 5.6.3, “Core Operating Limits Report (COLR).” Browns Ferry TS 5.6.5 requires core operating limits to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle. These limits are required to be documented in the COLR.

3.13.2.2 Proposed Changes to the TS

The licensee proposed to revise the MCPR SL to make it cycle-independent, consistent with the method described in Traveler TSTF-564, Revision 2.

The proposed changes to the Browns Ferry TS revise the value of the MCPR SL in TS 2.1.1.2 to 1.05, with corresponding changes to the associated bases. The change to TS 2.1.1.2 replaces the existing separate SLs for single- and two-recirculation loop operation with a single limit since the revised SL is no longer dependent on the number of recirculation loops in operation.

The $MCPR_{99.9\%}$ (i.e., the current MCPR SL) is an input to the MCPR OL in LCO 3.2.2, “Minimum Critical Power Ratio (MCPR).” While the definition and method of calculation of both the $MCPR_{99.9\%}$ and the LCO 3.2.2 MCPR OL remains unchanged, the proposed TS changes include revisions to TS 5.6.5, to require the $MCPR_{99.9\%}$ value used in calculating the LCO 3.2.2 MCPR OL to be included in the cycle-specific COLR.

3.13.2.3 Applicable Regulatory Requirements and Guidance

Under 10 CFR 50.36(a)(1), an applicant for an operating license must include in the application proposed TSs in accordance with the requirements of 10 CFR 50.36. The applicant must include in the application, a “summary statement of the bases or reasons for such specifications, other than those covering administrative controls.” However, per 10 CFR 50.36(a)(1), these TS bases “shall not become part of the technical specifications.”

As required by 10 CFR 50.36(c)(1), TSs will include items for safety limits, limiting safety system settings, and limiting control settings. As required by 10 CFR 50.36(c)(1)(i)(A), safety limits for nuclear reactors are “limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission.”

As required by 10 CFR 50.36(c)(2)(i), the TSs will include LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met.

General Design Criterion 10, “Reactor design,” of 10 CFR Part 50 Appendix A, “General Design Criteria of Nuclear Power Plants,” states:

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Most plants have a plant-specific design criterion similar to GDC 10. The limit placed on the MCPR acts as a specified acceptable fuel design limit to prevent boiling transition, which has the potential to result in fuel rod cladding failure.

The NRC staff’s review guidance contained in Revision 2 of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 4.4, “Thermal and Hydraulic Design,” (Reference 23) provides the following two examples of acceptable approaches to meeting the SRP acceptance criteria for establishing fuel design limits (as stated in SRP Acceptance Criterion 1):

A. For departure from nucleate boiling ratio (DNBR), CHFR [critical heat flux ratio] or CPR correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs.
B. The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs.

3.13.3 Technical Evaluation

3.13.3.1 Basis for the Proposed Change

As discussed in section 2.3 of the LAR and section 3.13.1.3 this SE, the current MCPR SL (i.e., the $MCPR_{99.9\%}$), is affected by the plant’s cycle-specific core design, especially including the core power distribution, fuel type(s) in the reactor, and the power-to-flow operating domain for the plant. As such, it is frequently necessary to change the MCPR SL to accommodate new core designs. Changes to the MCPR SL are usually determined late in the design process and necessitate an accelerated NRC review (i.e., license amendment request) to support the subsequent fuel cycle.

The licensee proposed to change the basis for the MCPR SL for Browns Ferry so that it is no longer cycle-dependent, reducing the frequency of revisions and eliminating the need for NRC’s review on an accelerated schedule. The proposed revised basis for the MCPR SL aligns it with that of the DNBR SL used in PWRs, which, as previously noted in section 3.13.2.3 of this SE, provides a 95 percent probability at a 95 percent confidence level that no fuel rods will experience departure from nucleate boiling.

The intent of the proposed basis for the revised MCPR SL is acceptable to the NRC staff based on the discussion in SRP section 4.4, SRP Acceptance Criterion 1. The remainder of this SE is devoted to ensuring that the methodology for determining the revised MCPR SL provides the intended result, that the revised MCPR SL can be adequately determined in the core using various types of fuel, that the proposed SL continues to fulfil the necessary functions of an SL without unintended consequences, and that the proposed changes have been adequately implemented in the Browns Ferry TS.

3.13.3.2 Revised MCPR SL Definition

As discussed in section 3.13.1.2 of this SE, a critical power correlation's ECPR distribution quantifies the uncertainty associated with the correlation. Traveler TSTF-564, Revision 2, provides a definition for a limit that bounds 95 percent of a correlation's ECPR distribution at a 95 percent confidence level, according to the following formula:

$$MCPR_{95/95}(i) = \mu_i + \kappa_i\sigma_i$$

Where μ_i is the correlation's mean ECPR, σ_i is the standard deviation of the correlation's ECPR distribution, and κ_i is a statistical parameter chosen to provide "95% probability at 95% confidence (95/95) for the one-sided upper tolerance limit that depends on the number of samples (N_i) in the critical power database." This formula is commonly used to determine a 95/95 one-sided upper tolerance limit for a normal distribution, which is appropriate for the situation under consideration. The factor κ is generally attributed to D. B. Owen (Reference 60) and was also reported by M. G. Natrella, as referenced in Traveler TSTF-564, Revision 2.

In the LAR, the licensee proposes variations from the TS changes described in Traveler TSTF-564. That is, Browns Ferry is transitioning from ATRIUM 10XM to ATRIUM 11 fuel, neither of which are identified in Table 1 of Traveler TSTF-564. As discussed in Traveler TSTF-564, other fuel vendors may determine the $MCPR_{95/95}$ for other fuel designs using the methodology. The licensee in the LAR provided the required description of the deviations of the $MCPR_{95/95}$ for ATRIUM 10XM and ATRIUM 11, which is based on the information contained in each fuel type's NRC staff-approved CPR correlation that is referenced in Browns Ferry TS 5.6.5.b. The NRC staff finds that the difference is within the scope of the Traveler TSTF-564 approval and does not affect the applicability of Traveler TSTF-564 to the Browns Ferry TS.

As discussed by Piepel and Cuta (Reference 61) for DNBR correlations, the acceptability of this approach is predicated on a variety of assumptions, including the assumptions that the correlation data comes from a common population and that the correlation's population is distributed normally. These assumptions are typically addressed generically when a critical power or critical heat flux correlation is reviewed by the NRC staff, who may apply penalties to the correlation to account for any issues identified. Traveler TSTF-564, Revision 2, states that such penalties applied during the NRC's review of the critical power correlation would be imposed on the mean or standard deviation used in the calculating the $MCPR_{95/95}$. These penalties would also continue to be imposed in the determination of the $MCPR_{99.9\%}$, along with any other penalties associated with the process of (or other inputs used in) determining the $MCPR_{99.9\%}$ (e.g., penalties applied to the $MCPR_{99.9\%}$ SL for operation in the MELLLA+ operating domain).

The NRC staff finds the definition of the $MCPR_{95/95}$ will appropriately establish a 95/95 upper tolerance limit on the critical power correlation and that any issues in the underlying correlation will be addressed through penalties on the correlation mean and standard deviation, as necessary. Therefore, the NRC staff concludes that the $MCPR_{95/95}$ definition, as proposed, establishes an acceptable fuel design limit and is acceptable.

3.13.3.3 Determination of a Revised MCPR SL for Mixed Cores

Traveler TSTF-564, Revision 2, proposed that a core containing a variety of fuel types would evaluate the $MCPR_{95/95}$ for all of the fresh and once-burnt fuel in the core and apply the most limiting (i.e., the largest) value of $MCPR_{95/95}$ for each of the applicable fuel types as the MCPR SL. As stated in section 3.1 of Traveler TSTF-564, Revision 2, this is because bundles that are twice-burnt or more at the beginning of the cycle have significant MCPR margin relative to the fresh and once-burnt fuel. The justification is that the MCPR for twice-burnt and greater fuel is far enough from the MCPR for the limiting bundle that its probability of boiling transition is very small compared to the limiting bundle and it can be neglected in determining the SL. Results of a study provided in the traveler indicate that this is the case even for fuel operated on short (12-month) reload cycles. As discussed in the traveler, twice-burnt or greater fuel bundles are included in the cycle-specific evaluation of the $MCPR_{99.9\%}$ and the MCPR OL. If a twice-burnt or greater fuel bundle is found to be limiting, it would be governed by the MCPR OL, which will always be more restrictive than both the $MCPR_{95/95}$ and the $MCPR_{99.9\%}$. The NRC staff found this justification to be appropriate and determined that it is acceptable to determine the $MCPR_{95/95}$ SL for the core based on the most limiting value of the $MCPR_{95/95}$ for the fresh and once-burnt fuel in the core.

The NRC staff reviewed the information furnished by the licensee and determined that the process for establishing the revised MCPR SL for mixed cores ensures that the limiting fuel types in the core will be evaluated and the limiting $MCPR_{99.9\%}$ will be appropriately applied as the SL. The NRC staff therefore found this process to be acceptable.

3.13.3.4 Relationship between MCPR Safety and Operating Limits

As discussed in the Traveler TSTF-564, Revision 2, the $MCPR_{99.9\%}$ is expected to always be greater than the $MCPR_{95/95}$ for two reasons. First, because the $MCPR_{99.9\%}$ includes uncertainties not factored into the $MCPR_{95/95}$, and second, because the 99.9 percent probability basis for determining the $MCPR_{99.9\%}$ is more conservative than the 95 percent probability at a 95 percent confidence level used in determining the $MCPR_{95/95}$. The level of conservatism in the $MCPR_{95/95}$ SL is appropriate because the lead fuel rod in the core (i.e., the limiting fuel rod with respect to MCPR) is used to evaluate whether any fuel rods in the core are susceptible to boiling transition, which is also discussed in the traveler). This is consistent with evaluations performed for PWRs using a 95/95 upper tolerance limit on the correlation uncertainty as an SL.

Consistent with Traveler TSTF-564, Revision 2, the MCPR OL defined in LCO 3.2.2 would continue to be evaluated using the $MCPR_{99.9\%}$ as an input. The $MCPR_{99.9\%}$ will continue to be evaluated in the same way as it is currently, using the whole core. The licensee is not proposing a change to LCO 3.2.2 and will continue to determine the MCPR operating limits for LCO 3.2.2 at Browns Ferry.

Consistent with Traveler TSTF-564, Revision 2, the licensee proposed to revise the COLR TS (i.e., TS 5.6.5) to require the cycle-specific value of the $MCPR_{99.9\%}$ to be included in the COLR. The methods supporting the inclusion of the $MCPR_{99.9\%}$ must also therefore, be included in the list of COLR references contained in TS 5.6.5.b. The changes to Browns Ferry TS 5.6.5.b support that the uncertainties being removed from the MCPR SL are still included as part of the MCPR OL and will continue to appropriately inform plant operation.

Based on the review, the NRC staff, therefore, finds that the changes proposed by the licensee will retain an adequate level of conservatism in the MCPR SL in TS 2.1.1.2 while appropriately ensuring that plant- and cycle-specific uncertainties will be retained in the MCPR OL. The $MCPR_{95/95}$ represents a lower limit on the value of the $MCPR_{99.9\%}$, which should always be higher since it accounts for numerous uncertainties that are not included in the $MCPR_{95/95}$ (as discussed in section 3.1 of Traveler TSTF-564, Revision 2).

3.13.3.5 Implementation of the Revised MCPR Safety Limit in the TS

The licensee proposed to change the value of the SL in Browns Ferry TS 2.1.1.2 for ATRIUM 10XM and ATRIUM 11 to ≥ 1.05. The value reported in TS 2.1.1.2 was calculated using Equation 1 from Traveler TSTF-564, Revision 2, with the exception that ECPR was calculated using Equation 3 of ANP-3857 (Attachment 5a to Reference 1), which is the inverse of the quantity used in Traveler TSTF-564, Revision 2, as is discussed in section 3.13.1.2 of this SE. The value was reported at a precision of two digits past the decimal point with the hundredths digit rounded up.

Consistent with Traveler TSTF-564, Revision 2, the licensee also proposed to modify Browns Ferry TS 5.6.5 to include the value of the $MCPR_{99.9\%}$ to ensure that the cycle-specific $MCPR_{99.9\%}$ value will continue to be determined for LCO 3.2.2 and reported in the COLR. The COLR, therefore, will continue to report the cycle-specific value of the MCPR OL contained in LCO 3.2.2, and TS 5.6.5.b will continue to reference appropriate NRC staff-approved methodologies for determination of the $MCPR_{99.9\%}$ and the MCPR OL. Therefore, the NRC staff finds the proposed change to Browns Ferry TS 5.6.5 to be acceptable.

In the LAR, the licensee provides the details of the calculation of the $MCPR_{95/95}$ for ATRIUM 10XM using the statistics from the ACE/ATRIUM 10XM CPR correlation database contained in ANP-10298P-A, Revision 1 (Reference 30). The licensee also provides the details of the calculation of the $MCPR_{95/95}$ for ATRIUM 11 using the statistics from the ACE/ATRIUM 11 CPR correlation database contained in ANP-10335P-A, Revision 0 (Reference 31).

The NRC staff assessed the licensee's deviations for ATRIUM 10XM and ATRIUM 11 and determined that they are consistent with the process described in Traveler TSTF-564, Revision 2. The NRC staff, therefore, finds the proposed change to the SL in TS 2.1.1.2 is acceptable because the licensee derived the SL consistent with the process described in Traveler TSTF-564, Revision 2.

The NRC staff notes that Browns Ferry TSs have a different numbering than the STS for the Core Operating Limits Report; specifically, Browns Ferry TS 5.6.5 versus STS 5.6.3. The NRC staff confirmed that the licensee made appropriate conforming changes in its proposal to adopt this TSTF traveler.

3.13.4 <u>Technical Conclusion Concerning Adoption of Traveler TSTF-564, Revision 2</u>

The NRC staff reviewed the licensee's proposed TS changes and determined that the proposed SL associated with TS 2.1.1.2 was calculated in a manner consistent with the process described in Traveler TSTF-564, Revision 2, and was therefore acceptably modified to suit the revised definition of the MCPR SL. Under the new definition, the MCPR SL will continue to protect the fuel cladding against the uncontrolled release of radioactivity by preventing the onset of boiling transition, thereby fulfilling the requirements of 10 CFR 50.36(c)(1) for SLs. The MCPR OL in LCO 3.2.2 remains unchanged and will continue to meet the requirements of 10 CFR 50.36(c)(2) and GDC 10, by ensuring that no fuel damage results during normal operation and AOOs. The NRC staff determined that the proposed changes to TS 5.6.5 are acceptable; upon adoption of the revised MCPR SL, the COLR will be required to contain the $MCPR_{99.9\%}$, supporting the determination of the MCPR OL using current methodologies.

3.14 TECHNICAL EVALUATION CONCLUSION

The NRC staff reviewed the licensee's analyses related to the effect of the proposed amendments for Browns Ferry to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed MELLLA+ operating domain under extended power uprate conditions. The NRC staff further reviewed the licensee's proposed changes to TS 5.6.5.b that support adoption of the intended Framatome analysis methodologies and to TS 2.1.1.2 to revise the SLMCPR. Based on its review and for the reasons summarized in this SE, the NRC staff concludes that the proposed amendments for Browns Ferry are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendments on January 5, 2023. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The NRC has previously issued a proposed finding that the amendments involve no significant hazards consideration in the *Federal Register* on March 8, 2022 (87 FR 13014), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from Barstow, J., TVA, to U.S. NRC, “Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535),” dated July 23, 2021 (ML21204A128).
2. Letter from Barstow, J., TVA, to U.S. NRC, “Supplement 1 to Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) (EPID L-2021-LLA-0132),” dated April 8, 2022 (ML22098A188).
3. Letter from Barstow, J., TVA, to U.S. NRC, “Supplement 2 to Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) (EPID L-2021-LLA-0132),” dated May 27, 2022 (ML22151A236).
4. Letter from Barstow, J., TVA, to U.S. NRC, “Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) - Supplement 3, Response to Request for Additional Information (EPID L 2021-LLA-0132),” dated July 18, 2022 (ML22200A308).
5. Letter from Barstow, J., TVA, to U.S. NRC, “Supplement 4 to Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) (EPID L-2021-LLA-0132),” dated July 28, 2022 (ML22209A238).
6. Letter from Barstow, J., TVA, to U.S. NRC, “Supplement 5 to Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) (EPID L-2021-LLA-0132),” dated September 13, 2022 (ML22266A106).
7. Letter from Barstow, J., TVA, to U.S. NRC, “Supplement 6 to Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) (EPID L-2021-LLA-0132),” dated September 29, 2022 (ML22272A054).
8. Letter from Barstow, J., TVA, to U.S. NRC, “Supplement 7 to Request for License Amendment Regarding Application of Advanced Framatome Methodologies, and Adoption of TSTF-564 Revision 2 for Browns Ferry Nuclear Plant Units 1, 2, and 3, in Support of ATRIUM 11 Fuel Use at Browns Ferry (TS-535) (EPID L-2021-LLA-0132),” dated December 9, 2022 (ML22343A092).
9. Letter from Technical Specification Task Force to U.S. NRC, “Transmittal of TSTF-564, Revision 2, ‘Safety Limit MCPR,’” dated October 24, 2018 (ML18297A361).
10. Letter from Hulvey, K. D., TVA, to U.S. NRC, “Browns Ferry Nuclear Plant, Units 1, 2, and 3, Information Transmittal for NRC Confirmatory Calculations Regarding Transition to ATRIUM 11, Fuel,” dated August 6, 2021 (ML21218A192).
11. Letter from Cusumano, V. G., U.S. NRC, to Technical Specifications Task Force, “Final Safety Evaluations of Technical Specifications Task Force Traveler TSTF-564, Revision 2, ‘Safety Limit MCPR,’ Using the Consolidated Line Item Improvement Process (CAC No. MG0161, EPID L-2017-PMP-0007),” dated November 16, 2018 (ML18299A048; Safety Evaluation ML18299A069).

12. Letter from Cusumano, V. G., U.S. NRC, to Technical Specifications Task Force, “Revised Final Safety Evaluations of Technical Specifications Task Force Traveler TSTF-564, Revision 2, ‘Safety Limit MCPR,’ Using the Consolidated Line Item Improvement Process,” dated May 13, 2020 (ML20133J816).
13. NUREG-1433, “Standard Technical Specifications, General Electric BWR/4 Plants,” Volumes 1 and 2, dated June 2021 (ML21272A357 and ML21272A358).
14. U.S. NRC Staff Requirements Memorandum, “SECY-92-223 – Resolution of Deviations Identified During the Systematic Evaluation Program,” dated September 18, 1992 (ML18100B279).
15. Letter from Rasmussen, M., TVA, to U.S. NRC, “Browns Ferry Nuclear Plant - Updated Final Safety Analysis Report, Amendment 29,” dated October 4, 2021 (ML21286A574).
16. Regulatory Guide 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” dated June 2020 (ML20055F490).
17. Draft Regulatory Guide (DG)-1327, “Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents,” dated November 2016 (ML16124A200).
18. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” dated July 2000 (ML003716792).
19. Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors,” dated May 2003 (ML031490640).
20. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition,” dated March 2007 (ML070660036).
21. NUREG-0800, Section 4.2, “Fuel System Design,” dated March 2007 (ML070740002).
22. NUREG-0800, Section 4.3, “Nuclear Design,” Revision 3, dated March 2007 (ML070740003).
23. NUREG-0800, Section 4.4, “Thermal and Hydraulic Design,” dated March 2007 (ML070550060).
24. NUREG-0800, Section 16.0, “Technical Specifications,” Revision 3, dated March 2010 (ML100351425).
25. 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, dated January 1987 (ML19292G690).
26. 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, dated June 1986 (ML20202D789 (Non-proprietary), ML19292G690 (Proprietary)).
27. ANP-10333P-A, Revision 0 “AURORA-B: An Evaluation Model for Boiling Water Reactors, Application to Control Rod Drop Accident (CRDA),” dated March 2018 (ML18208A447 (Non-Proprietary), ML18208A448 (Proprietary)).
28. ANP-10344P-A, Revision 0, “Framatome Best-estimate Enhanced Option III Methodology,” dated March 2021 (ML21131A197).
29. NEDC-33075P-A, Revision 8, “GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density,” dated November 2013 (ML13324A393).
30. ANP-10298P-A Revision 1, “ACE/ATRIUM 10XM Critical Power Correlation,” AREVA Inc., dated March 2014 (ML14183A729).
31. ANP-10335P-A Revision 0, “ACE/ATRIUM 11 Critical Power Correlation,” Framatome, dated May 2018 (ML18207A382).
32. ANP-10307PA Revision 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” dated June 2011 (ML11259A022).

33. ANP-10300P-A, Rev. 1, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios,” dated January 31, 2018 (ML18186A181).
34. EMF-2158(P)(A), Revision 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 /MICROBURN-B2,” Siemens Power Corporation, dated October 1999 (ML003698556).
35. ANF-89-98(P)(A) Revision 1 and Supplement 1, “Generic Mechanical Design Criteria for BWR Fuel Designs,” Advanced Nuclear Fuels Corporation, dated May 1995 (ML081350281 (Proprietary), ML20084V160 (Non-proprietary)).
36. EMF-93-177(P)(A), Revision 1, “Mechanical Design for BWR Fuel Channels,” Framatome NP, Inc., dated August 2005 (ML050240024 (Proprietary), ML050240022 (Non-Proprietary)), as approved by USNRC in (ML052370370).
37. EMF-93-177P-A, Revision 1, Supplement 1P-A, Revision 0, “Mechanical Design for Boiling Water Reactor Fuel Channels Supplement 1: Advanced Methods for New Channel Designs,” AREVA NP Inc., dated September 2013 (ML14198A133).
38. BAW-10247P-A, Supplement 2P-A, Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods,” Framatome Inc., dated August 2018 (ML18249A110).
39. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as incorporated by reference in 10 CFR 50.55a.
40. Huang, P. Y., Mahmood, S. T., and Adamson, R. B. “Effects of Thermomechanical Processing on In-Reactor Corrosion and Post-Irradiation Properties of Zircaloy-2,” Zirconium in the Nuclear Industry: Eleventh International Symposium, ASTM STP 1295, Bradley, E. R. and Sabol, G. P., Eds., American Society for Testing and Materials, pp. 726-757, 1996.
41. O’Donnell, W. J. and Langer, B. F., “Fatigue Design Basis for Zircaloy Components,” Nuclear Science and Engineering, Volume 20, dated January 1964.
42. BAW-10247PA, Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” AREVA NP Inc., dated April 2008 (ML081340208 (Non-Proprietary), ML081340383 and ML081340385 (Proprietary)).
43. ANP-10340P-A, Revision 0, “Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods,” Framatome Inc., dated May 2018 (ADAMS Accession No. ML18171A119 (Non-Proprietary), ML18171A120 (Proprietary)).
44. ANP-2637P, Revision 7, “Boiling Water Reactor Licensing Methodology Compendium,” Framatome Inc., dated September 2018, (ML18264A016 (Proprietary)).
45. Madni, I.K., et al., “Development of Correlations for Pressure Loss/Drop Coefficients Obtained from Flow Testing of Fuel Assemblies In Framatome ANP’S PHTF,” Paper Number 22428, Proceedings of ICONE10, Arlington, VA, dated April 14-18, 2002.
46. XN-NF-79-59(P)(A), “Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies, Exxon Nuclear Company,” dated November 1983 (ML19274C269 (Proprietary), ML20082T520 (Non-proprietary)).
47. Letter from Copeland, R. A., ANF, to Jones, R. C., NRC, “Explicit Modeling of BWR Water Rod in XCOBRA,” RAC:002:90, dated January 9, 1990.
48. Letter from Jones, R. C., NRC, to Copeland, R. A., ANF, no subject (regarding XCOBRA water rod model), dated February 1, 1990 (ML19330D160 (Non-public)).
49. EMF-CC-074(P)(A) Volume 1, “STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain”; and Volume 2, “STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain – Code Qualification Report,” Siemens Power Corporation, dated July 1994 (ML19304C452 (Proprietary), ML20072A263 (Non-proprietary)).

50. ANP-10332P-A, Revision 0, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," Framatome, dated March 2019 (ML19163A231).
51. NEDO-20566A, "General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10CFR50 Appendix K," dated September 1986 (ML092110816 (Proprietary)).
52. XN-NF-82-07(P)(A) Revision 1, Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, Exxon Nuclear Company, dated August 2018 (ML19295G649).
53. BAW-10247P-A, Revision 0 Supplement 1P-A Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," AREVA Inc., dated April 2017 (ML17206A240).
54. NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," dated March 1980 (ML053490337).
55. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Light Water Reactor Edition," dated March 2007 (ML070660036).
56. ANP-10346P-A, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA," Framatome Inc., dated October 2019 (ML20034E952).
57. Draft Regulatory Guide DG-1389, "Proposed Revision 1 to Regulatory Guide 1.183," dated April 2022 (ML21204A065).
58. EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," Siemens Power Corporation, dated September 2000 (ML003753226).
59. Technical Specifications Task Force, "Notification of an Error in Approved Traveler TSTF-564, 'Safety Limit MCPR,'" dated October 14, 2022 (ML22290A167).
60. Owen, D. B., "Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," Sandia Corporation, SCR-607, dated March 1963 (ML14031A495).
61. Piepel, G. F, and Cuta, J. M., "Statistical Concepts and Techniques for Developing, Evaluating, and Validating CHF Models and Corresponding Fuel Design Limits," SKI Technical Report, 93:46, 1993.

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Date: January 13, 2023

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 325, 348, AND 308 REGARDING APPLICATION OF ADVANCED FRAMATOME METHODOLOGIES, AND ADOPTION OF TSTF-564-A, REVISION 2, "SAFETY LIMIT MCPR," IN SUPPORT OF ATRIUM 11 FUEL USE (EPID L-2021-LLA-0132) DATED JANUARY 13, 2023

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