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

March 6, 2020

Mr. John A. Krakuszeski
Vice President
Brunswick Steam Electric Plant
Duke Energy Progress, LLC
8470 River Rd. SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 299 AND 327 TO REVISE TECHNICAL SPECIFICATION 5.6.5b TO ALLOW APPLICATION OF ADVANCED FRAMATOME ATRIUM 11 FUEL METHODOLOGIES (EPID L-2018-LLA-0273)

Dear Mr. Krakuszeski:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 299 to Renewed Facility Operating License No. DPR-71 and Amendment No. 327 to Renewed Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2. The amendments are in response to your application dated October 11, 2018, as supplemented by letters dated November 28, 2018; May 14, 2019; May 23, 2019; May 29, 2019; June 18, 2019; July 2, 2019; October 17, 2019; October 23, 2019; and December 31, 2019.

The amendments allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain.

The NRC staff has completed its review of the information provided by the licensee. Enclosure 3 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 nonproprietary version (Enclosure 4). The NRC staff will delay placing the nonproprietary SE in 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 5 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 nonproprietary SE will be made publicly available.

Enclosure 3 to this letter contains sensitive unclassified non-safeguards information.
When separated from Enclosure 3, this document is DECONTROLLED.

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

Sincerely,

/RA/

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 299 to
   License No. DPR-71
2. Amendment No. 327 to
   License No. DPR-62
3. Safety Evaluation (Proprietary Information)
4. Safety Evaluation (Nonproprietary Information)

cc: w/Enclosures 1, 2, and 4: Listserv **(6 working days after issuance of the amendments to the licensee)**

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 299
Renewed License No. DPR-71

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

    A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated October 11, 2018, as supplemented by letters dated November 28, 2018; May 14, 2019; May 23, 2019; May 29, 2019; June 18, 2019; July 2, 2019; October 17, 2019; October 23, 2019; and December 31, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 299, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. In addition, the license is amended by changes as indicated in the attachment to this license amendment, and paragraph 3 of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   Additional Conditions contained in Appendix B, as revised through Amendment No. 299, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

4. Renewed Facility Operating License No. DPR-71 is also amended by the addition of a new license condition to Appendix B, "Additional Conditions," as indicated in the attachment to this amendment, which reads as follows:

Amendment Number	Additional Conditions	Implementation Date
299	When determining the core operating limits, the Licensee shall apply the conditions outlined in the NRC's Request for Additional Information dated October 9, 2019, when applying ANP-3703P, BEO-III Analysis Methodology for Brunswick Using RAMONA5-FA, and DPC-NE-1009-P, Brunswick Nuclear Plant Implementation of Best-estimate Enhanced Option-III (i.e., Technical Specification 5.6.5.b.19 and 5.6.5.b.22, respectively).	Upon implementation of Amendment No. 299.

5. This license amendment is effective as of the date of its issuance and shall be implemented prior to start-up from the 2020 Unit 1 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachments:
Changes to the Renewed Operating License, Technical Specifications, and Appendix B, “Additional Conditions”

Date of Issuance: March 6, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 299

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace pages 6 and 10 of Renewed Facility Operating License No. DPR-71 with the attached pages 6 and 10.

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 Pages	Insert Pages
3.3-10	3.3-10
5.0-21	5.0-21
5.0-22	5.0-22

Replace the following page of the Appendix B Additional Conditions with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
---	App. B-6

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

C. This renewed 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; and is subject to all applicable provisions 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 2923 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 299, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Renewed Facility Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203, including SRs with modified acceptance criteria and SRs whose frequency of

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 299, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 1 - Technical Specifications -Appendices A and B

Date of Issuance: June 26, 2006

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)					
c. Neutron Flux—High	1	3[(c)]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 118.7% RTP
d. Inop	1,2	3[(c)]	G	SR 3.3.1.1.5 SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	NA
f. OPRM Upscale	≥ 18% RTP	3[(c)]	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	(d)
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1077 psig
4. Reactor Vessel Water Level—Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 153 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 10% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.8 psig

(continued)

(c) Each APRM channel provides inputs to both trip systems.
(d) See COLR for OPRM Confirmation Density Algorithm (CDA) setpoints.

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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

10. ANP-10333P-A, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Revision 0, March 2018.

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

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

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

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

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

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

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

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. ANP-3703P, BEO-III Analysis Methodology for Brunswick Using RAMONA5-FA, Revision 0, August 2018.

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

22. DPC-NE-1009-P, Brunswick Nuclear Plant Implementation of Best-estimate Enhanced Option-III, Revision 0, September 2018.

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

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

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

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

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Oscillation Power Range Monitor (OPRM) Report

When a report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," a report shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

(continued)

Amendment Number	Additional Conditions	Implementation Date
299	When determining the core operating limits, the Licensee shall apply the conditions outlined in the NRC's Request for Additional Information dated October 9, 2019, when applying ANP-3703P, BEO-III Analysis Methodology for Brunswick Using RAMONA5-FA, and DPC-NE-1009-P, Brunswick Nuclear Plant Implementation of Best-estimate Enhanced Option-III (i.e., Technical Specification 5.6.5.b.19 and 5.6.5.b.22, respectively).	Upon implementation of Amendment No. 299.

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 327
Renewed License No. DPR-62

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

   A. The application for amendment filed by Duke Energy Progress, LLC (the licensee), dated October 11, 2018, as supplemented by letters dated November 28, 2018; May 14, 2019; May 23, 2019; May 29, 2019; June 18, 2019; July 2, 2019; October 17, 2019; October 23, 2019; and December 31, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 327, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. In addition, the license is amended by changes as indicated in the attachment to this license amendment, and paragraph 3 of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

    Additional Conditions contained in Appendix B, as revised through Amendment No. 327, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

4. Renewed Facility Operating License No. DPR-62 is also amended by the addition of a new license condition to Appendix B, "Additional Conditions," as indicated in the attachment to this amendment, which reads as follows:

Amendment Number	Additional Conditions	Implementation Date
327	When determining the core operating limits, the Licensee shall apply the conditions outlined in the NRC's Request for Additional Information dated October 9, 2019, when applying ANP-3703P, BEO-III Analysis Methodology for Brunswick Using RAMONA5-FA, and DPC-NE-1009-P, Brunswick Nuclear Plant Implementation of Best-estimate Enhanced Option-III (i.e., Technical Specification 5.6.5.b.19 and 5.6.5.b.22, respectively).	Upon implementation of Amendment No. 327.

5. This license amendment is effective as of the date of its issuance and shall be implemented prior to startup from the 2021 Unit 2 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachments:
Changes to the Renewed Operating License, Technical Specifications, and Appendix B, “Additional Conditions”

Date of Issuance: March 6, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 327

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace pages 6 and 10 of Renewed Facility Operating License No. DPR-62 with the attached pages 6 and 10.

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 Pages	Insert Pages
3.3-10	3.3-10
5.0-21	5.0-21
5.0-22	5.0-22

Replace the following page of the Appendix B Additional Conditions with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
---	App. B-6

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except Item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

C. This renewed 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 2923 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 327, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Renewed Facility Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233. For SRs that existed prior to Amendment 233,

M. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(1) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(2) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(3) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

N. The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 327, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 2 - Technical Specifications - Appendices A and B

Date of Issuance: June 26, 2006

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)					
c. Neutron Flux—High	1	3[(c)]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 118.7% RTP
d. Inop	1,2	3[(c)]	G	SR 3.3.1.1.5 SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	NA
f. OPRM Upscale	≥ 18% RTP	3[(c)]	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	(d)
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1077 psig
4. Reactor Vessel Water Level—Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 153 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 10% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.8 psig

(continued)

(c) Each APRM channel provides inputs to both trip systems.
(d) See COLR for OPRM Confirmation Density Algorithm (CDA) setpoints.

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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

10. ANP-10333P-A, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Revision 0, March 2018.

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

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

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

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

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

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

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

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. ANP-3703P, BEO-III Analysis Methodology for Brunswick Using RAMONA5-FA, Revision 0, August 2018.

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

22. DPC-NE-1009-P, Brunswick Nuclear Plant Implementation of Best-estimate Enhanced Option-III, Revision 0, September 2018.

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

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

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

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

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Oscillation Power Range Monitor (OPRM) Report

When a report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," a report shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

(continued)

Amendment Number	Additional Conditions	Implementation Date
327	When determining the core operating limits, the Licensee shall apply the conditions outlined in the NRC's Request for Additional Information dated October 9, 2019, when applying ANP-3703P, BEO-III Analysis Methodology for Brunswick Using RAMONA5-FA, and DPC-NE-1009-P, Brunswick Nuclear Plant Implementation of Best-estimate Enhanced Option-III (i.e., Technical Specification 5.6.5.b.19 and 5.6.5.b.22, respectively).	Upon implementation of Amendment No. 327.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 299 AND 327

TO RENEWED FACILITY OPERATING LICENSES NOS. DPR-71 AND DPR-62

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

Proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390 has been redacted from this document. Redacted information is identified by blank space enclosed with boldface double brackets as shown here [[ ]].

Table of Contents

1.0 INTRODUCTION

By a letter dated October 11, 2018 (Reference 1), as supplemented by the following letters dated:

- November 28, 2018 (Reference 2),
- May 15, 2019 (Reference 3),
- May 23, 2019 (Reference 4),
- May 29, 2019 (Reference 5),
- June 18, 2019 (Reference 6),
- July 2, 2019 (Reference 7),
- October 17, 2019 (Reference 8),
- October 23, 2019 (Reference 9), and
- December 31, 2019 (Reference 10).

Duke Energy (the licensee) submitted a license amendment request (LAR) for Brunswick Steam Electric Plant (Brunswick), Units 1 and 2, to allow the application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain.

The supplements listed above provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 30, 2019 (84 FR 492).

The proprietary information in this document is marked with double brackets and bold font such as **[[Example]]**.

2.0 REGULATORY EVALUATION

Brunswick is currently using Framatome (formerly AREVA, Siemens Power Corporation, Advanced Nuclear Fuels Corporation, and Exxon Nuclear) ATRIUM 10XM fuel in the approved operating domain that includes MELLLA+ conditions. The LAR supports the transition to ATRIUM 11 fuel in the currently approved operating domain.

In Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, the NRC established its regulatory requirements related to the content of technical specifications (TSs). Section 50.36(b) of 10 CFR requires that each license authorizing the operation of a facility will include TSs and that the TSs will be derived from the safety analysis. Section 50.36(c) of 10 CFR specifies the categories that are to be included in the TSs, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

In the LAR, the licensee requests a revision to TS 5.6.5, "Core Operating Limits Report (COLR)," paragraph b, to change the references to reflect the new advanced Framatome methodologies for determining core operating limits in support of loading Framatome fuel type

ATRIUM 11. In addition, the licensee requests that Note (f) to Table 3.3.1.1-1 of the Brunswick TSs be deleted as a result of the change in analytical methods for ensuring stability.

Section 50.46 of 10 CFR established the acceptance criteria for emergency core cooling systems (ECCS) for light-water nuclear power reactors.

Section 50.62 of 10 CFR established the requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

Appendix K to 10 CFR Part 50 established the requirements for acceptable ECCS evaluation models. It also specified the documentation requirements.

The licensee stated in the LAR that the General Design Criteria (GDC) are applicable to this request. These GDC are listed in Section 3.1 of the Updated Final Safety Analysis Report (UFSAR (Reference 11)). The following GDC are applicable to this review:

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

- GDC 12, “Suppression of reactor power oscillations,” requiring 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.

- GDC 13, “Instrumentation and control,” addresses the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 15, “Reactor coolant pressure boundary,” requiring the RCS and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

- GDC 20, “Protection system functions,” requiring that the protection system shall be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC 25, “Protection system requirements for reactivity control malfunctions,” requiring that the protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

- GDC 26, “Reactivity control system redundancy and capability,” requiring two independent reactivity control systems of different design principles to be provided, one of which is capable of holding the reactor subcritical under cold conditions.

- GDC 27, “Combined reactivity control system capability,” requiring the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

- GDC 28, “Reactivity limits,” requiring the reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel (RPV) internals to impair significantly the capability to cool the core.

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

Specific regulatory requirements and standards are discussed in more detail in the subsections of the technical evaluation below.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the submittal in conjunction with the supplemental information and the responses to the NRC staff’s requests for additional information (RAIs) (Reference 3), (Reference 5), and (Reference 6)) to (1) evaluate the acceptability of the Brunswick transition to Framatome ATRIUM 11 fuel, (2) evaluate the use of the associated Framatome methodologies for licensing applications, and (3) confirm adequate technical basis for the proposed TS changes. In addition, the NRC staff held regulatory audits in February and March of 2019 to review the Brunswick-specific safety analyses and associated fuel methodologies.

3.1 Applicability of Framatome BWR Methods to Brunswick with ATRIUM 11 Fuel

Applicability of the methods is addressed in the boiling water reactor (BWR) compendium (Reference 12), which is referenced as part of Attachment 5 to the LAR (Reference 1). While the NRC staff did not separately review and approve this compendium, the NRC staff reviewed it for applicability to the use of ATRIUM 11 fuel at Brunswick. Many of the methodologies discussed in the compendium have previously been confirmed to be applicable to the ATRIUM 10XM fuel at Brunswick, and also apply to the use of ATRIUM 11 fuel. This is because the ATRIUM 11 fuel is fundamentally an evolutionary fuel design with similar geometry and composition characteristics to the ATRIUM X10 fuel. When appropriate, the applicability of methodologies to specific safety analyses is addressed in the discussion later in this safety evaluation (SE) associated with that analysis.

In order to perform evaluations of the ATRIUM-11 fuel assembly design (namely the fuel assembly mechanical design evaluation, the fuel rod thermal-mechanical evaluation, the fuel assembly thermal hydraulic evaluation, and the critical power ratio (CPR) performance evaluation), the licensee utilized specific NRC-approved methodologies in topical reports (TRs). NRC approval of these methodologies is conditional on meeting the limitations and conditions listed in the NRC staff's SE for each of these TRs. Note that much of this information is provided in Attachment 5 of the LAR (Reference 1) or the BWR compendium (Reference 12), which is referenced in the licensee's technical evaluation for the LAR (Reference 1). A discussion of how these limitations and conditions are met or dispositioned for Brunswick is provided below for each of the TRs directly supporting the ATRIUM 11 fuel assembly design evaluations. The applicability of the TRs that were already in use at Brunswick for analysis of the ATRIUM 10XM fuel assembly design that may not automatically apply to the ATRIUM 11 fuel assembly design are also discussed below.

3.1.1 ANP-10340P-A, "Incorporation of Cr-Doped Fuel Properties in AREVA-Approved Methods"

The chromia-doped fuel properties and models described in TR ANP-10340P-A, "Incorporation of Chromia Doped Fuel Properties in AREVA-Approved Methods," Revision 0, May 2018 (Reference 13), 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 12). The automated software checks are managed through the Framatome software quality assurance (QA) program, which is subject to normal NRC oversight activities as part of verifying compliance with Appendix B to 10 CFR Part 50. The TR's limitations and conditions for average fuel rod burnups is up to 62 gigawatt-days per metric ton of uranium (GWd/MTU). The NRC staff noted that the licensee currently maintains an upper design limit on average fuel rod burnup of 60 GWd/MTU at Brunswick, which is less than the maximum allowed burnup for NRC-approved use of this TR. Thus, the NRC staff finds the limitations and conditions are satisfied.

3.1.2 ANP-10336P-A, "Z4B™ Fuel Channel Irradiation Program"

TR ANP-10336P-A Revision 0, July 2017 (Reference 14), differs somewhat from other TRs listed in this section of this SE in that it is not a TR describing an evaluation methodology or acceptance criteria; rather, it describes a program to allow Framatome to expand the use of Z4B channels beyond what would normally fall within the bounds of a typical lead test assembly campaign. Most of the limitations and conditions specified in this SE apply not to specific licensees, but to Framatome's ongoing data collection and surveillance activities to identify and address any potential non-conservatism in use of existing methodologies. No credit is being taken for the apparent improvements in Z4B channel performance relative to traditional Zry-2 and Zry-4 channels. The licensee stated in its LAR that it would initially comply with the restriction on the maximum number of Z4B channels in the core being 8 percent, consistent with the limitations on the NRC-approved use of the referenced mechanical design analysis methods for Z4B channels that existed at the time of submittal. The language in the LAR allows flexibility for the licensee to eventually expand to a full core of Z4B channels once the NRC has approved loading in batch quantities. Prior to issuance of this SE, the NRC approved EMF-93-177, Revision 1, Supplement 2P-A, Revision 1 (Reference 15), which removed the aforementioned restriction on use of the mechanical design analysis methods with Z4B channels. Therefore, this restriction does not form part of the basis for the NRC approval of this LAR. The currently approved NRC uses for the referenced mechanical design analysis methods extend to

evaluation of a full complement of Z4B channels loaded with the ATRIUM 11 fuel planned for loading at Brunswick.

3.1.3 ANP-10335P-A, "ACE/ATRIUM 11 Critical Power Correlation"

To address TR ANP-10335P- A, "ACE/ATRIUM 11 Critical Power Correlation," Revision 0, May 2018 (Reference 16), Table 5-2 in Attachment 5 to the LAR (Reference 1) confirms that the bounds of applicability for the ACE/ATRIUM 11 critical power correlation (CPC) are maintained for the planned application to Brunswick. Section 4-8 of the BWR compendium (Reference 12) states that the limitation and condition associated with additive constant uncertainties for local peaking factors greater than 1.4 is directly implemented in the safety limit application for the identified uncertainties. The limitation and condition limiting use of this correlation without verification to the XCOBRA-T and AURORA-B analysis methodologies are not applicable to the ATRIUM 11 fuel assembly design evaluations, since the thermal hydraulic design evaluations were performed using XCOBRA-T. The NRC finds this disposition is acceptable.

3.1.4 BAW-10247P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for BWRs"

To address TR BAW-10247P-A, Revision 0, February 2008 (Reference 17), Section 3.2.2.3 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 13) contains a similar limitation and condition on the **[[   ]]**, which is addressed through an automated software check. Therefore, the same limitation and condition will be enforced in a consistent manner for the ATRIUM 11 fuel as well as any co-resident ATRIUM 10XM fuel. The remaining limitations and conditions are addressed by only utilizing the methodology within the bounds defined by the limitations and conditions.

There are two supplements to this TR: BAW-10247P-A, Supplement 1P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," Revision 0, April 2017 (Reference 18), and 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 19). These two supplements extended the applicability of the RODEX4 methodology to cover new materials and to incorporate new, improved models for specific phenomena.

The ATRIUM 11 fuel mechanical design evaluation, as discussed in Section 3.2.2.2 of this SE, confirms that the **[[   ]]** and that **[[   ]]**, which are two key limitations and conditions associated with the supplements. The remaining limitations and conditions are met for the ATRIUM 11 fuel assembly design, since the water 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 and hydrogen pickup model.

3.1.5 EMF-93-177P-A, "Mechanical Design for BWR Fuel Channels"

The NRC staff SE for TR EMF-93-177P-A Revision 1, August 2005 (Reference 20), and Supplement 1P-A, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced

Methods for New Channel Designs," Revision 0, September 2013 (Reference 21), specified a number of limitations and conditions that have already been shown to be met at Brunswick for the channels associated with the ATRIUM 10XM fuel. Since the ATRIUM 11 channels are the same with two exceptions, the disposition of the limitations and conditions remains applicable. The two exceptions are the use of Z4B channels, consistent with the NRC approval of EMF-93-177, Revision 1, Supplement 2P-A, Revision 1 (Reference 15), and interior milling, which is addressed through use of the Supplement 1P-A methodology (Reference 21) to this TR. The Supplement 1P-A methodology was approved with no limitations or conditions.

3.1.6 ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs"

TR ANF-89-98(P)(A) Revision 1, and Supplement 1, May 1995 (Reference 22), provides some NRC-approved generic mechanical design criteria for use with evaluation of Framatome fuel assembly designs. The ATRIUM 11 fuel mechanical design evaluation, as discussed in Section 3.2.2.2 of this SE, describes how the design criteria in this TR apply to the ATRIUM 11 fuel assembly design. The original TR's limitations and conditions are no longer applicable, since the NRC approved a revised burnup limit as part of the RODEX4 methodology (Reference 17), and the ANF correlation is no longer used.

3.1.7 XN-NF-80-19(P)(A), "Application of the ENC Methodology to BWR Reloads"

There are no limitations or conditions associated with the NRC approval of TR XN-NF-8019(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," June 1986 (Reference 23).

3.1.8 XN-NF-80-19(P)(A), "THERMEX: Thermal Limits Methodology Summary Description"

The only limitation and condition for the use of TR XN-NF-80-19(P)(A), Volume 3, Revision 2, January 1987 (Reference 24), is related to the plant monitoring system. This TR is primarily referenced for the use of the XCOBRA methodology, in order to perform the thermal hydraulic compatibility evaluation for mixed cores with ATRIUM 10XM and ATRIUM 11 fuel. Since Brunswick is utilizing a different plant monitoring system, this limitation and condition is not applicable to this LAR.

3.1.9 EMF-2158(P)(A), Evaluation and Validation of CASMO-4/MICROBURN-B2

The limitations and conditions associated with TR EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO 4/MICROBURN B2," Revision 0, October 1999 (Reference 25), were reviewed for applicability to the ATRIUM 11 fuel assembly design. In general, the limitations and conditions are intended to ensure that the methodology is not used to analyze a fuel assembly design that departs from the geometries, compositions, and conditions for which the codes were validated. While the ATRIUM 11 fuel assembly design represents an 11x11 fuel lattice, which is different from prior fuel assembly designs, the resulting geometry remain consistent with the rod dimensions and rod pitches for which the neutronics methodologies have been validated. The additional design features associated with the ATRIUM 11 fuel assembly design are either neutronically insignificant or are similar to other design features that the EMF-2158(P)(A) methodology has already been used for in licensing applications. As a result, the NRC staff finds that the limitations and conditions for this TR are met.

3.2 ATRIUM 11 Fuel Assembly Design

3.2.1 Regulatory Basis

The Framatome ATRIUM 11 fuel design was developed using the thermal-mechanical design bases and limits as outlined in ANF-89-98(P)(A) (Reference 22), compliance with which ensures the fuel design meets the regulatory requirements for fuel system damage, fuel failure, and fuel coolability criteria identified in Section 4.2 of the NRC's Standard Review Plan (SRP) (Reference 26). The SRP is intended to provide comprehensive guidance for staff review of LARs and establishes the regulatory requirements applicable to fuel designs when evaluating the safety of light-water nuclear power plants and their plant-specific Safety Analysis Reports.

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

In accordance with Section 4.2 of the SRP, the NRC staff's fuel system safety review provides assurance that:

- the fuel system is not damaged as a result of normal operation and AOOs,
- fuel system damage is never so severe as to prevent control rod insertion when it is required,
- the number of fuel rod failures is not underestimated for postulated accidents, and
- coolability is always maintained.

The NRC staff will evaluate the applicability of the Framatome methodology for the use of ATRIUM 11 fuel at Brunswick to confirm that using the methodology is within the NRC-approved ranges of applicability and to verify that the results of the analyses are in compliance with the GDC requirements specified in Appendix A to 10 CFR Part 50.

In addition, the NRC staff will verify that the licensing basis analyses for nuclear criticality safety in storage configurations when ATRIUM 11 fuel assemblies are stored in the new fuel vault or spent fuel pool are compliant with the relevant requirements of 10 CFR 50.68, "Criticality accident requirements," as approved by the NRC for application to Brunswick (Reference 27).

3.2.2 Technical Evaluation

3.2.2.1 Summary of Framatome ATRIUM 11 Fuel Assembly Design for Brunswick

Attachment 6 of the LAR (Reference 1), ANP-3686P, provides key fuel assembly design details for the Framatome ATRIUM 11 fuel assembly design planned for use at Brunswick. The fuel design is comprised of an 11 x 11 array of fuel rods with a square internal water channel that displaces a 3x3 array of rods, with **[[**

]] Table 2-1 of ANP-3686P lists the fuel assembly and component description of the ATRIUM 11 fuel assembly design.

The overall makeup of the ATRIUM 11 fuel assembly consists of **[[**

]] Further descriptions of the fuel assembly components are provided in ANP-3686P.

The NRC staff noted that most of the changes relative to the ATRIUM 10XM fuel assembly design are evolutionary changes. These changes include the use of an 11 x 11 array of fuel rods, **[[**

]], the use of chromia-doped fuel pellets, the use of non-lined stress relieved annealed (SRA) cladding, and the use of Z4B material (a proprietary zirconium alloy) for the water channel and some fuel channels. This is the first use of an 11 x 11 fuel array in reload quantities in the United States; however, the change in geometry is not expected to result in any significant change to the analysis methodologies for structural integrity. The NRC has previously reviewed and approved the use of chromia-doped fuel pellets (Reference 13) and Z4B material in-reactor cores (Reference 14). The other attributes, while novel relative to the ATRIUM 10XM, are consistent with other modern BWR fuel designs used elsewhere in the industry.

3.2.2.2 Fuel Assembly Mechanical Design Evaluation

The objectives of the fuel assembly design are to ensure that (1) the fuel assembly (system) does not fail as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, (4) fuel coolability is always maintained, (5) the mechanical design of the fuel assemblies shall be compatible with co-resident fuel and the reactor core internals, and (6) fuel assemblies shall be designed to withstand the loads from handling and shipping.

The first four objectives are discussed in Section 4.2 of the SRP, and the latter two assure the structural integrity of the fuel and compatibility with the existing reload fuel (co-resident fuel). ANP-3686P (Attachment 6 of the LAR (Reference 1)) provides the mechanical design details and fuel structural analysis results of the Framatome ATRIUM 11 fuel assembly design for use at Brunswick. This report contains only fuel structural analyses; the fuel rod evaluation is documented in ANP-3668P (Attachment 9 of the LAR (Reference 1)) and will be discussed later in Section 3.2.2.3 of this SE.

3.2.2.2.1 Stress, Strain, Loading, and Deformation Limits on Assembly Components

The American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (B&PV Code) (Reference 28) was used as guidance in establishing acceptable stress, deformation, and load limits for standard fuel assembly components and fuel channels. These limits are applied to the design and evaluation of the upper tie plate (UTP), lower tie plate (LTP), spacer grids, springs, and load chain components, as applicable. The fuel assembly structural component criteria under in-reactor accident conditions are based on Appendix F of the ASME B&PV Code, Section III, and SRP Section 4.2, Appendix A, with some criteria derived from

component tests. Outside of in-reactor accident conditions, most of the structural components are under the most limiting loading conditions during fuel handling.

For in-reactor accident conditions, the dynamic characteristics of the fuel assembly and grids were obtained from testing the assemblies for stiffness, natural frequencies, and damping values, which were then used as inputs to analytical models for the fuel assembly and fuel channel. These tests were conducted with and without a fuel channel. The test results, when compared with analysis results, have shown the dynamic response of the ATRIUM 11 fuel assembly design to be similar to other BWR fuel designs that have the same basic channel configuration and weight. The design criteria and analysis methodologies for evaluation of the fuel assembly components and channels are described in further detail in ANF-89-98(P)(A) (Reference 22) and BAW-10247P-A, Supplement 2P-A (Reference 19). Evaluations of fuel under accident loadings include mechanical fracturing of the fuel rod cladding, assembly structural integrity, and fuel assembly liftoff.

For the fuel handling accident, the primary design criteria given in the NRC-approved TR ANF-89-98(P)(A) is that the fuel assembly and load chain components must be able to withstand an axial tensile force of at least **[[**

]]

Fuel structural characteristics are not expected to be limiting for normal and AOO conditions due to the significantly smaller loads. However, if necessary, as prescribed by the methodology and plant-specific characteristics, some conditions may be evaluated to confirm that they continue to be bounded by the analyses for the accident or fuel handling scenarios. The evaluations would be performed consistent with the analysis methodologies described in ANF-89-98(P)(A) and/or the licensing basis for Brunswick.

Based on the above, the NRC staff reviewed the evaluation of the structural design of the assembly and fuel channel and finds that the fuel assembly and channel meet all primary mechanical compatibility and strength requirements for use at Brunswick based on evaluations performed using NRC-approved methodologies to demonstrate that appropriate acceptance criteria are met.

3.2.2.2.2 Fatigue and Fretting Wear

Fatigue of structural components is low because of a small number of cycles (reactor startup) or small amplitudes. The fatigue loads on the fuel channels remain under the fatigue life curve determined by O'Donnell and Langer as per Section 2.3 of ANF-89-98(P)(A) (Reference 22). While some of the fuel channels will be constructed with Z4B rather than conventional zirconium alloys, **[[**

]] The NRC has approved use of Z4B channels as a direct replacement for channels made of conventional zirconium alloys in ANP-10336P-A (Reference 14) and EMF-93-177, Revision 1, Supplement 2P-A, Revision 1 (Reference 15) . Therefore, the fatigue life curves remain applicable.

Although there is no specific wear limit for fretting, a general acceptance criterion is that fuel rod failures due to grid-to-rod fretting shall not occur. **[[**

]]. Post-test inspections of the fuel assembly showed no significant wear on fuel rods. While 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 fatigue loads, the fuel channels would continue to perform their function and not interfere with control blade insertion due to fatigue-induced distortions. Furthermore, the NRC staff finds that based on the results of the fretting wear testing, widespread rod failures would not be expected as a result of fretting effects.

3.2.2.2.3 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 consider the potential impact on thermal margins. The rod bow is calculated using an NRC-approved rod-to-rod gap closure correlation described in BAW-10247P-A, Supplement 2P-A (Reference 19), with the intent of ensuring that any impacts to thermal margins are identified and adequately dispositioned. The BAW-10247P-A, Supplement 2P-A, correlation was not explicitly approved for ATRIUM 11 fuel; however, Framatome states in ANP-3668P (Attachment 9 to the LAR (Reference 1)) that the latest experience from ATRIUM 11 lead test assembly (LTA) post-irradiation exams shows that minimal rod bow exists for exposures up to 35 GWd/MTU. This result is consistent with NRC staff expectations, given the material and geometry characteristics of the ATRIUM 11 fuel design. Since other fuel assembly designs used as the basis for the BAW-10247P-A, Supplement 2P-A, correlation showed rod bow well before this exposure, the lack of significant rod bow observed in the ATRIUM 11 LTA inspections indicates that any rod bow would be conservatively bounded by the BAW-10247P-A, Supplement 2P-A, correlation. The NRC staff expects that any rod bow detected in ATRIUM 11 fuel assemblies for burnups beyond 35 GWd/MTU would continue to be bounded by the BAW-10247P-A, Supplement 2P-A, correlation. Framatome is expected to verify this as part of its ongoing post-irradiation examinations of ATRIUM 11 fuel to support compliance with the testing and verification requirements of their 10 CFR Part 50, Appendix B, quality assurance compliance program.

The NRC staff finds the use of the BAW-10247P-A, Supplement 2P-A, correlation to address the impact of rod bowing on the thermal margins to be acceptable, based on the fact that ATRIUM 11 has been shown to exhibit less rod bowing than the fuel assembly designs used as a basis for the aforementioned correlation. Any future data suggesting that this is not the case would be addressed by Framatome as a potential Part 21 issue.

3.2.2.2.4 Axial Irradiation Growth

Rod growth, assembly growth, and fuel channel growth are calculated using correlations that were reviewed and approved by the NRC in BAW-10247P-A, Supplement 2P-A (Reference 19). In accordance with BAW-10247P-A, Supplement 2P-A, **[[**

]] The channel materials that will be used in Brunswick, Zry-4 and Z4B, are both within the scope of the NRC approval of BAW-10247P-A, Supplement 2P-A. Furthermore, the NRC considered and accepted data for the ATRIUM 11 fuel assembly design as part of the basis, applicability, and approval of the BAW-10247P-A, Supplement 2P-A, methodology.

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

3.2.2.2.5 Assembly Liftoff

The licensing basis requirements for Brunswick indicate that no fuel assembly will be allowed to levitate under normal operating, AOO, or applicable plant licensing basis accident conditions. One corollary to this stringent requirement is that if no fuel assemblies lift off, then all fuel assemblies may be assumed to be fully engaged with their fuel supports. As a result, control rod insertion will not be impaired.

The general approach adopted by the licensee was to perform an evaluation to demonstrate that the combination of hydraulic resistance, fluid momentum, buoyancy, and vertical seismic forces are not sufficient to overcome the gravitational force holding the ATRIUM 11 fuel assemblies down. This evaluation was performed **[[**
]] and the fuel assembly was confirmed to not experience liftoff. This analysis was also performed for normal conditions **[[**
]] to confirm that liftoff would not occur under such conditions.

Mixed core effects are addressed on a cycle-specific basis. Based on information from Attachments 7 and 8 of the LAR (Reference 1), the difference in thermal hydraulic characteristics between the ATRIUM 10XM and ATRIUM 11 fuel assembly designs is small. **[[**
]] Therefore, the NRC staff does not expect that the difference in hydraulic resistance between the ATRIUM 10XM and ATRIUM 11 fuel assembly designs would result in liftoff.

The NRC staff finds the liftoff evaluation acceptable because it was performed in a manner consistent with the Brunswick licensing basis, and the available data indicates that the ATRIUM 11 fuel assembly design will not result in the need for any changes to the evaluation approach previously submitted to the NRC in support of a prior LAR (Reference 29) beyond the changes in input parameters associated with the ATRIUM 11 fuel design.

3.2.2.2.6 Fuel Channel Irradiation-Induced Changes

The fuel channel was specifically evaluated for changes due to exposure to the reactor environment that may lead to loss of strength or deformation. These types of changes are

critical for the fuel channel because the fuel channel typically absorbs most of the load from seismic events and other similar design-basis events and is also the component most likely to interfere with control blade insertion. ANP-10336P-A (Reference 14) states that data shows that the Z4B material performance is bounded by the Zry-4 materials that form the basis for the existing methods used to perform these evaluations for Brunswick, which will continue to be used. **[[**

]].

The NRC staff finds this disposition of the potential changes to the fuel channel as a result of irradiation and exposure to the coolant to be acceptable because the Z4B material performance reviewed by the NRC in ANP-10336P-A is bounded by the Zry-4 material performance.
[[
]]

3.2.2.2.7 <u>Summary</u>

Tables 3-1 through 3-3 of Attachment 6 of 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 considerations of the approach used to perform the dispositions are documented in the above subsections. As a result, the NRC staff finds that the evaluations are acceptable to ensure that the mechanical design criteria for the ATRIUM 11 fuel assembly design are met for use in the Brunswick reactor core.

3.2.2.3 <u>ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation</u>

This section of this 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 the NRC-approved acceptance criteria contained in ANP-89-98(P)(A), Revision 1, and Supplement 1 (Reference 22), and the RODEX4 analysis methodology described in BAW-10247PA (Reference 17). In addition, the methodology described in ANP-10340P-A (Reference 13) 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.

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 differ from the ATRIUM 10XM 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-3668P (Attachment 9 of the LAR (Reference 1)). The fuel rod geometry and compositions fit within the applicability of the NRC-approved RODEX4 thermal-mechanical analysis methodology (Reference 17), with the addition of the chromia doped fuel properties and models reviewed and approved by the NRC (Reference 13). Therefore, the RODEX4 code was used to evaluate the fuel rod thermal-mechanical performance of the ATRIUM 11 fuel rod, as needed to demonstrate compliance with regulatory requirements.

Table 2-1 of ANP-3668P provides a summary of the findings from the fuel rod design evaluations that demonstrates 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 considerations in reviewing each acceptance criterion is provided below.

3.2.2.3.1 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 of ANP-3668P, 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 the NRC-approved mechanical design criteria in ANP-89-98(P)(A), Revision 1, and Supplement 1 (Reference 22).

3.2.2.3.2 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 (Reference 17) 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-approved fuel rod evaluation methodology in ANP-89-98(P)(A), Revision 1, and Supplement 1 (Reference 22), and therefore, is acceptable.

3.2.2.3.3 Overheating of Fuel Pellets (Fuel Centerline Temperature)

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. For each fuel rod, the melting point is adjusted to account for **[[** **]]**. RODEX4 (Reference 17) is used to determine the fuel centerline temperature for normal operating conditions and AOOs in order to establish an upper limit on the linear heat generation rate (LHGR), which ensures that no centerline melting will occur. This approach is consistent with the use of the RODEX4 methodology, and therefore, is acceptable.

3.2.2.3.4 Stress and Strain Limits

Under transient conditions, the fuel pellet expands more rapidly than the inner diameter of the cladding due to differences in their rates of change in temperature. If the inner cladding surface presses against the fuel pellet, this results in the pellet-clad interaction (PCI) phenomenon. The pressure of the fuel pellet 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 1 percent. This is consistent with the RODEX4 methodology, and the 1 percent limit for strain is consistent with the NRC-approved fuel rod evaluation methodology in ANP-89-98(P)(A), Revision 1, and Supplement 1 (Reference 22), 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 28). This is the approach prescribed in the NRC-approved mechanical design criteria in in ANP-89-98(P)(A), Revision 1, and Supplement 1 (Reference 22), and therefore, is acceptable.

3.2.2.3.5 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 (Reference 17). The NRC has reviewed and approved the models used in RODEX4 to address these phenomena and the methodology is applicable to Brunswick as discussed in Section 3.2.2.3; therefore, this is an acceptable disposition.

3.2.2.3.6 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, and **[[**

]]

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-approved fuel rod evaluation methodology in BAW-10247PA (Reference 17), and the evaluation is performed with a combination of an NRC-approved fuel rod analysis methodology with applicable data, the NRC staff finds this acceptable.

3.2.2.3.7 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-approved RODEX4 fuel rod evaluation methodology in

BAW-10247PA (Reference 17), 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 Brunswick is discussed below.

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

- BAW-10247PA (Reference 17) discusses what constitutes "abnormal crud" and how to capture the effect by the use of 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. Any abnormal increase in crud would be addressed by increasing the crud assumed in the RODEX4 calculations based on plant-specific analyses. A similar approach would be used to address abnormal corrosion. However, no such observations have been made at Brunswick. The cladding properties for the ATRIUM 11 fuel assembly design do not differ from the ATRIUM 10XM fuel assembly design, so no change is expected as a result of transitioning to ATRIUM 11 fuel.

- In a previous license amendment request for Brunswick (Reference 29) the NRC approved an upper limit on the calculated peak oxide thickness such that sufficient margin exists to accommodate the effect of non-uniform corrosion such as localized hydride formations. The ATRIUM 11 fuel assembly design utilizes the same cladding material. Therefore, the approved criteria apply to the ATRIUM 11 fuel design.

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

3.2.2.3.8 Rod Internal Pressure

The fuel rod internal pressure is calculated using the RODEX4 code and methodology (Reference 17). 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 22). The NRC staff finds this approach to be acceptable since it is based on a methodology and acceptance criteria that the NRC has previously reviewed and approved.

3.2.2.3.9 Summary

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 22) and BAW-10247PA (Reference 17), in the fuel rod thermal-mechanical analyses for the Framatome ATRIUM 11 fuel design for use at Brunswick. The NRC staff finds that the fuel design criteria have been satisfied and provide reasonable assurance for safe operation at Brunswick.

3.2.2.4 Thermal Hydraulic Design of ATRIUM 11 Fuel Assemblies for Brunswick

This section describes the NRC staff's evaluation of the Brunswick thermal-hydraulic analyses to demonstrate the hydraulic compatibility of the ATRIUM 11 fuel with the co-resident ATRIUM 10XM fuel. Duke Energy is proposing to transition from the current ATRIUM 10XM fuel design to Framatome ATRIUM 11 fuel starting with Unit 1, Cycle 23 (i.e., spring of 2020). Attachments 7 and 8 of the LAR (Reference 1) (for Units 1 and 2, respectively) provide the results of the thermal-hydraulic analyses to show ATRIUM 11 fuel is hydraulically compatible with the co-resident ATRIUM 10XM fuel. The results from the thermal-hydraulic analysis are compared to acceptance criteria established in the NRC-approved TRs ANF-89-98(P)(A), Revision 1, Supplement 1 (Reference 22), and XN-NF-80-19(P)(A), Volume 4, Revision 1 (Reference 24).

The thermal-hydraulic analyses were performed to verify that the design criteria were satisfied and further establish thermal operating limits with acceptable margins of safety during normal reactor operation and AOOs. Due to reactor and cycle operating differences, many of the analyses supporting these thermal-hydraulic operating limits were performed on a non-cycle-specific and cycle-specific basis and are documented in plant- and cycle-specific reports. Table 3.1 of both ANP-3643NP and ANP-3644NP (Attachments 7b and 8b to (Reference 1)) lists the applicable thermal-hydraulic design criteria, analyses, and results for hydraulic compatibility, thermal margin performance, fuel centerline temperature, rod bow, bypass flow, stability, LOCA analysis, CRDA analysis, ASME overpressurization analysis, and seismic/LOCA liftoff for ATRIUM 11 fuel. The subsections below summarize the results from selected design criteria and analyses results.

3.2.2.4.1 Hydraulic Characterization

Basic dimension parameters for the ATRIUM 10XM and ATRIUM 11 fuel assembly designs are summarized in Table 3.2 of both ANP-3643NP and ANP-3644NP (Attachments 7b and 8b to (Reference 1)). Table 3.3 of the same references provides a comparison of key hydraulic characteristics, including loss coefficients, flow resistances, and friction factors for the two fuel assembly designs. A summary of the testing and analysis performed to determine the hydraulic characteristics for the fuel assembly designs is included in Section 3.1 of ANP-3643P and ANP-3644P.

The testing and analysis approaches used for the ATRIUM 11 fuel assembly design have previously been reviewed and approved for use to characterize the hydraulic characteristics of the ATRIUM 10XM fuel assembly design for other plants operating in the extended flow window (EFW) domain. One such example is the Monticello adoption of Framatome methods for the EFW domain (Reference 30), which covers similar operating conditions to those expected at Brunswick. There are no attributes associated with the ATRIUM 11 fuel assembly design that would be expected to require special treatment relative to the ATRIUM 10XM fuel assembly design. Therefore, the NRC staff finds the hydraulic characterization of the ATRIUM 11 fuel assembly design to be acceptable.

3.2.2.4.2 Thermal-Hydraulic Compatibility

The thermal-hydraulic compatibility analyses were performed in accordance with the NRC-approved Framatome thermal hydraulic methodology for BWRs utilizing XCOBRA

(Reference 24). The XCOBRA code predicts the steady-state thermal hydraulic performance of fuel assemblies in BWR cores at various operating conditions and power distributions. The thermal-hydraulic compatibility analysis evaluates the relative thermal performance of the ATRIUM 10XM and ATRIUM 11 fuel assembly designs that will be inserted in the Brunswick core. The analyses were performed for full-core and mixed-core configurations.

The hydraulic compatibility analysis **[[**

]] This analysis is performed utilizing different typical axial power shapes and radial power factors for rated and off-rated conditions. The input conditions used for the analysis are listed in Table 3.4 of both ANP-3643P and ANP-3644P (Attachments 7b and 8b to (Reference 1)), while representative results are given in Tables 3.5 through 3.8 and Figures 3.1 and 3.2. **[[**

]] Thermal hydraulic compatibility is obtained when the following parameters do not change significantly throughout the transition from a full complement of ATRIUM 10XM fuel to a full complement of ATRIUM 11 fuel: **[[**

]]

The performance characteristics important for safety analysis purposes are captured by the CHF correlations and thermal hydraulic specifications unique to each fuel assembly design that are used with the methodologies used to analyze the thermal limit margins.

Based on the changes in pressure drop, bypass flow, and assembly flow caused by the transition from ATRIUM 10XM fuel to ATRIUM 11 fuel, the NRC staff finds that the hydraulic compatibility analyses for the transition cores at Brunswick provide reasonable assurance that the resident and co-resident fuel designs will satisfy the thermal-hydraulic design criteria for mixed cores.

3.2.2.4.3 Thermal Margin Performance

The thermal margin analyses were performed using the NRC-approved thermal-hydraulic methodology for steady state CPR evaluations with XCOBRA listed in the Brunswick TSs, XN-NF-80-19(P)(A), Volume 4, Revision 1 (Reference 24). Empirical correlations from ANP-10298-NP-A, Revision 1 (Reference 31), and ANP-10335NP-A, Revision 0 (Reference 16), for the ATRIUM 10XM and ATRIUM 11 fuel assembly designs, respectively, are used based on results of boiling transition test programs. The CPR correlations are discussed in Section 3.2.2.7 of this SE and account for the assembly design features that are different between the two fuel designs through modification of the K-factor term in the CPR correlations.

The hydraulic compatibility analysis discussed in the previous subsection included steady-state CPR values calculated for various radial peaking factors. As expected, **[[**

]] Therefore, there is no significant impact on the thermal margin performance for either fuel assembly design as a

result of mixed core operations. Since the fuel assembly design-specific considerations are addressed by use of fuel assembly design-specific CPR correlations, appropriate thermal margins will be maintained through use of appropriate operating limits on design and operation of the cores throughout the transition, as controlled by the SLMCPR and OLMCPR values in the TSs and COLR.

Based on the above, the NRC staff finds that the introduction of ATRIUM 11 fuel will not cause an adverse impact on thermal margin for the co-resident ATRIUM 10XM fuel.

3.2.2.4.4 Rod Bow

Rod bow is addressed as part of the mechanical design analyses (see Section 3.2.2.2 of this SE for further discussion). **[[**

]]

The NRC staff finds this disposition to be acceptable based on the fact that this is consistent with the NRC-approved Framatome methodologies in BAW-10247PA (Reference 17) and the impact is appropriately dispositioned.

3.2.2.4.5 Bypass Flow

As discussed earlier in this section of this SE, **[[**

]]

Based on the above, the NRC staff finds that adequate bypass flow will be available with the introduction of the ATRIUM 11 fuel design and that applicable design criteria will be met.

3.2.2.4.6 Summary

The NRC staff reviewed the thermal hydraulic compatibility analytical approaches and results intended to demonstrate that the ATRIUM 11 fuel design is hydraulically compatible with the ATRIUM 10XM fuel currently used at Brunswick. The NRC staff determined that the generic thermal-hydraulic design criteria as approved by the NRC in ANF-89-98(P)(A) (Reference 22) have been used in the analyses. The NRC staff finds that although the ATRIUM 10XM and ATRIUM 11 fuel assemblies contain a number of differences in their geometric and hydraulic characteristics, they remain hydraulically compatible.

3.2.2.5 Stability

The thermal-hydraulic design criteria approved by the NRC in ANF-89-98(P)(A) (Reference 22) includes 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. This evaluation is performed using the STAIF code as prescribed in ANF-89-98(P)(A), and the results are documented in ANP-3643P and ANP-3644P (Attachments 7b and 8b to (Reference 1)) for Brunswick, Units 1

and 2. This evaluation demonstrates that the requirements within the NRC-approved generic fuel assembly mechanical design criteria used by Framatome to qualify new fuel designs are met. However, the NRC staff did not review the STAIF evaluation in detail because the Confirmation Density Algorithm (CDA)-based hardware trip is expected to detect and suppress any power oscillations resulting from stability issues, as confirmed through the use of the BEO-III analytical methodology discussed in Section 3.6 of this SE. Additionally, the fact that the ATRIUM-11 fuel assembly design does not represent a significant departure from prior fuel assembly designs provides assurance that the assumptions made in the stability analyses remain valid. This ensures that the regulatory requirements associated with stability performance are met.

3.2.2.6 Brunswick Fuel Transition – Equilibrium Fuel Cycle Design

ANP-3661P (Attachment 10 of the LAR (Reference 1)) summarizes the equilibrium core design and fuel management calculations for a representative full core of ATRIUM 11 fuel loaded at Brunswick. 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 25).

The equilibrium core design is not intended to reflect the actual nuclear design of fuel assemblies or a loading pattern for use at Brunswick. 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 Brunswick. 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 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. Any change would be evaluated by 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 ratio (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 Brunswick, 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 to be 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 its intended uses.

3.2.2.7 Critical Power Correlation (CPC) for ATRIUM 11 Fuel

The CPR values for ATRIUM 11 are calculated with the ACE/ATRIUM 11 CPC (Reference 16) and the CPR values for ATRIUM 10XM are calculated with the ACE/ATRIUM 10XM CPC (Reference 31). Both CPCs were reviewed and approved by the NRC, as described in the referenced TRs. Section 3.1.3 of this SE discusses the applicability of the ATRIUM 11 CPC TR (Reference 16) and its limitations and conditions for the use of ATRIUM 11 fuel at Brunswick.

Based on the above, the NRC staff finds that the licensee will utilize NRC-approved CPCs to generate CPR values for both ATRIUM 11 and ATRIUM 10XM fuel at Brunswick during and after the transition within the bounds of the limitations and conditions specified in the SEs for the referenced TRs. Therefore, the licensee's use of the CPCs is acceptable.

3.2.2.8 Spent Fuel Storage for ATRIUM 11 Fuel

The licensee did not explicitly address the nuclear criticality safety analyses performed to qualify the ATRIUM 11 fuel assembly design for storage in the Brunswick new fuel vault and spent fuel pool. However, The NRC staff reviewed and approved a nuclear criticality safety analysis methodology for Brunswick for the ATRIUM-10 Framatome fuel assembly design (Reference 27). A review of this methodology indicated that no new elements are necessary to address the ATRIUM 11 fuel assembly design, and no TS changes are necessary. Even though this did not form part of the technical basis for a safety finding, the NRC staff confirmed during the regulatory audit supporting this LAR review (Reference 32) that the licensee performed updated nuclear criticality safety analyses consistent with its previously approved methodology, utilizing the ATRIUM 11 fuel assembly design-specific geometry, composition, and manufacturing tolerances.

The NRC staff finds the licensee's approach to utilize its current approved methodology to perform nuclear criticality safety analyses qualifying Brunswick for storage of ATRIUM 11 fuel assemblies per 10 CFR 50.68 and to update its licensing basis without including the nuclear criticality safety analyses in an LAR to be acceptable based on the fact that no new features are incorporated in the ATRIUM 11 fuel assembly design that would necessitate use of new methods in addition to what has previously been reviewed and approved by the NRC.

3.2.3 Fuel Assembly Design Conclusion

The NRC staff reviewed the information in the licensee's submittal pertaining to the ATRIUM 11 fuel assembly design (Reference 1). The NRC staff's review was further supported by a regulatory audit (Reference 31), which was used to confirm information included in docketed submittals. As summarized in the above subsections, the NRC staff finds the licensee's disposition of the fuel assembly design-related impacts to the safe operation of Brunswick to be acceptable.

3.3 AOOs/ATWS

3.3.1 Regulatory Evaluation

In accordance with 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," licensees are required to provide the means to address an ATWS event. An AOO, followed by the failure of the reactor trip portion of the protection system is defined in GDC 20 of 10 CFR Part 50, Appendix A.

3.3.2 Technical Evaluation

The licensee submitted information in ANP-3705P and ANP-3702P as Attachments 5 and 12, respectively, of the LAR (Reference 1) in conjunction with the supplemental information and the responses to the NRC staff's RAIs (Reference 3), (Reference 5), and (Reference 6). The purpose is to show the applicability of the approved AURORA-B AOO methodology (Reference 33) for Brunswick, in particular, compliance with the limitations and conditions imposed for application of the AURORA-B AOO TR. It also provided a demonstration analysis of select licensing basis events using the AURORA-B AOO methodology to demonstrate that the results of the analyses meet the applicable acceptance criteria.

3.3.2.1 AURORA-B AOO Methodology Overview

The AURORA-B AOO methodology and the NRC staff's SE of the methodology is found in ANP-10300P-A, Revision 1 (Reference 33). The methodology is used to evaluation transients, postulated accidents, and beyond design-basis scenarios for BWRs. The methodology is built upon the following three computer codes:

- S-RELAP5 provides the thermal-hydraulic code to simulate BWR system response,
- MB2-K provides the core neutronic response, and
- RODEX4 provides the thermal-mechanical response of the individual fuel rods.

The methodology uses non-parametric order statistics to evaluate the impact of uncertainties in the methodology. This means that for each scenario analyzed, a number of runs are executed (e.g., 59 runs), varying certain parameters to achieve a result at a certain confidence level. In the case for 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 (Reference 33) contains the uncertainty parameters used for the uncertainty analysis.

The licensee provided a demonstration analysis in ANP-3702P (Attachment 12 of the LAR (Reference 1)). The demonstration analysis provided analyses for the following transients, accidents, and beyond design-basis events: load rejection no bypass, turbine trip no bypass, feedwater controller failure, ASME overpressurization analysis, and ATWS overpressurization analysis.

3.3.2.2 Applicability of the AURORA-B AOO Methodology to Brunswick

The NRC staff reviewed the LAR to ensure that the AURORA-B AOO methodology was applicable to Brunswick. As described in Section 3.1 of the SE for the AURORA-B AOO

methodology (Reference 33), the methodology is applicable, in part, to BWR/3 through BWR/6 plants. Since Brunswick is a BWR/4 plant, the methodology is applicable to Brunswick. The NRC staff considered three additional major considerations to determine the applicability of the methodology to Brunswick: (1) applicability for use with ATRIUM 10XM fuel, (2) applicability for use with ATRIUM 11 fuel, and (3) applicability for use in the MELLLA+ operating domain. Note that the LAR and AURORA-B AOO methodology use MELLLA+ and the EFW interchangeably.

Upon initial implementation of the AURORA-B AOO methodology, the Brunswick core will still contain ATRIUM 10XM fuel. Therefore, the NRC staff considered the applicability of the AURORA-B AOO methodology to this fuel design. In general, the AURORA-B AOO methodology was developed around the ATRIUM 10 and ATRIUM 10XM fuel bundle design (see Section 3.3.1 of the SE for the AURORA-B AOO methodology). Also, as implied in Limitations 4 and 5 in Section 5.0 of the SE for the AURORA-B AOO methodology, ATRIUM 10 and ATRIUM 10XM are not new fuel designs relative to the AURORA-B AOO methodology and need not be explicitly justified for use with the method. Brunswick is operating with the ATRIUM 10XM fuel within the fuel design limits. Since the AURORA-B AOO methodology was developed based on the ATRIUM 10 and ATRIUM 10XM fuel design, and Brunswick is operating the ATRIUM 10XM within its approved design, the NRC staff finds that the AURORA-B AOO methodology is applicable to Brunswick with ATRIUM 10XM fuel.

As described in Limitations 4 and 5 in Section 5.0 of the SE for the AURORA-B AOO methodology, an applicant is required to justify new fuel designs relative to those approved for use in the AURORA-B AOO methodology. ATRIUM 11 is a new fuel design for use with the AURORA-B AOO methodology. The licensee provided justification in the ANP-3705P attachment of the LAR. Specifically, the licensee provided justification for ATRIUM 11 with respect to transients and accidents in Section 6.0 of ANP-3705P and ATWS in Section 7.0 of ANP-3705P. The major concern for the transients and accidents is how the void prediction uncertainties are incorporated into the analyses. These uncertainties are important because they could impact the results of the analyses (e.g., minimum critical power ratio (MCPR)). Note that it is also important for the licensee to use models that can accurately predict the void fraction it is using. For Brunswick, the licensee stated it will be using the **[[ ]]** void correlation for the ATRIUM 11 fuel. This correlation is discussed in the MELLLA+ submittal (Reference 34).

As described in the LAR, the licensee stated that these uncertainties were not explicitly included in the transient and accident analyses. Rather, they are implicitly included in the power prediction, and the uncertainties in the power prediction are included in the analysis to determine the safety limit critical power ratio (SLMCPR). Brunswick uses the SAFLIM3D methodology (Reference 35). The NRC staff confirmed that the power prediction was incorporated into the SAFLIM3D methodology. Additionally, the NRC staff confirmed that the Brunswick methodology used to calculate the power prediction, MICROBURN-B2 (Reference 36), incorporated the void-quality correlation. Since the licensee incorporates the void fraction uncertainty in the power prediction, and the power prediction uncertainty is included in the calculation of the SLCMPR, the NRC staff finds the licensee appropriately addressed the ATRIUM 11 fuel for SLMCPR.

The LAR describes how the void prediction uncertainty is incorporated into the delta critical power ratio (ΔCPR) as a result of a transient that is used to determine the operating limit

minimum critical power ratio (OLMCPR).[1] The licensee discusses how the void prediction uncertainty can be implicitly accounted for by conservatism in the computer code models and input parameters used for the analysis. The conservatism in the computer codes exists because it is tuned to bound the power increases relative to the benchmark tests. The uncertainty in the void prediction uncertainty will impact the uncertainty in the power prediction (which has a direct influence on ΔCPR). Since the computer codes are tuned to bound the power predictions in the benchmark tests, they will inherently incorporate the void prediction uncertainty. The licensee also stated conservative input parameters (for sampled uncertainty parameters (see Table 2.2 of ANP-3702)) are used for the transient analysis to account for void prediction uncertainty. Since the void prediction is inherently accounted for in the transient analysis to determine ΔCPR, and the initial conditions are conservatively biased, the NRC staff finds the licensee has adequately addressed the ATRIUM 11 fuel for ΔCPR.

The licensee intends to use the AURORA-B AOO methodology, which is approved to analyze ATWS events, with the exception of ATWS-I. In Section 7.1 of ANP-3705P (Attachment 5 of the LAR), the licensee justifies that the ATWS vessel overpressurization event in the AURORA-B AOO code suite is not impacted by the ACE/ATRIUM 11 CPC that was approved for ATRIUM 11 fuel. The justification provided is that the AURORA-B AOO methodology ignores dryout (and therefore, does not need to use a CPC) in the ATWS vessel overpressurization event because it is more conservative to assume maximum heat transfer to the coolant for an overpressure event. The NRC staff finds that this justification is reasonable because maximizing heat transfer to the coolant will increase the pressure in the vessel, which is appropriate for analyzing an overpressure event. The NRC staff also finds that ignoring the dryout in the fuel is conservative because once the fuel is in dryout, heat transfer from the rod to the coolant is diminished and heat transfer to the coolant would, therefore, be reduced.

The licensee also discussed the void-quality correlation's impact on the ATWS vessel overpressure analysis. Similar to the transient and accident discussion above, the licensee justified that the void prediction uncertainties are inherently incorporated into the code and the input parameters are conservatively biased to account for uncertainties. Therefore, the NRC staff finds the void prediction uncertainties are appropriately accounted for in the ATWS methodology. Note that for ATWS analyses, the void correlation is more important for predicting peak vessel pressure. For Brunswick, the licensee stated it will be using the **[[ ]]** void correlation for the ATRIUM 11 fuel.

Section 7.3 of ANP-3705P (Attachment 5 of the LAR) contains an evaluation of the ATWS containment heatup calculation. The licensee provided justification that **[[ ]]**. The ATWS containment heatup evaluation is discussed in Section 3.3.2.5 of this SE.

The final major applicability consideration is the use of the methodology in the MELLLA+ operating domain. Brunswick was approved to operate in the MELLLA+ operating domain in License Amendment Nos. 285 and 313 Brunswick Units 1 and 2, respectively (Reference 37). In the demonstration analysis in the ANP-3702P attachment of the LAR, the licensee analyzed some of the events in extended power uprate conditions and MELLLA+ conditions. In the SE for the AURORA-B AOO methodology, the NRC staff considered the applicability of the MELLLA+ operating domain. In its review, the NRC staff determined that the methodology was

[1] OLMCPR is calculated as the sum of the SLMCPR and the ΔCPR. Brunswick operates above the OLMCPR to ensure that an AOO does not cause the plant to violate the SLMCPR.

acceptable to use in the MELLLA+ domain. However, Limitations and Conditions 4 and 5 in Section 5 of the AURORA-B methodology states that there must be justification to use the void-quality correlation for new fuel at extended power uprate and EFW conditions. For Brunswick, the licensee stated it will be using the **[[ ]]** void correlation for the ATRIUM 11 fuel. Since the NRC staff determined that the void-quality correlation is acceptable to use with ATRIUM 11 fuel, and the correlation was approved for use in the MELLLA+ operating domain, the NRC staff finds that the AURORA-B methodology is acceptable for Brunswick to use in the MELLLA+ operating domain.

3.3.2.3 <u>AURORA-B Methodology Limitations and Conditions</u>

The AURORA-B AOO methodology contains 26 limitations and conditions in Section 5.0 of the NRC staff's SE (ANP-10300P-A, Revision 1 (Reference 33)). As described on page 7 of the LAR (Reference 1), the licensee stated that ANP-3705P (Attachment 5 of the LAR) demonstrates that the Framatome licensing methodologies presented in ANP-2637P are applicable to the ATRIUM 11 fuel type and operation of Brunswick in the currently approved operating domain. The licensee further stated that the limitations and conditions for the Framatome TRs are included in ANP-2637P (Reference 12), and compliance with the limitations and conditions is assured by implementing them within the engineering guidelines or by incorporating them into the computer codes. Discussion of the limitations and conditions for the AURORA-B AOO methodology is found starting on page 5-32 of ANP-2637P (Reference 12).

Note that Limitations and Conditions 20 through 26 in Section 5.2 of the SE for the AURORA-B AOO methodology are related to the change process of the methodology itself. The licensee is requesting AURORA-B AOO methodology as approved; therefore, these limitations are not applicable to this Brunswick LAR.

Limitation and Condition 1 relates to using the method's coupled calculational devices (CCD) within their approved range. The CCDs used for this analysis are RELAP5, MB2-K, MICROBURN-B2, and RODEX4. The NRC staff confirmed that these values are within with approved ranges.

Limitation and Condition 2 relates to the cladding oxidation limit (13 percent) when using the Cathcart-Pawal oxidation correlation. The NRC staff confirmed that the AURORA-B AOO results meet this limit.

Limitation and Condition 3 relates to using the approved uncertainty distributions in the analysis. The NRC staff confirmed that the generic uncertainty distributions presented in Table 2.2 of ANP-3702P are consistent with those in Table 3.6 of the SE for the AURORA-B methodology. For the **[[ ]]**, the licensee stated 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.

Limitation and Condition 4 relates to the justification of void fraction prediction for new fuel designs. The licensee discussed the void fraction prediction in Section 6.1 of ANP-3705P. The NRC staff reviewed the void fraction prediction in Section 3.3.2.2 of this SE and finds it was acceptable. Therefore, the NRC staff finds that the licensee adequately addressed this limitation.

Limitation and Condition 5 relates to the justification of the **[[ ]]** void-quality correlation for new fuel designs. The licensee discussed the void-quality correlation in Section 5.1 of ANP-3705P. The NRC staff reviewed this in Section 3.3.2.2. of this SE and finds it was acceptable. Therefore, the NRC staff finds that the licensee adequately addressed this limitation.

Limitation and Condition 6 relates to the use of the **[[ ]]** The licensee stated it followed the approved process of Sections 3.6.4.10 and 3.6.4.13 for **[[ ]]** of the methodology to determine the uncertainty range. Therefore, the NRC staff finds that the licensee adequately addressed this limitation.

Limitation and Condition 7 relates to the licensee providing justification for the key plant parameters and initial conditions selected for performing sensitivity analysis on an event-specific basis. The licensee described compliance with this requirement in the reload safety analysis report (RSAR (Reference 9)).

Limitation and Condition 8 relates to the truncation of sampling ranges for uncertainty distributions used in the non-parametric order statistics analyses. The licensee discussed in Section 2.2.2 of ANP-3702P (Attachment 12 of the LAR (Reference 1) how the sampling performed complies with the requirements of the SE for ANP-10300P-A, Revision 1 (Reference 33). The NRC staff confirmed that the licensee adequately addressed this limitation.

Limitation and Condition 9 relates to uncertainties of medium or highly ranked PIRT phenomena that are not addressed in given non-parametric order statistics analysis via sampling. To meet this limitation, AREVA modeled the phenomena as described in Tables 3.2 and 3.4 of the SE for ANP-10300P-A, Revision 1 (Reference 33). The NRC staff confirmed the licensee complied with the requirements of the tables, and therefore, has adequately addressed this limitation and condition.

Limitation and Condition 10 relates to the assumptions of **[[ ]].** The licensee stated it complied with the requirements of Tables 3.2 and 3.4 of the SE for ANP-10300P-A, Revision 1 (Reference 33), as they relate to this limitation. The NRC staff confirmed the licensee complied with the requirements of the tables, and therefore, has adequately addressed this limitation and condition.

Limitation and Condition 11 relates to justification for uncertainties used for highly ranked plant-specific PIRT parameters. The licensee described compliance with this requirement in the RSAR (Reference 9).

Limitation and Condition 12 relates to plant-specific changes to AURORA-B to enhance **[[ ]]** when applying the AURORA-B EM to the **[[ ]]**. The Brunswick UFSAR evaluates those events as non-limiting, and as such, they are not analyzed on a cycle-specific basis. The licensee stated that it is not making plant-specific changes, and if it should in the future, the licensee will request

NRC review and approval. Therefore, the NRC staff finds that the licensee adequately addressed this limitation.

Limitation and Condition 13 relates to the use of nominal calculations with the AURORA-B evaluation model. The events in this category are generally expected to be benign, and hence, non-limiting. The licensee dispositions events in this category as non-limiting in its UFSAR; therefore, no additional evaluation is required. The NRC staff finds that the licensee adequately addressed this limitation.

Limitation and Condition 14 relates to the scope of approval for AURORA-B. The approval does not include the ABWR. Since Brunswick is not an ABWR, its use is within scope. Therefore, the NRC staff finds that the licensee adequately addressed this limitation.

Limitation and Condition 15 relates to the application of AURORA-B to BWR/2s at EPU or EFW conditions. Brunswick is not a BWR/2, therefore, this limitation is not applicable.

Limitation and Condition 16 relates to the justification of a plant-specific conservative flow rate. The licensee described compliance with this requirement in the RSAR (Reference 9).

Limitation and Condition 17 relates to the uncertainty associated with heat transfer predictions in the film boiling regime. The licensee stated that no film boiling was encountered in the AOO analyses. Therefore, the NRC staff finds that the licensee adequately addressed this limitation.

Limitation and Condition 18 relates to using conservative measures with the justification for the method of determining and applying conservative measures in future deterministic analyses for each figure of merit (FoM) and re-performance of full statistical analysis if a scenario exceeds a 1σ magnitude difference. The licensee described compliance with this requirement in the RSAR (Reference 9)).

Limitation and Condition 19 relates to stipulations that would satisfy the 95/95 criterion for figures of merit calculated by AREVA in accordance with ANP-10300P-A. The licensee stated that all calculations completed in its demonstration analysis comply with the restrictions of Limitation and Condition 19. Therefore, the NRC staff finds that the licensee adequately addressed this limitation.

The NRC staff reviewed each limitation and finds that they were adequately addressed by the licensee for the demonstration case and supported by the RSAR (Reference 9).

3.3.2.4 <u>AURORA-B Methodology Analysis Results</u>

The plant-specific UFSAR for Brunswick (Reference 11) contains the design-basis analyses to evaluate the effects of a wide range of AOOs. Since these analyses are performed on a cycle and core configuration-specific basis during the standard reload analyses, the licensee provided demonstration analyses of the potentially limiting events.

Since the licensee's analysis in the LAR is a demonstration analysis, the NRC staff's review is to focus on ensuring the licensee can adequately evaluate AOOs with the new AURORA-B AOO methodology and ATRIUM 11 fuel. The NRC staff reviewed this section to ensure the

potentially limiting events are identified and considered for explicit analysis, the AOO results are realistic, and the results meet specified acceptable fuel design limits (SAFDLs).

In the LAR, the licensee provided demonstration analyses for the load rejection no bypass event, the turbine trip without bypass event, and the feedwater controller failure event. Since this is the first time ATRIUM 11 has been implemented and the first time the AURORA-B AOO methodology has been implemented, the NRC staff questioned how each event was to be dispositioned since only a subset of the analysis was provided.

For each cycle, the minimum set of analyses required to license the cycle is determined based on the disposition of events and operational flexibility needed such as equipment out of service and exposure windows. **[[**

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Additionally, the NRC staff reviewed the RSAR (Reference 9) for Brunswick, Unit 1 Cycle 23. The cycle-specific results in the RSAR confirmed all limits were met for the full range of operating conditions.

To ensure there is appropriate coverage of the parameters used in the uncertainty analysis and to ensure there are no significant trends with respect to the uncertainty parameters in the results, the NRC requested additional information in RAI 10. Specifically, the NRC staff requested to review the following data set for the load rejection with no bypass event at 100 percent power/104.5 percent flow and MSIV closure event at 100 percent power and 85 percent flow:

- the sampled values of the uncertainty parameters for all cases executed, and
- the FoM results for all cases executed.

The licensee's RAI response (Reference 6) showed that the Brunswick implementation of the AURORA-B AOO methodology is sufficient to meet the GDC 10 and ATWS acceptance criteria. The NRC staff reviewed the analysis approach for the initial transition to AURORA-B AOO methods and finds that the approach covers the full range of operating conditions and it is therefore acceptable.

3.3.2.5 <u>ATWS Containment Heatup</u>

Changes in fuel design can impact the power and pressure excursions during an ATWS event. The power and pressure excursion changes can impact the suppression pool and containment temperature and pressure responses.

[[

]] In NRC RAI 31 (Reference 38), the NRC staff asked the licensee to describe the analysis done to justify **[[** **]]** of ATRIUM 11. In its response (Reference 6), the licensee stated that it completed a **[[**

]]

Additionally, the NRC staff requested in RAI 31 that the licensee confirm the fuel transition is bounded by the current analysis of record and the quantitative results for containment pressure and suppression pool temperature response. In its response, the licensee stated, "The current licensing basis for Brunswick ATWS containment shows the peak suppression pool temperature for MELLLA+ was 174 °F and the peak containment pressure was 8.4 psig." The analysis is based on **[[** **]]** After this was completed, the licensee determined the **[[**

]]

Finally, in NRC RAI 31, the NRC staff requested a quantitative comparison of the decay heat because containment heatup is directly impacted by the stored energy in the fuel and decay heat. In its response, the licensee provided a table, which compared **[[**

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Therefore, the NRC staff finds that the analysis of record remains bounding for ATWS containment heatup with the transition to ATRIUM 11 at Brunswick such that GDC 16, 38, and 50 continue to be met.

3.3.3 AOO/ATWS Evaluation Conclusion

The NRC staff reviewed the information in the licensee's submittals pertaining to the analysis of AOO and ATWS events for Brunswick, Units 1 and 2, including the original submittal (Reference 1), as well as relevant responses to requests for additional information (Reference 6). The NRC staff's review was further supported by a regulatory audit (Reference 32), which was used to confirm information included in docketed submittals. Based upon its review, as documented above, the NRC staff has finds that:

(1) The licensee has proposed to implement the AURORA-B AOO evaluation model in an acceptable manner, and

(2) Compliance with the applicable regulatory requirements has been demonstrated.

3.4 Loss-of-Coolant Accident (LOCA) Analysis

NRC regulations require that licensees 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 ranges from scenarios with leakage rates just exceeding the capacity of normal makeup systems through those involving rapid coolant loss from the complete severance of the largest pipe in the reactor coolant system.

To support the planned transition to ATRIUM 11 fuel at Brunswick, Duke Energy analyzed the spectrum of LOCA events for this fuel design using the AURORA-B LOCA evaluation model (Reference 39). The NRC-approved AURORA-B LOCA evaluation model (Reference 40) uses the 10 CFR Part 50, Appendix K criteria in the analysis methodology. The licensee proposed

LAR (Reference 1) is the first plant-specific implementation of the AURORA-B LOCA methodology.

As described in the evaluation below, the NRC staff reviewed Duke Energy's implementation of the AURORA-B LOCA evaluation model for Brunswick to ensure compliance with applicable regulatory requirements. The NRC staff's review for Brunswick focused on the pertinent sections of the licensee's submittals (Reference 1) (particularly Attachment 13, ANP-3674P) and responses to RAIs (Reference 6). The NRC staff further conducted a regulatory audit on March 20-21, 2019 (Reference 32), which supported its review of the information docketed by the licensee.

During the NRC staff's review of the proposed license amendments, Duke Energy identified an error affecting previously submitted information related to the analysis of the LOCA event (Reference 41). Duke Energy provided a supplement on July 2, 2019 (Reference 7), which provided corrected analytical results.

3.4.1 Applicable Regulatory Requirements

The following regulatory requirements are pertinent to the analysis of the spectrum of LOCA events: 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors"; Appendix K to 10 CFR Part 50, "ECCS Evaluation Models"; Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants"; and Criterion 35, "Emergency Core Cooling"

3.4.1.1 10 CFR 50.46

In accordance with Limitation and Condition 4 from the NRC staff's final SE on ANP-10332P (Reference 40), the AURORA-B LOCA evaluation model may not be referenced as a basis for demonstrating adequate long-term core cooling for satisfying 10 CFR 50.46(b)(5). To demonstrate continued adherence to this requirement, Duke Energy cited the 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-20566-A (Reference 42). 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 NRC staff agrees that the existing licensing basis long-term core cooling analysis may continue to apply to ATRIUM 11 fuel because (1) specific details of the fuel assembly design do not play a significant role in the licensing basis long-term core cooling methodology, as compared to other factors such as the inherent plant design, decay heat, and peaking factors, and (2) the evolutionary design changes associated with the transition from ATRIUM 10 XM to ATRIUM 11 are not expected to have a significant effect on the results of the calculation.

3.4.1.2 Appendix K to 10 CFR Part 50

Appendix K to 10 CFR Part 50 consists of two parts: 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 Brunswick 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 (Reference 40).

3.4.1.3 Appendix A to 10 CFR Part 50, General Design Criterion 35

The GDC of Appendix A to 10 CFR Part 50 outline criteria for the design of nuclear power plants, typically in broad, qualitative terms. In particular, GDC 35 requires abundant core cooling sufficient to (1) prevent fuel and cladding damage that could interfere with effective core cooling and (2) limit the metal-water reaction on the fuel cladding to negligible amounts. GDC 35 further requires suitable redundancy of the ECCS such that it can accomplish its design functions assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources. Section 3.1 of the Brunswick UFSAR (Reference 11) describes how the plant design ensures conformance to GDC 35 and other GDC from Appendix A to 10 CFR Part 50.

3.4.2 Acceptability of LOCA Evaluation Model

Duke Energy analyzed the spectrum of postulated LOCA events to verify that all applicable regulatory requirements following the transition to ATRIUM 11 fuel are met. Duke Energy used the AURORA-B LOCA evaluation model developed by Framatome (Reference 39) to demonstrate compliance with the four acceptance criteria (i.e., subparagraphs (b)(1) through (b)(4)) from 10 CFR 50.46 that apply to the short-term LOCA analysis.

The AURORA-B LOCA evaluation model is an S-RELAP5-based methodology that incorporates a kernel of transient fuel rod thermal-mechanical subroutines from the RODEX4 code. As documented in an SE dated March 26, 2019 (Reference 40), the NRC staff found the AURORA-B LOCA evaluation model acceptable for application to LOCA analysis for BWR/3-6 plants.

While the generic evaluation model proposed by Duke Energy to support its proposed fuel transition has been previously found acceptable (Reference 40), the NRC staff reviews licensees' implementations of analytical evaluation models to ensure:

- Confirmation of acceptable plant-specific inputs to the evaluation model (Section 3.4.3.1),
- Confirmation of adherence to the approved evaluation model (Sections 3.4.3.2 and 3.4.3.3),
- Confirmation that results calculated using the evaluation model satisfy regulatory acceptance criteria and otherwise conform to expectations (Section 3.4.4), and
- Verification of acceptable responses to limitations and conditions specified in the NRC staff's safety evaluation (Section 3.4.5).

3.4.3 Evaluation Model Implementation

3.4.3.1 Plant-Specific Inputs

Several design differences exist between Brunswick, Units 1 and 2, that may affect the LOCA analysis, most notably the fuel inlet orifice diameter. The licensee stated in ANP-3674P (Attachment 13 of the LAR) that the reported FoMs derive from analysis using plant-specific inputs for Brunswick, Unit 2 (i.e., the unit with a smaller inlet orifice diameter); however, the licensee stated that the calculated results conservatively apply to Unit 1. During an audit conducted on March 20-21, 2019, the NRC staff confirmed that explicit calculations had been performed for both units and that the underlying calculation reports support the docketed conclusion in ANP-3674P that the reported results for Unit 2 bound both units.

The NRC staff's review found that the key plant parameters contained in ANP-3674P sufficiently conform to expected values from design-basis documentation for Brunswick with one exception. The NRC staff's audit further corroborated this conclusion. In RAI 19 (Reference 38), the NRC staff identified a potential exception to this general conclusion. From the information submitted by the licensee, it could not reasonably be determined whether the FoMs calculated in ANP-3674P for 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's response (Reference 6) to RAI 19 states that thermal-hydraulic compatibility analysis demonstrated **[[**
]]. The licensee stated that the
[[
]].
The licensee further stated that the LOCA analysis **[[**

]].

The NRC staff finds the licensee's response to RAI 19 acceptable because the licensee provided adequate evidence that the impacts of transition cycles containing co-resident ATRIUM 10XM fuel are **[[** **]]** established by the existing analysis.

3.4.3.2 Break Spectrum Implementation

The analysis for Brunswick considered a spectrum of postulated double-ended guillotine and split breaks in the recirculation system (i.e., **[[** **]]** suction piping, discharge piping). Non-recirculation-system breaks explicitly considered in the LOCA analysis included ruptures on the low-pressure core spray and feedwater system piping. In RAI 12 (Reference 38), the NRC staff requested that the licensee address postulated breaks on instrument lines and the reactor water cleanup system drain line from the bottom head of the reactor vessel. In light of the **[[** **]]** potential for maintaining an elevated liquid fraction at the break plane, the NRC staff did not agree with the conclusion stated in ANP-3674P that
[[
]].

The licensee's response to RAI 12 (Reference 6) states that instrument lines are **[[**

]].

The NRC staff's review did not fully agree with the licensee's statement in response to RAI 12 concerning the location of potential instrument line breaks. As described in Section 5.3.3.1.2.7 and Figure 5-5 of the Brunswick UFSAR, instrument line penetrations for incore neutron flux monitors exist on the reactor vessel bottom head. Regarding the bottom head drain line, the NRC staff finds that the calculation performed by the licensee provides sufficient evidence that a **[[**
]]. Furthermore, since the incore neutron flux instrumentation line is of similar dimension to the bottom head drain line, the NRC staff expects that similar conclusions apply. Therefore, the NRC staff finds the licensee's response to RAI 12 acceptable.

Table 5.1 of ANP-3674P identifies the single failures considered in the Brunswick LOCA analysis. The break spectrum analysis for Brunswick focused upon two potentially limiting single failures: (1) the failure of one train of direct current power (i.e., SF-BATT) and (2) the failure of a low-pressure coolant injection system injection valve (i.e., SF-LPCI). The licensee determined that other postulated single failures would result in equal or greater remaining capability for the ECCS. The NRC staff's review finds this determination appropriate, further observing that the licensee had considered the full set of postulated single failures defined in Brunswick's UFSAR (Reference 11) and cited in prior licensing applications for Brunswick (e.g., (Reference 43)) prior to focusing on the two potentially limiting cases noted above.

Consistent with ANP-10332P, break spectra were calculated for both mid- and top-peaked axial power profiles at the time of maximum fuel stored energy (i.e., near the beginning of the operating cycle). Furthermore, in light of Brunswick being licensed to the maximum extended load line limit analysis plus (MELLLA+) domain, a sufficient number of initial statepoints was considered in the break spectrum analysis to provide confidence that the most limiting conditions have been analyzed. Break spectra were performed for the following statepoints shown in Table 1 and Figure 1.

Table 1: LOCA Analysis Statepoints

Point	Operating Recirculation Loops	Reactor Power (percent rated)	**[[**
1	2	102	
2	2	102	
3	2	**[[ ]]**	
4	1	**[[ ]]**	**]]**

[[

]]

Figure 1: Brunswick Licensed Power/Flow Map

The first two analyzed statepoints were selected to envelop the full range of permissible core flows at rated thermal power. The third statepoint represents the **[[** **]]**. The fourth statepoint represents **[[** **]]** for single-loop operation (SLO). The NRC staff finds the selected analysis statepoints acceptable because they have been chosen consistent with previously approved methods for analyzing the MELLLA+ operating domain, particularly the NRC staff's SE on NEDC-33006P (Reference 44). Furthermore, in 2018, the NRC staff reviewed and approved a similar set of analyzed statepoints in a LOCA analysis for Brunswick (Reference 37).

In RAI 18 (Reference 38), the NRC staff requested additional information concerning how the LOCA analysis addresses the full suite of operating domains and equipment out-of-service conditions to which Brunswick has been licensed.[2] Table 2 summarizes the licensee's response to RAI 18 (Reference 6).

[2] Note that this information is necessary to satisfy Limitation and Condition 16 from the NRC staff's safety evaluation on ANP-10332P.

Table 2: Brunswick Licensed Operating Domains

Licensed Domain	Disposition
Two-Loop (Normal) Operation	Explicitly analyzed two statepoints that correspond to the maximum licensed power level.
Single-Loop Operation	Explicitly analyzed statepoint corresponding **[[ ]]** during single-loop operation.
MELLLA	Explicitly analyzed the **[[ ]]**.
MELLLA+	Explicitly analyzed three statepoints that **[[ ]]**.
Automatic Depressurization System Valve Out of Service	Explicitly analyzed, since all LOCA analyses for Brunswick assumed 2 automatic depressurization system valves are unavailable (1 assumed out of service, 1 assumed failed)
[[ ]]	**[[ ]]**
[[ ]]	**[[ ]]**
[[ ]]	**[[ ]]**
Main Steam Isolation Valve Out of Service	Licensee qualitatively dispositioned this operating condition, stating that operation in this domain is only allowed for two-loop operation and power levels below 70 percent where a linear heat generation rate reduction must be applied. The licensee stated that the **[[ ]]**

The NRC staff finds the licensee's response to RAI 18 acceptable because it identified the existing set of licensed operating domains and provided an appropriate basis in each case for concluding that the limiting FoMs calculated in its LOCA analysis bound all licensed operating conditions. The NRC staff's review finds that the break spectrum analysis described in ANP-3674P, Revision 2 (Reference 7), as supplemented by further information provided in response to RAIs conforms to the approved evaluation model documented in ANP-10332P (Reference 39).

3.4.3.3 Exposure Study Implementation

The NRC staff's review finds that the exposure study analysis described in ANP-3674P, as supplemented by further information provided in response to RAIs, conforms to the approved evaluation model documented in ANP-10332P (Reference 39). As shown in Table 9.1 of ANP-3674P, the exposure study considered **[[ ]]**. In particular, ANP-3674P displays results for **[[ ]]** accounting for exposure-dependent limiting values of the linear heat generation rate and maximum average planar linear heat generation rate.

The NRC staff observed that the exposure study for Brunswick described in ANP-3674P appeared to deviate from the methodology approved in the NRC staff's SE on ANP-10332P in that, at **[[**

]] approved by the NRC staff's SE. However, because these **[[** **]]** do not appear to produce limiting results in the analysis under review, the NRC staff finds that the exposure study results described in ANP-3674P provide sufficient confidence that the limiting results have been identified for the proposed Brunswick LOCA analysis. Furthermore, the licensee's stated adherence to the approved evaluation model in ANP-10339P provides adequate confidence that the approved break spectrum resolution has been analyzed, despite the omission of non-limiting intermediate exposure points from its submittal.

3.4.4 Calculated Results

3.4.4.1 Break Spectrum Results

Based upon the information submitted in Revision 2 of ANP-3674P, Table 3 summarizes limiting results for each power/flow statepoint considered in the licensee's break spectrum analysis. The limiting cases shown in Table 3 are **[[**

]].

Table 3: Summary of Break Spectrum Analysis Limiting Results

[[

]]

The results shown in Table 3 reflect the corrected data that was submitted in a supplement (Reference 10) after a code error was corrected. The code correction eliminated an error in the calculated results described in the original submittal (Reference 1). As described in ANP-3772P, an error in the automation software used by Framatome resulted in the creation of

RODEX4 input decks with certain fuel parameter input values shifted or truncated, which led to erroneous results in the FoMs calculated by S-RELAP5 (Reference 41).

In RAI 14, the NRC staff requested additional information concerning one case among the results in the original submittal for which the predicted peak cladding temperature **[[**
]]. The licensee responded that the case that **[[**
]]. The NRC staff finds the licensee's response to RAI 14 acceptable because it provides a reasonable physical explanation for the observed difference; furthermore, the NRC staff noted that the corrected results in Revision 2 of ANP-3674P, as excerpted above in Table 3, do not exhibit a similar issue.

ANP-3674P contains plots of key parameters as a function of time for the limiting scenario in the break spectrum analysis. These plots, as supplemented by additional information in response to RAIs, adequately conform to the NRC staff's expectations and are similar to the BWR/4 demonstration case included in ANP-10332P (Reference 39). Specifically, in RAI 22, the NRC staff requested that the licensee address two significant differences, namely that the Brunswick results predict two events **[[**
]].

The licensee's response to RAI 22 explained key differences between the Brunswick analysis and the demonstration case from ANP-10332P relative to **[[**
]]. First, the licensee identified that the Brunswick analysis **[[**
]]
demonstration case from ANP-10332P. Secondly, the licensee identified that the Brunswick analysis **[[**
]]. Both of these factors result in **[[** **]]** for the Brunswick analysis. The licensee's response to RAI 22 is supported by (1) included comparison plots showing key parameters for both the Brunswick and demonstration case analyses and (2) an additional sensitivity case that showed a **[[**
]]. The NRC staff finds the licensee's response to RAI 22 acceptable because the licensee provided credible physical explanations for the observed differences and supported them with convincing analytical evidence (i.e., comparison plots and sensitivity analysis).

ANP-3674P does not contain plots of the peak cladding temperature as a function of break size. To ensure that the evaluation model made reasonable predictions for Brunswick across the entire spectrum of breaks, the NRC staff requested in RAI 13 that the licensee provide break spectra for **[[**
]]. In response, the licensee provided the requested break spectrum plots. **[[**

]]. The licensee stated that the analysis was performed in accordance with the break spectrum resolution specified in its response to RAI 29.b from the NRC staff's review of ANP-10332P (Reference 40). The NRC staff's review found the licensee's response to RAI 13 acceptable because it provided the requested plots, and the NRC staff's review found the results consistent with both (1) expected physical behavior for the LOCA event at a BWR and (2) the procedure for break spectrum analysis in the approved AURORA-B LOCA evaluation model described in ANP-10332P (Reference 39).

In RAI 11, the NRC staff requested justification for a statement in ANP-3674P that the limiting break would not be affected by a change in fuel design. Such a conclusion was not approved in the NRC staff's SE on ANP-10332P (Reference 39), which contains a limitation and condition requiring the evaluation of new fuel designs to ensure compatibility with the AURORA-B LOCA evaluation model. The licensee responded that the statement in question was intended to reference the technical basis for the AURORA-B LOCA evaluation model to **[[** **]]** as approved by the NRC staff in its SE on ANP-10332P. The statement that the limiting break would not be affected by a change in fuel design was unintended, and the licensee struck this language from Revision 2 of ANP-3674P. The NRC staff finds the response acceptable because the elimination of the unintended language prevents a potential misinterpretation of the basis for the NRC staff's approval of the AURORA-B LOCA evaluation model.

3.4.4.2 <u>Exposure Study Results</u>

As shown in Table 3, the break spectrum analysis found the most limiting scenario to be a double-ended guillotine rupture of the recirculation system suction line with a discharge coefficient of 1.0, initiating from the **[[** **]]** statepoint with a top-peaked power shape and a single failure in the direct current power supply (i.e., SF-BATT). The licensee's exposure study for this limiting scenario predicted the FoMs shown in Table 4.

Table 4: Predicted Figures of Merit for Brunswick LOCA Analysis

Figure of Merit	Limiting Exposure	Predicted Value	Acceptance Criterion
Peak Cladding Temperature	**[[** **]]**	**[[**	≤ 2,200 °F
Maximum (Local) Cladding Oxidation	**[[** **]]**		≤ 17 percent
Maximum (Core-Wide) Hydrogen Generation	All	**]]**[3]	≤ 1 percent

The NRC staff compared the predicted results in Table 4 to those from a previous LOCA analysis for Brunswick performed in 2015 (Reference 45). Although the previous results were calculated for a different fuel type (i.e., ATRIUM 10XM) and used a significantly different evaluation model (i.e., EXEM BWR-2000 (Reference 46)), the calculated peak cladding temperatures were found to be in a similar range (i.e., within 50 °F).

In RAI 15, the NRC staff requested that the licensee clarify the influence of pre-transient oxidation on the trend of the maximum local cladding oxidation as a function of assembly

[3] As stated in ANP-3674P, Revision 2, the FoM for corewide hydrogen generation was conservatively estimated using **[[** **]]**

average planar exposure in Table 9.1 of ANP-3674P. The licensee responded by explaining the effect of pre-transient oxidation and **[[**

]]. The NRC staff finds the response acceptable because the licensee provided the requested information, and the NRC staff's review found it consistent with physically expected cladding oxidation behavior during normal operation and LOCA conditions.

3.4.5 Conformance with Limitations and Conditions

Revision 1 of ANP-3674P (Attachment 13 of the LAR (Reference 1)) was submitted prior to the issuance of the NRC staff's final SE on ANP-10332P (Reference 40) and (Reference 47). It contains the licensee's rationale for concluding that all limitations and conditions from the NRC staff's draft SE on ANP-10332P (Reference 48) have been satisfied. As such, several of the NRC staff's RAIs also refer to these draft limitations and conditions (Reference 38). Subsequently, the NRC staff issued its final SE on ANP-10332P, which contained fewer limitations and conditions with some modifications relative to those in the draft SE.

To support the present LAR, the NRC staff determined that Duke Energy must assure consistency with the AURORA-B LOCA evaluation model approved in the final SE on ANP-10332P (Reference 40), including all limitations and conditions. Therefore, the licensee subsequently reviewed the limitations and conditions in the NRC staff's final SE and, in Revision 2 of ANP-3674P (Reference 7), the licensee addressed the full set of limitations and conditions from the NRC staff's final SE on ANP-10332P. To promote clarity, this SE will refer to limitations and conditions according to the numbering in the final SE on ANP-10332P, with the numbering from the draft SE provided parenthetically.

The licensee's proposed disposition of limitations and conditions in most instances affirmed, prima facie, conformance to the regulatory position imposed therein. However, in certain instances that are discussed below, the NRC staff finds a more detailed review necessary to confirm that Duke Energy had appropriately addressed the applicable limitations and conditions.

Regarding Limitation and Condition 11 (draft Limitation and Condition 15), the NRC staff requested justification in RAI 16 for the method of determining the fuel cladding temperature ramp rate when calculating cladding strain and rupture behavior.

The licensee responded to RAI 16 by stating that the **[[**

]] and that the trend of peak cladding temperature with time was nearly identical to the baseline calculation.

The NRC staff finds the response to RAI 16 acceptable because the licensee provided a credible physical explanation to justify its methodology for determining the temperature ramp rate and supported its explanation with appropriate analytical evidence (i.e., a sensitivity study **[[** **]]** and showed minimal differences relative to the baseline case).

Regarding Limitation and Condition 13 (draft Limitation and Condition 19), the NRC staff requested confirmation in RAI 17 that the LOCA analyses adequately account for **[[** **]]** when determining the start of the refill and reflood phases that are used to trigger the release of heat-transfer lockouts imposed by Appendix K to 10 CFR Part 50. The licensee responded by stating that an additional sensitivity case was performed that **[[** **]]**. The licensee found that this sensitivity case resulted in the **[[** **]]**. The NRC staff finds the response to RAI 17 acceptable because the licensee provided a credible physical explanation for the observed behavior that was supported with appropriate analytical evidence (i.e., a sensitivity study **[[** **]]** for the Brunswick analysis).

Regarding Limitation and Condition 14 (draft Limitation and Condition 20), the NRC staff's review evaluated whether a **[[** **]]** in ANP-3674P, Revision 2.[4] The NRC staff viewed an additional small-break LOCA scenario during the regulatory audit conducted on March 20-21, 2019 (Reference 32), which was also consistent with this conclusion. The licensee further confirmed in ANP-3674P that the analyses **[[** **]]**. Based upon this information, the NRC staff considered Limitation and Condition 14 to have been acceptably addressed.

Regarding Limitation and Condition 16 (draft Limitation and Condition 27), which requires that the licensee justify that the LOCA analysis bounds all licensed operating domains, the NRC staff obtained the information necessary to address this item via RAI 18, as documented in Section 3.4.3.2.

Regarding Limitation and Condition 17 (draft Limitation and Condition 28), Framatome had originally proposed that analyses with the AURORA-B LOCA evaluation model focus upon scenarios involving the loss-of-offsite power. Although the NRC staff generally agreed with this position, Limitation and Condition 17 from the NRC staff's SE requires consideration of

[4] The NRC staff's review found that a similar conclusion also held for the results described in Revision 1 of ANP-3674P (Attachment 13 of the LAR).

scenarios with offsite power available, in accordance with GDC 35, in the event **[[**

]]. The NRC staff audited this sensitivity study and finds that it supports the conclusion docketed in ANP-3674P. Therefore, the NRC staff considered Limitation and Condition 17 to be adequately addressed.

Regarding Limitation and Condition 23 (draft Limitation and Condition 35), the NRC staff requested in RAI 20 that the licensee provide **[[**

]]. The licensee's response described a sensitivity study that **[[**

]] The NRC staff found the licensee's reasoning consistent with physical expectations, **[[**

]] The NRC staff observed that the licensee had performed its sensitivity study for the limiting top-peaked case from Revision 1 of ANP-3674P (Reference 1). The NRC staff considered the use of this case reasonable because it is similar to the limiting case in Revision 2 of ANP-3674P. Therefore, considering both the physical justification and sensitivity studies described by the licensee, the NRC staff finds the licensee's response to RAI 20 acceptable.

Regarding Limitation and Condition 25 (draft Limitation and Condition 37), the NRC staff requested in RAI 21 that the licensee provide **[[**

]] The licensee's response described the results of a sensitivity study that **[[**

]]. The licensee's sensitivity analysis was based on the limiting case from the break spectrum analysis in ANP-3674P, Revision 1 (Reference 1). The NRC staff considered the use of this case reasonable because it is similar to the limiting case in Revision 2 of ANP-3674P (Reference 7). The NRC staff finds the licensee's response to RAI 21 acceptable because the licensee performed the requested sensitivity study, and the sensitivity study demonstrated that the effect of the four model changes did not significantly affect the results calculated in the Brunswick LOCA analysis.

As described above, the NRC staff's review of the information provided in Appendix A of ANP-3674P, Revision 2, found that the implementation of the AURORA-B LOCA evaluation model for Brunswick has complied with the limitations and conditions specified in the NRC staff's final SE for ANP-10332P (Reference 40).

3.4.6 LOCA Analysis Conclusion

The NRC staff reviewed the information in the licensee's submittals pertaining to the analysis of the spectrum of postulated LOCA events for Brunswick, Units 1 and 2, including the original submittal (Reference 1), as well as relevant responses to requests for additional information (Reference 6) and relevant supplementary submittals, (Reference 41). The NRC staff's review was further supported by a regulatory audit (Reference 32), which was used to confirm information included in docketed submittals. Based upon its review, as documented above, the NRC staff has concluded that:

(1) the licensee has proposed to implement the AURORA-B LOCA evaluation model described in ANP-10332P (Reference 39) in an acceptable manner, and

(2) compliance with the applicable regulatory requirements described above in Section 3.4.1 (i.e., 10 CFR 50.46, Appendix K to 10 CFR Part 50, and GDC 35) has been demonstrated.

3.5 ATWS-I

3.5.1 Regulatory Evaluation

Section 50.62 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.

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 26), establishes acceptance criteria for ATWS events. SRP 15.8 includes additional GDC beyond those listed above; however, they define vessel, ECCS, and containment performance requirements. This is not a significant concern for ATWS-I events; therefore, these GDC were not considered as part of the review of the ATWS-I methodology submitted for review by the licensee.

The NRC staff used the review guidance in SRP Chapter 15.0.2, along with the applicable acceptance criteria in SRP Chapter 15.8, in conducting its review of the LAR. To the extent possible, the NRC staff leveraged the prior review and approval of the RAMONA5-FA long-term stability solution (LTSS) methodology and the Monticello ATWS-I methodology (Reference 30).

3.5.2 Plant-Specific Methodology

In its submittal, Brunswick describes a methodology by which the RAMONA5-FA code can be used for analysis of the ATWS-I event. The NRC staff's review of the ATWS-I portions of the submittal was performed by following the key elements of the evaluation model development and assessment process (EMDAP) outlined in Regulatory Guide (RG) 1.203 (Reference 49) and echoed in SRP 15.0.2 (Reference 26). While this guidance was intended mainly to address design-basis accidents, the general principles can be applied to ATWS-I analysis methodologies.

There are five elements provided by the guidance for model evaluations. The NRC staff reviewed each of the following specific elements:

1. Accident scenario description and phenomena identification and ranking – Brunswick's break-down of the ATWS-I event and its relevant phenomena and characterization of the consequences. The NRC staff utilized other available approved PIRTs and relevant guidance to inform their assessment of whether all the relevant phenomena are appropriately addressed in the validation basis, acceptance criteria, and/or procedure used to confirm that the acceptance criteria are met.

2. Evaluation methodology – the proposed ATWS-I analysis methodology, including initial conditions, assumptions, and approach to ensuring that the acceptance criteria are met. Since this methodology includes use of the evaluation model by extension, this area includes the models and correlations within the RAMONA5-FA code.

3. Code assessment – the assessments performed by Brunswick to validate the RAMONA5-FA performance for the thermal hydraulic and neutronics phenomena expected during ATWS-I events, particularly during unstable power oscillations and for the specific fuel design currently used by Brunswick.

4. Uncertainty analysis – This area is not formally required since the ATWS-I event is not a design-basis event. However, the NRC staff did confirm that the licensee adequately addressed the parameters that have the most impact on the results of the analyses through conservative assumptions or sensitivity studies.

5. Documentation – The NRC staff reviewed Brunswick's documentation of the various aspects of this analysis methodology, including the submittal as well as various documents supporting the RAMONA5-FA code and calculational files or procedures that provide detail on the intended steps to be taken when performing ATWS-I analyses or qualifying the methodology for different plant configurations and fuel designs.

To address review area 5, the documentation associated with the submittal is contained in various calculational files, validation reports, technical references, code documentation, and the submittal itself. Additional documentation reviewed by the NRC staff during the audit of ANP-10346P (Reference 50) was not formally submitted on the docket but was summarized in the audit report (Reference 51). This information was not necessary to make a safety finding; however, the NRC staff did confirm that the information was consistent with information in the LAR and in the RAI responses. The documentation included sufficient information for the NRC staff to understand the intended application and validation of the methodology described in the LAR and make their safety finding. As such, NRC staff acceptance of the adequacy of the licensee's discussion of each area includes acceptance of the licensee documentation associated with that area.

RG 1.203 also discusses a sixth key element of the EMDAP QA processes. This aspect is not explicitly discussed in this SE because the QA processes are captured within the Brunswick QA program, which is consistent with the requirements in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The NRC staff inspects licensee's QA programs to confirm they meet all regulatory requirements.

The licensee submitted Framatome's RAMONA5-FA in Attachment 14 to the LAR (Reference 1) as a plant-specific ATWS-I methodology while the same methodology was under review by the NRC as a generic TR (Reference 50). Because of schedular necessity, the licensee chose not to wait for the TR to be approved and reference it in the LAR. During the review of the LAR, Framatome responded to the NRC staff's RAIs on the TR under review (Reference 52). The licensee then supplemented the LAR to adopt the same Framatome RAI responses in Appendix A of its response letter to the NRC staff's RAIs in other area of the LAR (Reference 6). The NRC staff incorporated this supplement in their review and this SE to be consistent with that of the Framatome TR.[5] Specifics on the application of the methodology to Brunswick and the ATRIUM 11 fuel are contained in Section 3.5.5 of this SE.

3.5.2.1 Accident Scenario Description and Acceptance Criteria

Per the review guidance in Chapter 15.0.2 of the SRP, the accident scenario description and phenomena identification and ranking process are intended to ensure that the dominant physical phenomena influencing the outcome of the given accident scenario are correctly identified and ranked. Once an accident scenario has been described, then FoMs can be determined for use in evaluating whether acceptance criteria are met. The subsequent phenomena identification and ranking process will determine the physical phenomena affecting the FoMs and rank them by their importance. By doing so, a licensee can demonstrate that reasonable assurance exists that it is accurately capturing and modeling the dominant physical phenomena necessary for evaluation of the accident scenario in question.

Section 4.0 of the submittal provides an extensive description of the various characteristics of the large coupled neutronic/thermal-hydraulic oscillations that uniquely characterize the ATWS-I event. In addition, other potential characteristics of an ATWS-I event that are potentially important are discussed, including potential prompt criticality, the possibility of boiling within bypass flow channels, and the cyclical dryout/rewetting that may be experienced by fuel. The licensee's understanding of the ATWS-I event characteristics was used to develop a phenomena identification and ranking table (PIRT), which identifies specific physical processes and parameters that are expected to be relevant to the ATWS-I event.

The PIRT is intended to identify the dominant phenomena pertaining to ATWS-I licensing analyses. Because the RAMONA5-FA ATWS-I methodology is based on a preexisting approved methodology (the RAMONA5-FA LTSS methodology), the licensee used the ATWS-I PIRT to determine which equations and closure relations required development or enhancement in order to apply the methodology to ATWS-I. In addition to model development, the licensee also used the ATWS-I PIRT to define the types of validation and sensitivity studies that were needed to support the methodology.

Accordingly, an important step in the NRC staff's evaluation of the RAMONA5-FA ATWS-I methodology was to determine whether the ATWS-I PIRT portions of the submittal encompassed all the important phenomena for ATWS-I analyses, and whether the importance levels indicated were consistent with the NRC staff's current knowledge of the ATWS-I phenomena. To make this determination, the NRC staff reviewed PIRTs developed in 2001 and 2011 under the guidance of the NRC ((Reference 53) and Section 5 of (Reference 54)), more recent NRC published studies of ATWS-I scenarios (Reference 55) (Reference 56) and other

[5] U.S. NRC, Final Safety Evaluation for Framatome Inc., TR ANP-10346, Revision 0, "ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA," October 30, 2019 (ADAMS Accession No. ML20034E889).

available sources of information from open literature or internal NRC experience based on reviewing ATWS-I methodologies.

An important basis for the PIRT is identification of appropriate FoMs that correlate with the acceptance criteria for the ATWS-I evaluation. The primary acceptance criterion is the peak cladding temperature (PCT), since the licensee elected to use a 2,200 °F upper limit on PCT to demonstrate that fuel/cladding damage sufficient to challenge core cooling will not occur. Secondary acceptance criteria are discussed in the Calculational Procedure section of this document, which are related to the timing of events in the ATWS-I accident progression (including any required mitigating actions). When appropriate FoMs are identified, the phenomena expected to affect the FoMs can be identified, as well as ranked, in importance.

The licensee identified three FoMs, which are evaluated by the NRC staff below:

- Oscillation inception, which is correlated with the decay ratio (DR). Since the DR describes the relative instability of a system, a higher DR leads to earlier oscillation inception, as well as a more rapid increase in oscillation magnitude. As such, this FoM directly affects the timing of failure to rewet, should it be predicted to occur. This is consistent with the primary FoM for the PIRT developed by the NRC staff (Reference 54).

- Limit cycle amplitude, which defines the worst possible oscillation that can occur for a given system and core configuration. The oscillations that arise during an ATWS-I will reach a maximum amplitude due to physical limitations on the severity of the density and power swings. As previous NRC experience (Reference 55) indicates that the limiting amplitude is not well correlated with the DR to ensure that the worst-case power oscillations are captured, a separate FoM is necessary.

- Post-dryout, which generally encompasses the dryout and rewetting behavior. This includes cyclical dryout and rewetting, as well as periods of extended dryout due to failure to rewet. This behavior directly affects the PCT, since loss of cooling due to dryout is the primary cause of any PCT increases during the ATWS-I event that are significant enough to challenge the 2,200 °F limit.

Based on the NRC staff's knowledge of the ATWS-I event as correlated with the information presented in the LAR, the licensee's characterization of the event and the relevant phenomena was acceptable. The licensee identified a key acceptance criterion – core coolability. Even though maintaining the PCT below 2,200 °F is not a precondition for ensuring that core coolability is maintained during ATWS-I conditions, the licensee proposed the use of this acceptance criterion as a proxy for core coolability. This proxy for core coolability using a PCT limit is already used in NRC regulatory requirements to ensure core coolability is maintained during design basis accidents. In addition, the licensee identified that the timing of specific transitions are important in providing reasonable assurance that the prescribed operator actions would occur in adequate time to mitigate the consequences of the event. This information was used with the event and phenomena characterization to develop a set of high importance phenomena that must be appropriately captured by the ATWS-I analysis methodology. The ranking of phenomena as described in the PIRT documented by Brunswick is consistent with the NRC staff's understanding of the ATWS-I event.

The NRC staff's conclusions regarding the ATWS-I PIRT were used in other portions of the review, specifically. in the Evaluation Model, Code Assessment, and Uncertainty Analysis sections of this SE. Table 5 indicates which section of this SE is associated with each of the high and medium ranked phenomena from the Brunswick PIRT. The low ranked phenomena are also captured or otherwise dispositioned in the analysis methodology but are not expected to have a sufficient impact on the ATWS-I evaluation results such that a high level of fidelity or sensitivity studies are required. While not explicitly mentioned in the below table, the integral benchmarks discussed in Sections 3.5.3.6 and 3.5.3.7 provide validation of the methodology's ability to conservatively predict the important phenomena affecting the FoMs associated with the ATWS-I event.

Table 5: Items from ATWS-I PIRT and Associated Section in this SE

High Importance Phenomena from Table 4-1	NRC Evaluation (including relevant sections from this SE)

High Importance Phenomena from Table 4-1	NRC Evaluation (including relevant sections from this SE)

Medium Importance Phenomena from Table 4-1	NRC Evaluation (including relevant sections from this SE)

Medium Importance Phenomena from Table 4-1	NRC Evaluation (including relevant sections from this SE)

As a result of the above evaluations of the postulated accident scenarios and phenomena evaluated, the NRC staff has determined that the licensee appropriately characterized the ATWS-I scenario, identified the appropriate acceptance criteria, and constructed a PIRT that identified the most important phenomena and processes for the analysis methodology to capture. The NRC staff considerations in determining whether the analysis methodology is acceptable with respect to each phenomenon are discussed in the following sections, as outlined in the above table. In general, the oscillation inception FoM was addressed by examining how the given model or correlation affects the timing of oscillation onset. The limit cycle amplitude and post-dryout FoMs were considered through use of the PCT as a proxy, since a conservative application of these FoMs would be expected to increase the PCT. In several cases, the FoMs were not explicitly evaluated because a model or correlation used accurately captured the phenomenon of interest.

3.5.2.2 Evaluation Methodology

Chapter 15.0.2 of the SRP describes the review of the evaluation model as part of the transient and accident analysis methods. The associated acceptance criteria indicate that models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident. In addition, it must be determined if the physical modeling described in the theory manual and contained in the mathematical models is adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.

Several models were previously reviewed and approved by the NRC for similar purposes. For those models, the scope of the NRC staff review was limited to confirming the applicability of those models to the ATWS-I event. The models described by the licensee are discussed in individual subsections below.

Section 4.16 of the LAR identifies major assumptions made in RAMONA5-FA ATWS-I relative to the previously approved RAMONA5-FA LTSS methodology. These major assumptions were identified and justified primarily through engineering judgment based on extensive application

experience with the approved RAMONA5-FA methodology, which is similar to the current methodology.

The major assumptions identified in Section 4.16 of the LAR are (1) the 3D nodal adaptive neutron kinetics methodology is assumed adequate for ATWS-I, and (2) new water property functions are used and **[[ ]].** The NRC staff's discussion and evaluation of these two assumptions is contained in Sections 3.5.2.3 and 3.5.2.5.11 of this SE, respectively.

3.5.2.3 <u>Neutronics Review</u>

The methodology uses an adaptive 3D neutron kinetics solution with **[[ ]]** to determine the time evolution of the 3D neutron flux distribution during anticipated transient events. This neutronic solution methodology is identical to that used in the RAMONA5-FA LTSS methodology and the Monticello ATWS-I methodology. This adaptive 3D neutron kinetics solution is a **[[ ]]** methodology, which means that it solves for the neutron flux level at each discretized axial level in each fuel assembly in the core, **[[ ]].** The neutronic and thermal hydraulic solutions are coupled on **[[ ]]** as well. This results in a coupled neutronic/thermal-hydraulic methodology that has sufficient fidelity to accurately resolve anticipated axial and radial oscillation patterns, including corewide and side-to-side radial mode behavior (including more complex modal interactions such as rotating modes) as well as single-channel instability. This is a key reason why the **[[ ]]**

During the large-amplitude oscillations that are characteristic of the ATWS-I event, up to and including limit cycles with dryout and failure to rewet, the neutronic solution becomes even more highly-peaked spatially and undergoes larger variations over time relative to LTSS applications with smaller oscillation amplitudes. However, based on the NRC staff's knowledge and experience with similar neutronics methodologies, this behavior is not expected to challenge the ability of the methodology to accurately represent the physical behavior under these conditions. In fact, the **[ ]]**. Therefore, the NRC staff has concluded that the adaptive 3D neutron kinetics solution remains applicable and appropriate for this application.

Because the neutronic methodology used by Brunswick did not change with respect to the RAMONA5-FA LTSS methodology, and this neutronic methodology remains suitable for ATWS-I applications, the NRC staff did not perform a detailed review of the entire neutronic methodology. However, the NRC staff did review the methodology for **[[ ]]** to the neutronic solution during a transient calculation. The NRC staff's experience has indicated that the method of **[[ ]]** is important for correctly determining the timing of oscillation onset, which affects the method's ability to predict whether operator actions occur in time to mitigate the potential public safety consequences of ATWS-I.

To assist in determining whether the implementation of **[[ ]]** was acceptable for ATWS-I, the licensee supplemented the LAR (Reference 6) with the generic Framatome ATWS-I methodology review's RAI responses (Reference 52) as Appendix A. Using the response to RAI 13 of the generic ATWS-I methodology, the licensee clarified that the **[[**

]] This information provides a high degree of confidence that both the in-phase and out-of-phase modes can be adequately and reliably excited in a timely fashion (i.e., shortly after one or both modes becomes unstable), **[[**

]].

The NRC staff finds the implementation of **[[ ]]** in analysis of the ATWS-I event using RAMONA5-FA as described in the submittal to be acceptable, in combination with the plant-specific inputs discussed in Section 3.5.5.2 of this SE. The use of the neutron kinetics solution implemented in RAMONA5-FA was also found to be acceptable based on previous NRC approvals and the known ability of this methodology to capture neutron kinetics responses similar to those expected during an ATWS-I event.

3.5.2.4 Fuel Thermodynamics Review

3.5.2.4.1 ATWS-I Fuel Pin Heat Conduction

The methodology described by the licensee determines the time-dependent axial and radial temperature distribution in the "average rod" within each fuel assembly, as well as in the "hot rod" (peak power rod) within each assembly. The average rod temperatures and heat generation rate are used **[[**

]]

At each axial level in the assembly, a one-dimensional (1D) radial time-dependent transient temperature calculation is performed from the radial center of the fuel pin to the outer surface of the cladding, similar to the Monticello ATWS-I methodologies. This is consistent with the previous methodologies, **[[ ]]** which is acceptable based on the fact that the model exhibits good agreement with experimental benchmarks, **[[**

]].

Unlike the **[[ ]]** in the Monticello ATWS-I methodology, the licensee's methodology solves the radial temperatures **[**

]]

Therefore, the NRC staff finds the licensee's fuel rod conduction methodology to be acceptable based on its use of previously approved modeling approaches, combined with state-of-the-art computational solution schemes appropriate for the intended application.

3.5.2.4.2 ATWS-I Heater Rod Conduction Model

A separate heat conduction model is used for calculating time-dependent axial and radial temperature distribution in heater rods representative of the KATHY facility. The NRC staff determined that the only difference between this model and the one in the Monticello ATWS-I methodology was that the latter calculated the **[[**

]]

Because the heater rod conduction model in the methodology is more accurate than the previously-accepted model used in the Monticello ATWS-I methodology and is used for the same scope and range of application, namely to determine the heater rod temperature response during the KATHY ATWS-I experiments, the NRC staff finds that the previous approval of the heater rod conduction model in the Monticello ATWS-I methodology is applicable to the methodology, and no further review of the model was performed.

3.5.2.4.3 Heat Transfer Coefficient

The ability of the fluid to transfer heat from the outer surface of the clad or heater rod is strongly dependent on the phase of the fluid (liquid, vapor or both) and the ability of the liquid phase to contact the surface. The methodology calculates a wetted heat transfer coefficient (HTC) under single-phase conditions using the **[[ ]]** correlation, a wetted HTC under two-phase conditions using **[[ ]]** correlation, **[[ ]]**, and models for transitions between these regimes.

The **[[ ]]** single-phase liquid correlation and the **[[ ]]** correlation are the same as in the RAMONA5-FA LTSS methodology and the Monticello ATWS-I methodology. **[[**

]] The NRC staff finds that the single phase liquid and boiling heat transfer models are acceptable based on their previous validation and approved use in the RAMONA5-FA LTSS methodology, and that the regime transition and **[[ ]]** are acceptable because they are based on realistic physical principles

and demonstrate good agreement with measured data in the benchmarks given in Section 6.0 of the LAR, which cover a wide range of conditions applicable to ATWS-I.

The dry HTC is determined using a correlation **[[**

]] The NRC staff notes that the heat transfer situation **[[**

]]

To address the potential concerns (RAI 3 of the generic ATWS-I methodology), regarding the acceptability of using a **[[**

]], in particular, the need to justify **[[**

]], the licensee provided (Reference 6) additional plots **[[**

]]

With regard to the dependence **[[** **]]**, the NRC staff examined the **[[**

]] Therefore, the NRC staff finds the licensee's use of the **[[** **]]** to be acceptable because this ensures **[[**

]]

However, the NRC staff observed the following:

(1) the magnitude of the power oscillations following failure to rewet was **[[**

]];

(2) the dry HTC correlation was **[**

]]; and

(3) the licensee only used the **[[**
]].

Additional data points were provided by the licensee from **[[**

]] the NRC staff finds reasonable assurance that the **[[** **]]** This is due partly to the fact that KATHY provides prototypical ATWS-I conditions, and the NRC staff does not expect the heat flux to go significantly beyond the KATHY data range without causing the fuel to exceed 2,200 °F.

The data provided by the licensee demonstrates a clear correlation between **[[**

]] This relationship, as well as the reasonableness of the data points, were confirmed by comparison to independent KATHY testing (Reference 57) performed by the NRC. Furthermore, the NRC staff determines that the approach used by the licensee establishes a correlation that is reasonably bounding by incorporating the conservatisms discussed above as items (2) and (3).

The reference temperature treatment is identical to the Monticello ATWS-I methodology, and the **[[**
]] These models and correlations have previously been approved by the NRC for use at Monticello as part of ANP-3274P (Reference 58), and the validation suite (discussed in Section 3.5.3 of this SE) provides reasonable assurance that the methodology presented for determining heat transfer is applicable to Brunswick for the conditions expected during the ATWS-I event. Therefore, the NRC staff finds this aspect of the methodology to be acceptable for use in analysis of the ATWS-I event, including the **[[** **]]**.

3.5.2.4.4 Hot Fuel Pin Model

As discussed in Section 3.5.2.4.1 of this SE, the submittal methodology provides a separate calculation for temperature and heat transfer in the hot fuel pin, as opposed to the average fuel pin, which the NRC staff finds to be an acceptable approach to calculate the maximum cladding temperature and provide realistic coolant temperatures and reactivity feedback for the neutronics solution.

[[

]] The NRC staff finds this approach acceptable

because it provides conservative **[[   ]]**, which is expected to increase the calculated PCT values during ATWS-I analyses.

3.5.2.4.5 Review of Section 5.2.5 – Material Properties

The submittal methodology uses fuel pellet and cladding thermophysical properties based on **[[   ]]**. The NRC staff finds this approach acceptable for use in the RAMONA5-FA ATWS-I calculations because these models account for all important fuel characteristics relevant to ATWS-I, including the **[[**

]].

Appendix A of RAI response (Reference 52) includes an update to ANP-10346P that, among other changes, appends Appendix D, which presents modified fuel rod models that account for chromia doping of the UO_2 fuel pellets. The fuel thermal conductivity model was adapted from the approved RODEX4 model (Reference 13). The **[[   ]]** model was developed by benchmarking to the approved RODEX4 model (Reference 13). Because these models are based on previously reviewed and approved models for chromia doped fuel, the NRC staff finds these models acceptable for use in characterizing chromia doped fuel properties for ATWS-I analyses performed using the methodology as described by the licensee.

3.5.2.4.6 Pellet Clad Gap Heat Transfer Coefficient

The gas gap between the fuel pellet and cladding may introduce a large thermal resistance that affects both the amplitude and phase shift of fluctuations in heat flux at the cladding outer surface during a given oscillation period. In turn, the decay ratio and oscillation frequency of predicted ATWS-I oscillations may be significantly affected, which may impact whether the fuel remains protected within the time required for the ATWS-I mitigation actions to take effect. Due to burnup and history effects, the fuel-clad gap in twice- and even once-burned fuel will typically be closed at normal operating conditions, resulting in only a small thermal resistance; however, after the recirculation pump trip during the postulated turbine trip with bypass (TTWB) and two recirculation pump trip (RPT) ATWS-I events, the gap will typically re-open and result in significant thermal resistance that must be accurately accounted for in the ATWS-I methodology.

[[

]]

In the LAR supplement (Reference 6), the licensee provided the NRC staff additional information on how the fitting parameters for the gap conductance model were determined from measured data, particularly when direct experimental validation for each parameter was not possible or not available. This supplemental information also address the NRC staff's review of the generic ATWS-I methodology (RAI 2) to better understand the method of **[[** **]]**, as was stated in the submittal. In the supplement, the licensee indicated that **[[** **]]** Because these values cannot be directly measured but have a potentially significant effect on stability behavior, these values could have subsequently been adjusted **[[** **]]** After examining the relevant models and experimental database, the NRC staff concluded that the use of **[[** **]]** is acceptable because it provides a physically reasonable model of gap behavior as a function of fuel conditions (burnup, temperature, etc.) in relevant ATRIUM fuel types. The good agreement with **[[** **]]** provides strong evidence that these values remain applicable and acceptable under BWR stability conditions.

The NRC staff's review of the gap heat transfer coefficient model concluded that it includes the important physics required to calculate the intra-pin heat transfer behavior during a postulated ATWS-I event, including the initial transient, onset and growth of oscillations, dryout/rewet phase, and high-temperature failure-to-rewet phase of the event. The models for calculating **[[** **]]** In the LAR supplement (Reference 6), the licensee provided the NRC staff with justification that the gap heat transfer coefficient model provides a reasonable and accurate representation of gap behavior during postulated ATWS-I events.

In the supplement, the licensee indicated that the **[[** **]]**, as discussed above. **[[**

]] While these data do not provide a direct "separate-effects" validation of the fuel and gap heat transfer models, the close overall agreement with these measured integral effects data, with little or no average bias in the errors, provides the NRC staff with confidence that the fuel and gap heat transfer was modeled in a reasonable and acceptable manner.

The licensee supplemented the LAR (Reference 6) to clarify the role of the gap model during ATWS-I events. In the referenced documentation, the licensee provided additional sensitivity results for several linear stability benchmark cases and a simulated ATWS-I event by artificially adjusting the gap conductance **[[**

]] Based on this **[[** **]]**, the NRC staff concluded that the gap conductance model has **[[** **]]** on the stability predictions for regional mode cases. For the single linear benchmark case in which the global mode was dominant **[[**

]], respectively, compared to the result with non-adjusted conductance. The licensee indicated that **[[**

]] In this particular case, the more significant sensitivities were such that the base calculation represented a reasonably conservative result. However, the interrelationship between the gap conductance model and the stability phenomena is expected to be at least somewhat sensitive to the specific scenarios being analyzed.

The **[[** **]]** adjustment in gap conductance also had a relatively mild effect for the ATWS-I sample problem for which the sensitivity results were provided in the LAR supplement (Reference 6) as well. **[[**

]] Any impact due to uncertainties in the gap conductance model that may result in challenges to the regulatory limit would be expected to occur in situations where operator action is necessary to prevent the PCT from exceeding regulatory limits, AND where relatively small margins exist between the licensing basis operator action time and the time at which operator action would be too late to stop the PCT from increasing beyond the regulatory limit. In such cases, a significant increase in the DR may lead to an earlier failure to rewet and allow the PCT to increase for a longer time prior to mitigation. Once the margin in operator action time is appropriately justified, including

any consideration of the gap conductance uncertainty, the sensitivities are not expected to change significantly from cycle to cycle. However, if another new fuel design is introduced in the future, which changes the characteristics of the geometry or materials used in modeling the fuel rod, gap, and cladding, gap conductance uncertainty may change. As discussed in Section 3.5.5.3 of this SE, Brunswick addressed this concern by stating in the RAI 2 response (Reference 6) that "In the future, if a fuel design beyond ATRIUM 11 is introduced, the gap conductance will either be justified to be sufficiently similar to ATRIUM 11 or a new gap conductance sensitivity will be performed."

Based on the similarity **[[ ]]**, inclusion of the important physics relevant to ATWS-I, close agreement of the RAMONA5-FA ATWS-I results to measured BWR stability data, and **[[ ]]** of the stability results under most scenarios to variations in gap conductance, the NRC staff finds that the fuel rod heat transfer model, including the gap conductance model, is acceptable for use in the ATWS-I analyses.

3.5.2.4.7 Radial Power Deposition Distributions in Fuel Pellets

The licensee's methodology determines the radial power distribution within fuel pellets using the **[[ ]]**. The NRC staff has reviewed the methodology and determined that it provides the needed accuracy for calculating the radial power distribution in fuel pellets, including **[[ ]]** Therefore, the NRC staff finds the radial power distribution methodology to be acceptable.

3.5.2.5 Thermal-Hydraulic Model

The RAMONA5-FA ATWS-I methodology described by the licensee utilizes a **[[ ]]** TH model comprised of **[[ ]]**. A description and the NRC staff evaluation of the thermal-hydraulic model is given in the following subsections.

3.5.2.5.1 General Description of the System Considered

The general system modeling in the submittal consists of nine main components as shown in Figure 2 and is identical to the vessel methodology in the RAMONA5-FA LTSS methodology.
[[

]]

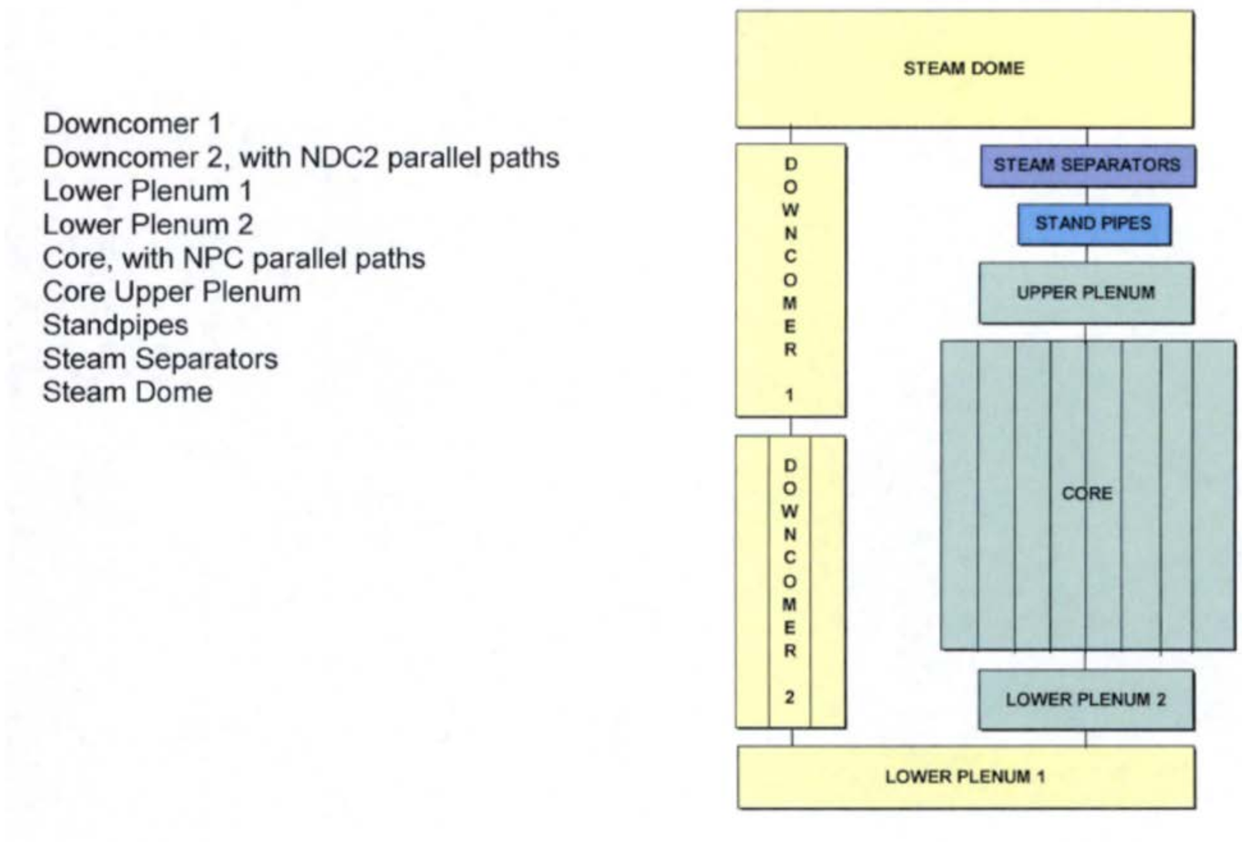

Figure 2: Loop Parts in the Vessel Hydraulics Model (Figure 5-3 (Reference 50))

Accurate modeling of the pressure losses and flow inertia in the vessel flow path is important for correctly determining flow rates and other core parameters during ATWS-I events; this is especially true for in-phase (corewide) oscillations in which the total core flow rate experiences large time-dependent changes that become coupled to time-dependent flow rate changes in the surrounding components. Vessel flow inertia is particularly dependent on the recirculation pump model, which is evaluated in Section 3.5.2.7.1 of this SE; however, the pressure losses and flow inertia in the remaining vessel components are relevant to ATWS-I analyses as well.

Because Brunswick's methodology contains the same vessel hydraulics treatment as the RAMONA5-FA LTSS methodology, which is approved for use in LTSS stability analyses, including the analysis of in-phase oscillations where vessel pressure losses and flow inertia are important, these aspects of the vessel hydraulics model were not reviewed for the submittal, and the existing approval of these modeling aspects from the RAMONA5-FA LTSS methodology remains applicable. Furthermore, the RAMONA5-FA LTSS methodology is approved for two recirculation pump trip (2RPT) LTSS analyses, which is an identical event to ATWS-I 2RPT, except that ATWS-I 2RPT assumes failure to scram. Although this failure to scram allows for larger-amplitude oscillations, and therefore, larger oscillations in flow rate and other thermal hydraulic parameters in the vessel components, these conditions do not impose additional physical modeling requirements on the vessel thermal hydraulic methodology, and this vessel methodology remains suitable for ATWS-I 2RPT.

The ATWS-I TTWB event is similar to ATWS-I 2RPT in that both involve a dual recirculation pump trip; however, the TTWB event requires modeling of turbine stop valve closure and turbine bypass valve opening, which impacts the pressure response in the vessel, including pressure wave propagation. Additionally, for the TTWB event, the decrease in feedwater temperature due to loss of feedwater heaters must be modeled, and the vessel model must be able to accurately model the mixing of the cold feedwater with the saturated liquid leaving the steam separator and accurately transport this fluid through the downcomer and lower plenum to ensure proper timing and magnitude of core inlet temperature decrease during the event. Accurate calculation of the time-dependent core inlet temperature is necessary to correctly predict the oscillation onset timing and magnitude. If the water level falls below the feedwater inlet to the vessel (feedwater spargers), significant heating of the subcooled liquid feedwater and condensation of the steam in the downcomer may occur, which may affect the core inlet temperature behavior as well. Details on the steam line flow dynamics, recirculation pump model, jet pump model, steam separator model, and feedwater sparger condensation model are given in Sections 5.4 through 5.5.4 of the submittal and are evaluated later in this SE.

The core region consists of a number of parallel fuel assembly channels and **[[**
]] or bypass channel. The bypass channel accounts for the inter-channel flow (between channel boxes), as well as the flow through the internal water rods in each assembly. There are numerous leakage paths from the lower plenum to the bypass region. **[[**

]] To allow the NRC staff to confirm their understanding of the process for passing thermal hydraulic information from MICROBURN-B2 to RAMONA5-FA for ATWS-I analyses and ensure consistent solutions between the two codes during initial steady-state conditions, the licensee (Reference 6) provided a detailed description of the **[[**
]]. These inputs to the RAMONA5-FA ATWS-I methodology are the same as used in the approved RAMONA5-FA LTSS methodology. Both versions of the RAMONA5-FA code use these inputs to perform a thermal hydraulic calculation **[[**

]]

[[

]], the NRC staff finds that the

approach for determining the initial thermal hydraulic solution in the RAMONA5-FA ATWS-I methodology is acceptable.

Bypass flow, including water rod flow, constitutes a relatively small fraction of the total flow through the core, which limits its hydraulic impact during oscillations. However, direct gamma heating of the bypass flow may cause localized boiling in the bypass region, which may have a significant effect on the power level of neighboring fuel bundles due to neutronic feedback. Bypass voiding is most likely when stagnant or reversed bypass flow is experienced, which occurs at very low core flow rates due to the relatively large gravitational pressure head of the bypass liquid column.

Including **[[**

]]. Therefore, the NRC staff finds that modeling the bypass **[[** **]]** is expected to give conservatively high PCT results and is, therefore, acceptable for the ATWS-I analyses.

In its supplement (Reference 6), the licensee clarified the number of nodes used in each region of the vessel model under the base nodalization scheme, and these node numbers were **[[**
]] for the nodalization study provided in response to RAI 9 in the referenced document (Reference 52). The licensee clarified that the most limiting nodes in the model, **[[**

]].
Therefore, the NRC staff was able to confirm that the vessel nodalization provides sufficient fidelity for liquid and vapor transport in the vessel such that the system behavior, including PCT, is accurately predicted for ATWS-I events.

In its supplement (Reference 6), the licensee clarified that the aforementioned vessel nodalization sensitivity cases provided in the generic Framatome TR's RAI 9 were performed for the Brunswick sample problem. **[[**

]], showed good agreement with the measured stability data using the base vessel nodalization. Therefore, the NRC staff finds that the momentum-related effects associated with vessel nodalization are expected to be insignificant, and the base vessel nodalization used in the LAR is acceptable in this regard.

During in-phase oscillations, the coolant flow rate and void fraction in the primary circulation loop will oscillate along with the oscillations in the core. These time-varying thermal hydraulic quantities in the vessel will impact the recirculation loop momentum dynamics and may, therefore, impact the stability characteristics of the system. This impact is expected to be negligible for out-of-phase oscillations because the thermal hydraulic conditions outside the core remain essentially constant in this case.

The vessel nodalization is expected to have an additional effect with respect to the time-dependent core inlet subcooling. This is due to the effect of numerical diffusion on energy transport in the vessel liquid. Therefore, the licensee provided a plot of core inlet subcooling versus time during the Brunswick sample problem for each nodalization case. The time-dependent subcooling behavior was **[[**

]].

As a result of this small change in inlet subcooling, the **[[**

]] The relative insensitivity and **[[**
]] in both the failure-to-rewet time and PCT, **[[**
]] will consistently produce results more or less conservative than other nodalizations, lead the NRC staff to conclude that the base vessel nodalization as specified in the RAI 9 response is reasonable and acceptable for use in the RAMONA5-FA ATWS-I methodology.

3.5.2.5.2 Flow behavior

This section of the submittal describes the spatial discretization scheme used by the RAMONA5-FA ATWS-I thermal hydraulic core channel solution. **[[**

]]. This method is suitable for determining the flow behavior during normal conditions (i.e., upward flow through the bundle), as well as transient behavior such as periodic flow reversal expected during large-amplitude ATWS-I oscillations. Therefore, the NRC staff finds this spatial discretization scheme, which is the same as used in the RAMONA5-FA LTSS methodology, to be acceptable for ATWS-I applications.

3.5.2.5.3 Vapor Generation Rate

The nodal vapor generation rate in the submittal is calculated **[[**

]]. This model is the same as in the Monticello ATWS-I methodology and similar to the RAMONA5-FA LTSS model, except for the modifications to allow for the
[[

]]; therefore, these modifications from the previously approved methodology are appropriate. The NRC staff has reviewed the new model and finds that it acceptably models vapor generation during ATWS-I events.

3.5.2.5.4 Review of Section 5.3.4 – Mass Conservation

The licensee's methodology solves separate liquid and vapor mass conservation equations, using **[[**

]]

Large-amplitude oscillations may exhibit any of the following flow scenarios: co-current upward flow (liquid and vapor both flowing upward), co-current downward flow (liquid and vapor both flowing downward), and counter-current flow (liquid and vapor flowing in opposite directions). The NRC staff reviewed the mass conservation model and finds that it properly accounts for all of these possible flow scenarios. This, in addition **[[**

]] and its ability to give realistic behavior at or near fully-voided conditions, has led the NRC staff to find that the mass conservation model in the submittal is acceptable for ATWS-I analyses.

3.5.2.5.5 Energy Conservation

The licensee's methodology uses a **[[**

]]

Since scalar quantities such as enthalpy are defined at the center of control volumes, and vector (or directional) quantities such as mass flow rate and velocity are defined at the edges of control volumes, the energy balance that includes energy entering or leaving through each edge of the control volume uses different scalar cell indices depending on the direction of flow through each control volume edge. In principle, these directions can be different at the bottom and top edge of each volume. The NRC staff finds that the energy equation formulation properly accounts for all possible combinations of flow directions for the bottom and top edges by making suitable adjustments to the enthalpy index for the donor cell scheme in calculating the rate of energy flow in or out of the control volume for each phase. Thus, the energy balance is properly conserved for any flow situation for both the liquid and vapor phases.

The licensee's methodology differs in the implementation of the overall core energy balance by including an **[[**

]]. The licensee supplemented the LAR with additional information (Reference 6) on the behavior of the **[[** **]]** during postulated ATWS-I events. The licensee provided a plot of **[[**

]]. The NRC staff expects that this effect would have a small impact on the ATWS-I results. Therefore, the NRC staff finds that the implementation of the **[[** **]]** is acceptable.

3.5.2.5.6 Modeling **[[** **]]**

Section 5.3.6 of the submittal discusses the approach for determining **[[**

]] the licensee supplemented additional information to the LAR (Reference 6). This information justified that **[[**

]] The NRC staff has reviewed the supplement and has concluded that the **[[** **]]** was developed using reasonable and mildly conservative assumptions.

The NRC staff has reviewed the **[[** **]]** and finds it to be acceptable. **[[**

]]

3.5.2.5.7 <u>**[[** **]]** Conservation</u>

Brunswick's methodology uses **[[** **]]** to account for flow inertia and acceleration terms and their effect on the time-dependent pressure drops. **[[**

]]. However, the NRC staff finds this implementation acceptable for the reasons given in Section 3.5.2.5.4 of this SE.

[[

]] With respect to momentum conservation, the basic phenomena and modeling requirements remain the same for large-amplitude oscillations characteristic of ATWS-I, and the **[[**
]] and acceptable for this use.

Special treatment is provided in the submittal (the same as in the RAMONA5-FA LTSS methodology) to calculate the pressure response due to valve closures in the steam line, which is relevant for ATWS-I TTWB events. Details of this special treatment are given in Section 5.4 of Attachment 14 to the LAR and evaluated in Section 3.5.2.6 of this SE. Since the pressure waves dissipate rapidly once they reach the larger volumes of the vessel, this treatment is not necessary for the vessel and core regions. Therefore, the **[[ ]]** is acceptable for use in the vessel and core regions for the reasons stated above.

The NRC staff also evaluated the acceptability of the **[[**
]]. In a BWR assembly, the liquid and vapor phases will, in principle, travel at different velocities; these velocities depend on a mass and momentum balance for each phase separately, as well as on mass and momentum exchange between the phases. **[[**

]]

In methodologies such as the one described by Brunswick in the submittal, **[[**

]] Based on this finding, which was obtained under realistic nonlinear ATWS-I analysis scenarios for multiple plant configurations and operating conditions, the NRC staff finds that **[[**

]] is acceptable.

Because the licensee's methodology uses **[[**
]], and because this approach remains suitable for large-amplitude ATWS-I oscillations, including the ability to model reversed and counter-current flow and the demonstrated conservatism of the single-momentum-equation approach, the NRC staff finds the momentum conservation model in the submittal to be acceptable.

3.5.2.5.8 <u>Pressure Calculation</u>

Section 5.3.8 of Attachment 14 to the LAR describes the methodology for calculating the time-dependent **system pressure, [[**

]] Therefore, the NRC staff finds the pressure calculation methodology to be acceptable.

3.5.2.5.9 Steam Dome Equations

The licensee described an **[[**

]] Therefore, the NRC staff finds the steam dome model acceptable.

3.5.2.5.10 Recirculation Flow

The licensee described an **[[**

]]

Both the TTWB and 2RPT ATWS-I event scenarios involve a dual recirculation pump trip, which is also true for the LTSS analysis scenarios for which the RAMONA5-FA LTSS methodology has been previously approved. **[[** **]]** therefore, the NRC staff finds the recirculation flow model acceptable for ATWS-I applications.

3.5.2.5.11 Constitutive Equations

The NRC staff reviewed several aspects of the constitutive equations provided in the submittal.

3.5.2.5.11.1 Friction and Two-Phase Friction Multiplier

In Section 5.3.11.1 of Attachment 14 to the LAR, the licensee described the same friction factor correlations as the RAMONA5-FA LTSS methodology and uses the **[[** **]]**, which is also available in the RAMONA5-FA LTSS methodology. Implementation in the licensee's methodology, **[[** **]]** via additional accounting for reverse and counter-current flow. This change provides the proper treatment of the reversed and/or counter-current flow experienced during large-amplitude ATWS-I oscillations. The NRC staff finds that the implementation of the friction and two-phase multipliers is acceptable and reasonable for this application.

3.5.2.5.11.2 Local Pressure Loss Models

In Section 5.3.11.2 of Attachment 14 to the LAR, the licensee described essentially the same local pressure loss model as the RAMONA5-FA LTSS methodology, with the primary exception being that the licensee's methodology accounts for the possibility of reversed flow. As in the previous section, the NRC staff finds that this implementation, including treatment of reversed flow, is appropriate and acceptable for ATWS-I applications.

3.5.2.5.11.3 Abrupt Contraction/Expansion Pressure Change Model

In Section 5.3.11.3 of Attachment 14 to the LAR, the licensee described a similar abrupt contraction/expansion reversible pressure change model as the RAMONA5-FA LTSS methodology, except for additional accounting for reverse and counter-current flow compared to the RAMONA5-FA LTSS methodology. This change is necessary for properly treating the reversed and/or counter-current flow experienced during large-amplitude ATWS-I oscillations. The NRC staff finds this treatment of reversed and counter-current flow to be acceptable for this application.

3.5.2.5.11.4 **[[** **]]** Correction

In Section 5.3.11.4 of Attachment 14 to the LAR, the licensee described **[[**

]] However, the licensee stated in the submittal that **[[**

]] The NRC staff concurs that the effect of this term **[[**

]]

Therefore, the NRC staff finds that **[[** **]]** is acceptable for ATWS-I calculations.

3.5.2.5.11.5 Thermodynamic Steam-Water Properties

In Section 5.3.11.5 of Attachment 14 to the LAR, the licensee's methodology uses the IF97 Properties of Water and Steam[6] as a function of enthalpy and pressure, the same as used in the Monticello ATWS-I methodology. This is an improvement over the RAMONA5-FA LTSS methodology, **[[**

]] This provides the best available representation of

[6] The International Association for the Properties of Water and Steam, Revised Release on the IAPWS Industrial Formulation 1997 for the Thermodynamic Properties of Water and Steam, Lucerne, Switzerland, August 2007.

thermodynamic fluid properties at the full range of possible conditions during ATWS-I, and therefore, the NRC staff finds this implementation acceptable.

3.5.2.5.11.6 **[[** **]]** Correlation

In Section 5.3.11.6 of Attachment 14 to the LAR, the licensee provided information to determine **[[**

]]

As a result, the NRC staff finds the **[[** **]]** to be acceptable for the purpose of establishing the parameters of **[[** **]]**

3.5.2.5.12 Numerical Integration Techniques

The RAMONA5-FA LTSS methodology utilizes **[[**

]] Furthermore, the benchmark results presented in the submittal demonstrate the numerically stable and robust performance of the methodology up to and including large-amplitude oscillations typical of ATWS-I analyses. Therefore, the NRC staff finds the numerical integration technique acceptable.

With respect to core axial nodalization, the NRC staff's experience has shown that the calculated DR of thermal hydraulic oscillations in other codes may be significantly artificially dampened by numerical errors (numerical diffusion) in the underlying equations, and this effect

may be strongly influenced by factors such as timestep size and spatial discretization scheme. **[[**

]] In addition, the NRC staff has identified that the course nodalization associated with a 25 uniform axial nodalization scheme may lead to significant error in the oscillation DR due to an insufficiently spatially resolved axial void profile, particularly near the bottom of the channel, and a resulting effect on neutronic feedback and oscillatory behavior.

The licensee supplemented the LAR with additional information (Reference 6) to justify that the axial nodalization scheme used in the licensee's methodology provides sufficient numerical fidelity to accurately represent the stability behavior for ATWS-I. The referenced documentation discusses three possible effects of numerical diffusion with regard to stability. The first effect is the kinematic diffusive "spreading" of solution variables over time as fluid moves along the channel. The information provided by the licensee stated that this effect is **[[**

]]. The NRC staff finds this reasoning to be logical and consistent with the theoretical formulation of the solution methodology presented in the submittal.

[[

]] has only a small or minimal impact on the calculated stability behavior for the RAMONA5-FA ATWS-I methodology.

The third effect of diffusion discussed by the information provided by the licensee is related to the momentum formulation itself with respect to the effect on the momentum components of the density head and axial distribution of friction resulting from increased axial attenuation of density waves. The referenced documentation stated that the improved axial resolution of void fraction gradients afforded by decreased node size, as proposed previously by the NRC staff and contractors, may be of more importance than the kinematic effect of numerical diffusion for which the Courant number plays a direct role. **[[**

]]

The discussion provided in the referenced documentation (Reference 52) supports a conclusion that the core axial nodalization scheme and associated numerical errors would be expected to

be relatively small for ATWS-I applications. However, to confirm the effect that nodalization may have on the code results for ATWS-I applications, the NRC staff reviewed the results of the nodalization sensitivity study provided by the licensee. In this study, the nodalization was increased from **[[ ]]** axial nodes in the core, which is the proposed value for the methodology, to **[[ ]]** axial nodes in the core. The NRC staff reviewed the licensee's approach and determined that the nodalization increase was performed in a suitable manner, **[[ ]]**; this ensures a consistent approach for determining the neutronics and thermal hydraulics initial conditions to ensure that the conclusions of the RAMONA5-FA nodalization study are valid.

The finer nodalization resulted in increased DRs for all cases included in the nodalization study – namely, the KATHY stability tests, the linear reactor benchmarks, the Oskarshamn-2 nonlinear benchmark, and the Brunswick TTWB sample problem included in the nodalization study. For the KATHY linear stability tests and the linear reactor benchmarks, the DR increased by amounts ranging from **[[ ]]** when comparing the finer nodalization results to the base nodalization results. This resulted in a clear average bias toward overprediction of the DR when considering all cases. The effect of nodalization on frequency was very small (approximately +/- 0.01 Hz change compared to the base nodalization case). For the Oskarshamn-2 nonlinear benchmark and the Brunswick sample TTWB problem, the DR **[[ ]]** The larger growth rate also led to earlier failure to rewet by approximately 20 seconds in the Brunswick TTWB sample problem.

After failure-to-rewet, the time-dependent PCT values appeared to be **[[ ]]** °C on average for the finer nodalization case compared to the base case. As a result, the maximum PCT throughout the event was **[[ ]] °**C higher with the finer nodalization for this problem. However, the NRC staff finds that this difference is likely due to the finer nodalization case reaching failure to rewet earlier than the base case, allowing failure to rewet to extend to lower elevations on the hot rod before the oscillations are suppressed, compared to the base case. These lower elevations would likely correspond to higher average LHGR values. Even if failure-to-rewet did not extend lower in the finer nodalization case, the smaller node sizes mean that the limiting node in the base nodalization case is split into two nodes in the finer nodalization case, and the lower of these two finer nodes would have a slightly higher LHGR than the larger node overlapping this location in the base case. The NRC staff suspects a cause for the higher PCT **[[ ]]** Therefore, the NRC staff finds that finer nodalization does not intrinsically cause higher PCT in the failure-to-rewet regime. Therefore, no penalty or added conservatism is necessary to account for this apparent increase in PCT for finer nodalizations due to the modest nature of the increase in PCT, inherent conservatisms in the methodology, and the lack of evidence that finer nodalization would capture new phenomena, which could have a significant impact on the PCT.

Although the finer nodalization resulted in greater instability – faster oscillation growth and earlier failure to rewet – in all linear and nonlinear analysis cases, the NRC staff finds that the "base nodalization" of **[[ ]]** in the core is acceptable because it gives the most consistent and non-biased overall agreement with the measured DR values across the various stability benchmarks. Further rationale for the acceptability of this nodalization scheme was provided by the licensee – namely, **[[**

]]

The NRC staff finds that the "base nodalization" of **[[ ]]** nodes in the core is acceptable for use in the RAMONA5-FA ATWS-I methodology. This determination is based primarily on the good agreement of the RAMONA5-FA ATWS-I methodology with measured stability data across the broad range of experimental conditions when using **[[ ]]** nodes in the core, compared to the **[[ ]]** As discussed previously in this section of this SE, the NRC staff also considered potential sources of error due to numeric diffusion, and determined that they would not be significant for the RAMONA5-FA ATWS-I methodology.

3.5.2.6 Steam Line Flow Dynamics

As discussed in Section 3.5.2.5.7 of this SE, the licensee's methodology uses **[[ ]].** However, a special model for the steam line is included in the licensee's methodology (identically to the RAMONA5-FA LTSS methodology) to calculate the propagation of pressure waves only within the steam line. For ATWS-I, this is relevant for calculating the pressure response after turbine valve closures following a turbine trip, as well as the pressure response following safety relief valve (SRV) closure and reopening that may occur during the oscillatory phase of the TTWB event.

[[ ]] The NRC staff reviewed this model and finds it to be a logical and acceptable method for determining pressure response in the steam line and vessel. This is primarily because, as discussed in Section 3.5.2.5.7 of this SE, **[[ ]]**. However, no detail is provided in

this section regarding how the steam line modeling is verified to capture reasonable behavior during the ATWS-I event for Brunswick. This is highly dependent on plant-specific configurations, closure time, and setpoints, so the NRC staff verified (see Section 3.5.5.2) that the resulting behavior (e.g., flow rates through the valves and pressure drop across the steam line(s)) from the steam line model is reasonably representative of expected Brunswick specific behavior during an ATWS-I event.

The NRC staff finds that the steam line flow dynamics model provides an accurate and realistic approach for calculating pressure response during ATWS-I events, including TTWB, and therefore, the NRC staff finds this model acceptable.

3.5.2.7 <u>Special Models</u>

3.5.2.7.1 <u>Recirculation Pump Model</u>

The recirculation pump model determines the relationship between pump rotational speed, pump torque, pump flow rate, and pump head. These define the steady state operating characteristics of the recirculation pumps, as well as their transient behavior. The primary relevance to stability analyses is in determining the recirculation pump coastdown behavior after a recirculation pump trip, as well as the recirculation pump inertia, which has an important impact on the growth rate and limit cycle amplitude of global flow oscillations. Note that the effect on regional flow oscillations is much smaller, as the total core flow rate remains relatively constant in that case.

The recirculation pump model in the submittal is identical to the model in the RAMONA5-FA LTSS methodology, which was approved for LTSS analyses, including the case of in-phase oscillations. The NRC staff has reviewed these models for the submittal and finds that the models include all necessary physics and remain acceptable for instability events up to and including large-amplitude limit cycle oscillations.

3.5.2.7.2 <u>Jet Pump Model</u>

Unlike the recirculation pump model, **[[**

]]. However, as with any pressure term in the primary loop, the calculated pressure head may affect the transient behavior during rapid pressure changes (such as immediately following a turbine trip), as well as affect the stability behavior particularly during global oscillations.

The jet pump model in the submittal is identical to the model in the RAMONA5-FA LTSS methodology, which was approved for LTSS analyses, including the case of in-phase oscillations. The NRC staff has reviewed these models for the submittal and finds that the models include all necessary physics and remain acceptable for instability events up to and including large-amplitude limit cycle oscillations.

3.5.2.7.3 <u>Steam Separator Model</u>

The modeling of the steam separator, in particular, its flow inertia, as well as the flow rate of vapor leaving the circulation loops and entering the steam dome above the coolant level (known as carry-under) – may have a significant effect on the stability characteristics of the reactor system. Flow inertia has a particularly strong impact for corewide (in-phase) oscillations.

The steam separator model in the submittal is identical to the model in the RAMONA5-FA LTSS methodology. This model determines the steam separator flow inertia based on **[[**

]]

The flow conditions and behavior of the steam separator follow the same physical principles and exhibit the same general characteristics under ATWS-I conditions as under smaller amplitude LTSS oscillation conditions, and therefore, the NRC staff finds that the steam separator model, which was previously approved for the RAMONA5-FA LTSS methodology, is applicable and acceptable for ATWS-I applications in the submittal.

3.5.2.7.4 Feedwater Sparger Condensation Model

When the water level in the vessel downcomer is below the level of the feedwater inlet (feedwater spargers), significant heating of the subcooled feedwater liquid, as well as condensation of the saturated steam, may occur as the liquid flows downward through a steam environment. This can affect the core inlet temperature, as well as the system pressure.

The submittal describes the same model as the RAMONA5-FA LTSS methodology to model the condensation rate as a function of **[[**

]]

Once the water level falls below the feedwater spargers, the nature of this condensation phenomenon is the same during ATWS-I as it is during other events currently approved for analysis using the RAMONA5-FA LTSS methodology. Additionally, the model provides physically reasonable and realistic relationships with physical parameters. Therefore, the NRC staff finds the feedwater sparger condensation model acceptable for use in the submittal for ATWS-I analyses.

3.5.2.7.5 Dryout and Rewetting Model

The prediction of dryout and possible subsequent rewet of the hot rod is of primary importance to ATWS-I analyses due to the dramatic increase in PCT associated with sustained dryout. Under sufficiently high cladding-to-coolant heat flux for a sufficient duration, all liquid in contact with the cladding surface evaporates, leaving only vapor in contact with the cladding surface ("dryout" conditions). Because vapor is much worse than liquid at conducting/convecting heat from the cladding surface, the temperature of the cladding (and also the fuel pellet) quickly increases after the onset of dryout. Due to the large, rapid changes in flow rate and thermodynamic quality of the coolant adjacent to the cladding surface during thermal hydraulic oscillations, there is a possibility that liquid will once again come into direct contact with the cladding surface ("rewet"), lowering the cladding temperature due to improved heat transfer. However, rewetting of the cladding surface becomes more difficult as the cladding temperature (and therefore, the evaporation capability) increases; this may lead to a runaway condition in which the liquid flow is no longer able to come into contact with the cladding surface for long enough to fully reverse the increase in cladding temperature. Under such a condition, the cladding temperature may "ratchet" up through multiple cycles of heatup and limited cooldown due to rewetting or experience a continuous increase in temperature due to loss of rewetting

ability. If the cladding temperature increase is not mitigated, very high cladding temperatures that might challenge the ATWS-I acceptance criteria may result. In this case, the cladding temperatures can only be brought down again by reducing the heat generation rate (power level) in the fuel; during ATWS-I, this is done either by increasing the average void fraction in the core (accomplished via water level reduction) or injection of soluble boron into the core (via the standby liquid control system).

Because of its strong impact on the PCT during ATWS-I events, the dryout and rewetting model in the submittal was one of the primary focuses of the NRC staff's review. The licensee's methodology provides the same fundamental approach as the Monticello ATWS-I methodology, but a fundamentally different approach than the RAMONA5-FA LTSS methodology to determine dryout and rewet of the hot rod surface. The RAMONA5-FA LTSS methodology **[[**

]] The wetting or dryout status of the cladding surface is used to determine the heat transfer regime (nucleate boiling, transition boiling, or film boiling heat transfer regimes), and heat transfer coefficients are applied correspondingly.

However, based primarily on analysis of the KATHY dryout/rewet test data presented in Section 6.5 of Attachment 14 to the LAR , the licensee concluded that a different modeling approach for determination of dryout and rewet behavior provided a better fit to the data under oscillatory conditions representative of ATWS-I. This model, similar to the one provided in the Monticello ATWS-I methodology, **[[**

]]

A thorough review of a similar dryout/rewet model was performed by the NRC staff in its review of the Monticello ATWS-I methodology. In that review, the NRC staff concluded that the model was acceptable for the plant-specific application for which the methodology was submitted. For Brunswick's submittal, the NRC staff reviewed the dryout/rewet model with particular focus on determining the acceptability of differences in the model relative to the Monticello ATWS-I methodology, as well as the applicability and acceptability of the model for use at Brunswick.

The NRC staff's review determined that the licensee's dryout/rewet model is largely similar to the dryout/rewet model used in the Monticello ATWS-I methodology, including **[[**

]]. Although the NRC staff's review and approval of the Monticello ATWS-I methodology was performed on a plant-specific basis, in its evaluation of the Monticello ATWS-I methodology dryout/rewet model, the NRC staff did not note any limitations or

shortcomings of the model that would specifically limit its use for Brunswick. For the submittal, the NRC staff further examined the model and determined that the experimental benchmarking, as discussed in Section 3.5.3.5 of this SE, covered a sufficiently broad range of representative ATWS-I conditions such that the model is acceptably applied to Brunswick.

Some differences were noted between the Monticello ATWS-I methodology and the licensee's dryout/rewet models, and these are evaluated in additional detail in the remainder of this section. These differences include the **[[**

]]

[[

]] The NRC staff finds the **[[**
]] to be reasonable on a physical basis, and the strong agreement of the dryout/rewet model with the cyclic dryout/rewet behavior observed in the KATHY experiments leads the NRC staff to find this revised model to be acceptable.

The NRC staff examined the addition of a **[[** **]]** to the licensee's model relative to the Monticello ATWS-I methodology based on consideration of the flow conditions under which the dryout/rewet model was derived. **[[**

]] The licensee supplemented the LAR (Reference 6) to justify the applicability **[[**

]].

Using the same the RAI 5 response to the generic Framatome ATWS-I TR (Reference 52), the licensee indicated that **[[**

]]

Specifically, the licensee applied the **[[**
]] to the base critical power reduced over model (CPROM). As depicted in Figures 8 through 14 in Appendix A of the supplement (Reference 6), the **[[**

]] The NRC staff finds that this **[[**

]], the NRC staff finds that the dryout/rewet model with **[[** **]]** provides an acceptable representation of dryout behavior for the full range of quality and void fraction conditions expected during ATWS-I.

The submittal describes a process for fitting the CPROM model to steady-state CPR data, which differs from the **[[** **]]** process used for the Monticello Nuclear Generating Plant (Monticello) ATWS-I methodology. This new process was determined by the NRC staff to be acceptable based on the evaluation provided in Section 3.5.2.11 of this SE.

The licensee provided additional information in the supplement (Reference 6) on how the fitting parameters for the dryout/rewet model were determined from measured data, particularly when direct experimental validation for each parameter was not possible or not available. Using the same response to RAI 2a for the Framatome generic ATWS-I TR (Reference 52), the licensee described the fitting process in detail, including a combination of **[[**

]] in a consistent, logical, and well-defined fashion to provide the most accurate and acceptable prediction of both steady state and transient behavior applicable to ATWS-I.

The NRC staff has reviewed the dryout/rewetting model in detail, including evaluating the applicability of the model to Brunswick, as well as evaluating the differences relative to the similar model in the Monticello ATWS-I methodology. The NRC staff has concluded that the model is acceptable because it is based on realistic physical principles, exhibits close agreement to measured CPR data under steady state conditions, and agrees closely with measured data under transient conditions for a wide range of operating conditions, which reasonably encompass the expected range of conditions expected to occur during postulated ATWS-I events at Brunswick.

3.5.2.8 Plant Control and Protection Systems

The NRC staff reviewed the plant control and protection systems methodology provided in Section 5.6 of the LAR, including the implementation of:

- Pressure control system consisting of turbine control, bypass valve, and safety relief valve (SRV),
- Plant protection systems (PPS), including recirculation pump trips, and
- Feedwater control system for water level control.

The pressure control system model is required to accurately model the system pressure response following a turbine trip and possible cycling of the SRVs during a TTWB event. Modeling of the recirculation pump trip function of the PPS is relevant for both the TTWB and 2RPT ATWS-I events to determine realistic timing of the recirculation pump trip and associated core flow rate reduction. Modeling of the feedwater control system is relevant for both the TTWB and 2RPT ATWS-I events to allow the water level to automatically adjust to the setpoint value by adjusting the feedwater flow rate, both during the initial event progression, as well as after the operator action to reduce the water level setpoint.

These models are essentially the same as in the RAMONA5-FA LTSS methodology, with the primary exception being that a software change was made to the RAMONA5-FA code to model the Brunswick manual operator actions to reduce the water level during an ATWS-I event. This software change allows the software user to specify the start time of operator actions, as well as a setpoint to which the water level will be reduced, thereby enabling the software code to accurately model the Brunswick event, including the existing manual operator actions developed for the ATRIUM 10 MELLLA+ ™ LAR (Reference 37). The model assumes the feedwater pumps trip at the specified start time and calculates the coast down behavior of the feedwater pumps, after which the feedwater controller maintains the water level at the new lower level based on the user defined setpoint.

The NRC staff reviewed these models and finds that these models are acceptable because they realistically and adequately represent the plant control and protection systems behavior during postulated ATWS-I events. In addition, there is no change in the ATWS-I related operator actions from the currently approved operations.

3.5.2.9 <u>Numerical Time Integration</u>

The numerical scheme used for time integration (time marching) in the neutron kinetics, thermal hydraulics, and fuel rod thermodynamics equations has a significant effect on the numerical robustness and accuracy of the solution. In particular, the large temporal and spatial gradients associated with rapidly changing conditions within the core and vessel during ATWS-I oscillations increase the potential for large numerical errors in the solution, particularly during sharp changes in the solution such as the expected changes due to the dryout and rewet phenomena. Such errors may lead to results that depart significantly from reality and prevent accurate determination of the system response with respect to the ATWS acceptance criteria.

In the licensee's methodology, the neutron kinetics, fuel thermodynamics, and vessel hydraulics equations are solved separately, with **[[** **]]** integrate each of these three calculation domains. Enforcing **[[** **]]** in each domain

ensures numerical consistency, improves numerical robustness, and avoids loss of accuracy in the overall coupled solution.

As described in Sections 3.5.2.3 and 3.5.2.4 of this SE, the neutron kinetics and fuel thermodynamics equations are integrated **[[** **]]** in time. The most significant impact of this is **[[**

]].

As indicated in Section 3.5.2.5.12 of this SE, **[[**

]]. The licensee supplemented the LAR with a discussion and sensitivity studies regarding vessel and core nodalization (Reference 6). The supplement adopted the same RAI 9 and RAI 8 responses for Framatome's generic ATWS-I TR (Reference 52). The NRC staff's evaluation of this additional information are discussed in Sections 3.5.2.5.1 and 3.5.2.5.12 of this SE.

The licensee also supplemented the LAR (Reference 6) to provide additional details to determine the acceptability of the timestep control scheme and the values used for the benchmarks and sample problem in the submittal. Using the same RAI 12 response for Framatome's generic ATWS-I TR (Reference 52), the licensee described the timestep control parameters specified in the input file. The NRC staff reviewed these parameters and determined that they provide adequate capability to control the numerical timestep size to ensure robustness and accuracy of the solution. Importantly, the NRC staff determined from the RAI response that the same timestep control parameters were used for all benchmark cases and the sample problem in the submittal. This ensures consistency and validity of the methodology across all benchmarks. **[[**

]].

Results of a timestep sensitivity study were provided in the licensee's supplement by varying the values of the timestep control parameters for the Brunswick sample problem. These sensitivity cases, which modeled timestep size differences that varied by up to **[[** **]]** throughout the event exhibited only minor differences in oscillation growth rate and failure-to-rewet times (no more than 10 seconds across the sensitivity cases). The failure-to-rewet time varied in an unpredictable fashion with no clear trend as a function of the timestep control parameter values. Because of this lack of a clear trend, and because the "base" timestep control values demonstrated good agreement with measured data across all benchmarks documented in the submittal, the NRC staff finds the base timestep control values to be acceptable for use in RAMONA5-FA ATWS-I applications.

3.5.2.10 Calculation Procedure

A portion of the ATWS-I analysis methodology described in the submittal is not captured within the RAMONA5-FA code or various assessments of the performance of its constituent models and correlations. In order to perform an ATWS-I analysis, an analyst must follow prescribed steps in order to ensure that the plant-specific analyses are performed in a manner consistent with the assumptions within the code and the intent of the methodology in demonstrating

regulatory compliance. A discussion of the key guidance provided in the submittal for performance of ATWS-I analyses is provided in the following subsections.

3.5.2.10.1 Statepoint Definition

Section 8.0 of ANP-3694P (Attachment 14a of the LAR (Reference 1)) defines the procedure to be used for plant-specific ATWS-I analyses using the licensee's methodology. The procedure defines the characteristics of the statepoint to be analyzed. This includes **[[**

]] is described in the submittal and the responses to NRC RAI 15 and RAI 16 from the generic ATWS-I methodology review (Reference 52) as being established and justified by the licensee. The NRC staff finds it is acceptable that **[[**

]] The NRC staff has reviewed the proposed statepoint definition and finds that it is specified in an acceptable manner for ATWS-I and is consistent with the approach for MELLLA+™ applications that has previously been reviewed and approved by the NRC staff (Reference 59).

3.5.2.10.2 Overview of the Analysis Procedure Defined in the LAR Submittal

Details of the calculation procedure and its acceptance criteria are described in Section 8.0 of Attachment 14 to the LAR submittal (Reference 1). Note that ATWS-I plant-specific calculations are performed for the first cycle planned for utilization of the MELLLA+™ operating domain, and subsequently, new calculations are performed only when a new fuel is introduced or another change is implemented that requires a new license amendment. Therefore, the calculation procedure must ensure that the ATWS-I results bound cycle-specific variations such that the analyses remain applicable for all MELLLA+™ cycles that would be allowed by implementation of the proposed license amendments.

The licensee supplemented the LAR (Reference 6) to provide a revised calculation procedure that may be used to determine that the calculated margin from the non-cycle-specific analysis is sufficient to ensure applicability to all MELLLA+™ cycles. In the same supplement, the licensee provided additional information to ensure that the modeling assumptions remain appropriate

when considering their effect on the time of oscillation onset. This information can be found in the responses provided by Framatome in responses to RAI 14 and RAI 15 on the generic ATWS-I TR (Reference 52).

The licensee's calculation procedure relies on the evaluation of two acceptance criteria: (1) that PCT remains below 2,200 °F if failure to rewet occurs, or (2) **[[**

]]

The revised calculation procedure provided by the licensee in the document referenced in the LAR supplement (Reference 52) involves **[[**

]]

The revised calculation procedure clarifies that "sufficiency," in the context of "sufficient margin to failure to rewet," refers to satisfying either of the two acceptance criteria listed above; in other words, the margin is sufficient if **[[** **]]** or if failure to rewet occurs and the PCT remains below 2,200 °F. In the proposed methodology, if the margin is sufficient based on these criteria, no further TTWB analyses are required, as discussed in Step 3.b. If the margin is insufficient, Step 3.c must be followed.

The NRC staff finds that the definition in the revised calculation procedure **[[**

]]

Based on the NRC staff's concerns that cycle-specific variations may potentially lead to large changes in PCT even if failure to rewet has occurred, the NRC staff believes that all transition cycles (i.e., up to two transition cycles) occurring during EFW operation must be addressed, either by explicit analysis as in Step 3.a or by appropriate justification, to provide adequate assurance that the most limiting cycle is analyzed. This latter justification was provided by Brunswick (see Section 3.5.5.3 of this SE for further discussion).

The revised calculation procedure applies a **[[**

]], based on the NRC staff's experience this is expected to provide sufficient added conservatism to compensate for the possibility of the most limiting point not occurring precisely at one of the **[[** **]]** analyzed.

The revised calculation procedure employs conservative assumptions with respect to the time of oscillation onset. **[[**

]] The licensee also discussed that the assumed feedwater (FW) temperature reduction rate in all cases must conservatively bound the expected Brunswick behavior. Furthermore, the definition of exposure points and transition cycles to be analyzed, as well as **[[** **]]** discussed above, provides additional assurance that the results remain bounding when considering cycle- and exposure-specific variations in oscillation onset time.

The NRC staff finds that the procedure described in the revised calculation procedure in the Framatome response to RAI 15 on the generic ATWS-I TR (Reference 52) submitted as a supplement to the LAR (Reference 6), provides an acceptable means of obtaining reasonable assurance that a PCT of 2,200 °F will not be exceeded for any cycle and exposure point during MELLLA+^{TM} operation. The licensee also provided justification that the ATRIUM 11 ATWS-I analysis performed for Brunswick contains sufficient conservatism to bound cycle-specific variations in neutronic characteristics due to different core designs, as discussed in Section 3.5.5.3 of this SE.

3.5.2.10.3 Selection of the Limiting Event

The licensee proposed to perform Steps 3 through 3.c for the TTWB event and, as discussed in Step 4 – 2RPT, analyses are only required if TTWB did not experience failure to rewet. The rationale for this, as understood by the NRC staff that the FW heaters remain active throughout

the 2RPT event but not in the TTWB event, resulting in higher FW temperatures, and therefore, lower average power, less severe oscillations, and lower PCT values than in the TTWB event.

In the ATWS-I analyses, the time-critical operator action is defined with respect to the time of identification of an ATWS. The 2RPT event may involve manual initiation of scram by the operator if the RPV level does not increase to the high level turbine trip setpoint (compared to automatic scram initiation in the TTWB event), and subsequently, a later time when the ATWS-I event is recognized and mitigation actions taken. Therefore, the analyses must justify the additional time that may occur prior to operator action to mitigate the 2RPT ATWS-I event. This effective delay in operator actions relative to the initiating event allows additional time for the FW temperature to decrease during the 2RPT event, such that the FW temperature at the time of operator actions may potentially be less than that for the TTWB event. Furthermore, for plants with steam-driven FW pumps, the TTWB event may include a trip of the FW pumps prior to operator action, which may mitigate the consequences of the ATWS-I event (due to reduction in water level before action is taken). Consequently, the TTWB event may not be limiting.

The TTWB event, due to turbine valve closure, leads to a higher system pressure than the 2RPT event. Based on the NRC staff's experience, competing effects may exist that lead to an unclear relationship between system pressure and stability behavior, which is difficult to ascertain *a priori*. Because of these two concerns, the NRC staff concluded that the licensee must justify the operator action time and FW temperature assumptions for both the TTWB and 2RPT events, and must perform analyses for both events using these assumptions to determine which event is limiting with respect to PCT, when appropriate. However, the NRC staff acknowledges that the limiting event is expected to be primarily a function of the operator action and FW temperature assumptions used in the analyses, which remain the same regardless of operating cycle or exposure conditions. Therefore, there is reasonable confidence that the limiting event – 2RPT or TTWB, will remain the same across all cycles and exposure points for a plant configuration and set of assumptions.

As discussed in Section 3.5.5.3, the licensee provided results showing that the TTWB event is currently the limiting event for Brunswick. The NRC staff finds that only this event needs to be considered for future analyses if the plant configuration and assumptions remain the same. In its response to RAI 5 from the NRC (Reference 6), the licensee stated that it will reevaluate the limiting event if key plant parameters or analysis assumptions change. As a result, the limiting event for ATWS-I will be appropriately identified for any given set of Brunswick plant configuration and analysis assumptions.

3.5.2.10.4 <u>Boron Injection</u>

In addition to taking action to reduce water level, the reactor operators must initiate standby liquid control system (SLCS) boron injection within the time-critical action interval following identification of an ATWS. The effect of SLCS injection is to deliver borated water to the core, which provides sufficient negative reactivity to shut the core down. This shutdown is capable of terminating any oscillations, as well as limiting the impact of high core power on the containment heat load. However, the operator actions to reduce water level are expected to mitigate the oscillations well before the borated water reaches the core. **[[**

]], Section 8.0 of Attachment 14 to the LAR defines a **[[**

]] The NRC staff has reviewed this approach and determined that it will provide a **[[**

]] Therefore, the NRC staff finds the modeling of SLCS injection as defined in Section 8.0 of Attachment 14 to the LAR to be acceptable.

3.5.2.10.5 Calculation Procedure Conclusions

Based on the evaluations given above, the NRC staff finds the calculation procedure to be acceptable for its intended purpose.

3.5.2.11 Steady State Dryout Correlation CPROM

Dryout of a fuel rod, unless it is quickly followed by rewet, leads to very large increases in cladding temperature. Therefore, accurate calculation of the timing and location of dryout in the fuel bundles is of high importance in determining the PCT and has a strong effect on whether the ATWS acceptance criteria are met during postulated ATWS-I scenarios. The approach used to develop the models for dryout (and rewet) in the submittal is twofold. **[[**

]] dryout/rewet model described in Section 5.5.5 of Attachment 14 to the LAR and evaluated in Section 3.5.2.7.5 of this SE.

The CPROM correlation was previously presented in the Monticello ATWS-I methodology and was reviewed and accepted by the NRC staff for plant-specific application at Monticello. However, the correlation was developed from **[[**

]]

With the exception of stagnant or reversed flow, which may occur during ATWS-I, especially near the inlet of the bundle, the range of operating conditions shown above encompasses the expected ATWS-I conditions for Brunswick and is, therefore, suitable for use in the Brunswick ATWS-I analyses. **[[**

]] For these reasons, the NRC staff finds the use of the CPROM correlation within the dryout/rewet model in the submittal to be acceptable for ATWS-I analyses.

In addition to the **[[**

]] The Monticello ATWS-I methodology used **[[**

]] By contrast, Brunswick used **[[**

]] The NRC staff, therefore, finds **[[**

]] in the Monticello ATWS-I methodology, to be acceptable.

Because of the wide range of operating conditions over which the licensee's CPROM **[[** **]]** is validated, and because the modifications relative to the previously reviewed Monticello ATWS-I CPROM **[[** **]]** are reasonable and result in comparable or improved accuracy relative to the measured data, the NRC staff finds the CPROM **[[** **]]** in the submittal to be acceptable for use as the underpinning CPC for use in the licensee's **[[** **]]** model for limiting ATWS-I analyses at Brunswick.

3.5.3 <u>Code Assessment</u>

Following the review guidance provided in Chapter 15.0.2 of the SRP, the next area of review for transient and accident analysis methods focuses on assessment of the code. The associated acceptance criteria indicate that all models need to be assessed over the entire range of conditions encountered in the transient or accident scenarios. The review procedures provided in Section III of Chapter 15.0.2 of the SRP also indicate that the assessment of these models is commensurate with their importance and required fidelity. This assessment is generally performed by comparison of predicted results against both separate effects tests and integral effects tests.

Separate effects tests are generally used to demonstrate the adequacy of individual models and the closure relationships contained therein. Complementary to these types of tests are integral tests, which are generally used to demonstrate physical and code model interactions that are determined to be important for a full-size plant. The NRC staff evaluation of the individual elements of the code assessment suite provided in the submittal is presented in the following subsections.

3.5.3.1 <u>Test Suite and Acceptance Criteria</u>

The following acceptance criteria were proposed by the licensee for validation against measured data:

- Calculated void fraction **[[ ]]** of measured
- Calculated pressure drop **[[ ]]** of measured
- Calculated DR **[[ ]]** of measured with the exception that higher DRs above this range are considered acceptable
- Calculated oscillation frequency within **[[ ]]** of measured
- **[[ ]]**
- Acceptance criteria for nonlinear benchmarks given on a case-specific basis

These acceptance criteria ranges for void fraction, pressure drop, DR, and oscillation frequency correspond to the uncertainties determined in the prediction of these parameters in the TR containing the STAIF methodology (Reference 60) and the RAMONA5-FA LTSS methodology. As discussed in Section 3.5.4 of this SE, ATWS analyses are not required to explicitly account for modeling uncertainties as they are required for design-basis event analyses such as those for which STAIF and the RAMONA5-FA LTSS methodology are used. However, the NRC staff finds the approach of defining validation acceptance criteria based on relevant uncertainty bounds for the previously approved stability analysis methods STAIF and RAMONA5-FA to be acceptable because this demonstrates that the licensee's methodology has similar or not significantly greater modeling uncertainty than the previously approved stability methodologies.

The NRC staff finds the added stipulation that calculated DRs are allowed to be more than **[[ ]]** higher than measured to be acceptable for this application because higher DRs are conservative and **[[ ]]** in calculated-versus-measured DR was seen in the benchmarking to KATHY stability tests (Section 3.5.3.4 of this SE), indicating good predictive capability of the licensee's methodology across a wide range of conditions.

The suitability of the acceptance criteria for the pin-dependent CPROM term and for the nonlinear benchmarks are discussed as part of their own separate subsections later in this SE.

3.5.3.2 Benchmarking to Void Fraction Tests

As described in Sections 3.5.2.5.4 and 3.5.2.5.11 of this SE, the licensee's methodology uses the **[[ ]]** to determine the relationship between quality and void fraction in the fuel bundles. The **[[ ]]** was approved for use in the RAMONA5-FA LTSS methodology.

Section 6.2 of Attachment 14 to the LAR states that the following steady-state void fraction data sets were used to validate this correlation:

- FRIGG (314 test points)
- ATRIUM 10 KATHY **[[ ]]**
- ATRIUM 10XM KATHY **[[ ]]**

These data include a wide range of pressure, inlet subcooling, and mass flow rate conditions. The ATRIUM-10 and ATRIUM 10XM data include maximum void fractions of **[[ ]]** and **[[ ]]**, respectively. These data are the same as were used to validate a different void fraction correlation in the Monticello ATWS-I methodology. However, all data collected for pressures outside the range of **[[ ]]** were discarded for the current application. The NRC staff finds this acceptable because this pressure range encompasses the expected pressure during postulated limiting ATWS-I events. Additionally, **[[ ]]**. However, the remaining data used for validation, especially the ATRIUM 10XM data, which covers the broadest range of conditions, provides very good coverage of the range of expected operating conditions during ATWS-I. The calculated void fraction demonstrates good agreement with the measured data and includes benchmarking directly relevant to the specific geometric configuration of current fuel types (ATRIUM 10 and ATRIUM 10XM). The use of this correlation for the ATRIUM 11 fuel design is discussed in Section 3.5.5.1 of this SE.

The **[[ ]]** As discussed in Section 3.5.2.5.7 of this SE, the dynamic transient behavior, especially under rapidly changing conditions such as large-amplitude oscillations, may cause significant departure of the relative phase velocity behavior from the behavior under steady-state conditions, and this behavior is not directly validated by the steady-state void fraction tests.

The linear and nonlinear stability tests discussed in Sections 3.5.3.6 and 3.5.3.7 of this SE provide an integral validation of the overall code behavior, including the void fraction correlation. The calculated stability behavior (e.g., oscillation growth rate) is highly sensitive to the void fraction correlation due to its impact on the pressure drop response and density reactivity feedback; therefore, the close agreement of the licensee's methodology with measured linear and nonlinear stability data under a wide range of conditions provides additional assurance that

the void fraction correlation does not impose a significant nonconservative error trend or bias in the calculated results under expected oscillatory conditions during ATWS-I.

3.5.3.3 <u>Benchmarking to KATHY Pressure Drop Tests</u>

Results from the RAMONA5-FA ATWS-I code for pressure drop were validated against the following steady state pressure drop measurements:

- KATHY ATRIUM 10
- KATHY ATRIUM 10XM

Benchmarking against pressure drop data allows for validation of the total pressure drop calculated in the licensee's methodology under steady-state conditions. For single-phase flow, the total pressure drop depends directly on the single-phase friction factor (as well as the liquid density thermophysical correlation, which has low uncertainty). For two-phase flow, the total pressure drop depends primarily on the single-phase friction factor, two-phase friction multiplier, and void-quality correlation (which determines the density and velocity of the fluid). Because the void-quality correlation was directly validated by the steady-state void fraction benchmarks and shown to give good agreement, the pressure drop tests are particularly useful in validating the single-phase friction factor and two-phase multiplier.

The measurements include a broad range of pressure, inlet temperature, and mass flow rate. A total of **[[ ]]** was included in the validation, covering single-phase and two-phase conditions. For ATRIUM 10, the mean relative error for the single-phase (two-phase) data points was **[[ ]]**. For ATRIUM 10XM, the single-phase (two-phase) mean relative error was **[[ ]]**. Both the single-phase and two-phase tests (for both fuel types) demonstrate close agreement between calculated and measured total pressure drop with no observable trends, and the NRC staff finds that the licensee's methodology is well-validated for calculating pressure drop over a wide range of operating conditions. As with the **[[ ]]** discussed above, the calculation of pressure drop is an important parameter to correctly determine thermal hydraulic stability, and the close agreement with measured linear and nonlinear stability data discussed below provides added assurance that the single-phase friction factor and two-phase multiplier and their implementation for oscillatory transient conditions is accurate and acceptable for ATWS-I applications.

The applicability of this benchmarking to ATRIUM 11 fuel is discussed in Section 3.5.5.1 of this SE.

3.5.3.4 <u>Benchmarking to KATHY Stability Tests</u>

Results from the RAMONA5-FA ATWS-I code for single-assembly thermal hydraulic stability were compared to measured stability data in KATHY for the following fuel designs:

- ATRIUM 10 (**[[ ]]** test points)
- ATRIUM 10XM (**[[ ]]** test points)

These tests included stable conditions (DR less than one), as well as unstable conditions (DR greater than one). For the stable test points, the DR and resonance frequency were determined from analysis of noise in the output signals using well-established numerical techniques. For the unstable points, the DR and resonance frequency were determined from analysis of the coherent oscillation signals above noise level in the output data.

The benchmarking results show acceptable agreement between measured and calculated DRs, with the majority of the calculated DRs **[[ ]]**. Most of the points that have **[[ ]]** are near or above the stability boundary, and the RAMONA5-FA ATWS-I code **[[**

]] The mean error in calculated versus measured frequency is **[[ ]]**, with very few points exhibiting error larger than **[[ ]]**. The comparison of calculated to measured DR and frequency satisfies the acceptance criteria discussed in Section 3.5.3.1 of this SE and demonstrates the ability of the RAMONA5-FA ATWS-I methodology to accurately predict the channel thermal hydraulic stability behavior of ATRIUM fuel types. These tests provide an integral validation of the thermal hydraulic phenomena important for stability, including fluid mass, momentum, and energy transport, as well as constitutive relations such as the void-quality, friction factor, and wall heat transfer coefficients in the subcooled and two-phase nucleate boiling regimes.

The applicability of this benchmarking to ATRIUM 11 fuel is discussed in Section 3.5.5.1 of this SE.

3.5.3.5 <u>Benchmarking to KATHY Dryout/Rewet Tests</u>

The RAMONA5-FA ATWS-I code was benchmarked against KATHY dryout/rewet stability tests for the following fuel types:

- ATRIUM 9 (**[[ ]]** test points)
- ATRIUM 10XM (**[[ ]]** test points)

These experiments were reviewed and evaluated in the SE for the Monticello ATWS-I methodology and were found to provide a realistic representation of the thermal hydraulic behavior, including the impact of neutronic feedback, during large-amplitude oscillations characteristic of ATWS-I conditions up to and including failure to rewet. In particular, the inclusion of realistically simulated neutronic feedback allows significant inlet flow reversal to occur and promotes the occurrence of dryout at low elevations as expected for ATWS-I events.

The NRC staff examined Figures 6-6 through 6-20 in Appendix A to the licensee's RAI responses (Reference 6) and finds that the RAMONA5-FA ATWS-I methodology provides a reasonable and realistic agreement with the qualitative behavior of the KATHY dryout/rewet tests, including the onset and growth of oscillations, cyclic dryout and rewet, and eventual failure to rewet. Furthermore, for each test case, the **[[**

]]. As discussed in Section 3.5.2.7.5 of this SE, the **[[**

]], leads the NRC staff to conclude that **[[** **]]** in the RAMONA5-FA ATWS-I methodology provides a realistic and accurate phenomenological representation of the **[[** **]]**

These experiments were also used to **[[** **]]** using a process described in Appendix B in Attachment 14 to the LAR (Reference 1). **[[** **]]** was previously reviewed in the SE for the Monticello ATWS-I methodology and was found to **[[** **]]** for ATWS-I oscillatory conditions. **[[** **]]** relative to the Monticello ATWS-I methodology was **[[**

]] as described in Section 3.5.2.4.3 of this SE. The NRC staff reviewed the information provided and has determined that **[[** **]]** is acceptable and that the **[[** **]]** was developed from **[[** **]]**

The KATHY dryout/rewet experiments serve as an extension of the model validation described in the previous section and additionally provide validation of **[[**

]].

The applicability of this benchmarking to ATRIUM 11 fuel is discussed in Section 3.5.5.1 of this SE.

3.5.3.6 Benchmarking to Linear Reactor Stability Benchmarks

Section 6.6 of Attachment 14 to the LAR describes the benchmarking performed with the RAMONA5-FA ATWS-I code for linear reactor stability data for the following BWR plant events:

- **[[**

  ]]

These linear reactor stability benchmarks were also included in benchmarking suites for the approved RAMONA5-FA LTSS and STAIF methodologies. These events involved measured oscillations with a DR of approximately **[[**

]]. Therefore, similar to the KATHY stability tests, these benchmarks provide an integral validation of the fluid mass, momentum, and energy transport, as well as constitutive relations in terms of their impact on system stability. Specifically, these benchmarks are used to validate the prediction of the timing of stability onset and the oscillation growth rate, but not the prediction of dryout or rewet that could potentially occur during the later stages of postulated ATWS-I events.

Compared to the KATHY single-assembly stability tests, the linear reactor stability benchmarks also provide validation of the effect of neutronic feedback, including the effect on wall-to-fluid heat transfer as a function of space and time, on the overall system oscillation characteristics. Another key difference relative to the KATHY stability tests is that these benchmarks involve mixed cores with multiple fuel types other than ATRIUM 10 and ATRIUM 10XM. The licensee also supplemented the LAR (Reference 6) to provide additional information regarding the linear reactor stability benchmarks, including information on operating conditions and fuel types for each benchmark case. Using the same response provided by Framatome in response to RAI 6 on the generic ATWS-I TR (Reference 52), the licensee provided the requested list of fuel types and conditions present in each of the benchmarked cores. The licensee indicated that the majority of fuel-specific inputs were available from the past benchmarking for the STAIF and RAMONA5-FA codes. In some cases, some input for the **[[**

]]. The NRC staff reviewed the information that was inferred and finds that this inference was performed in an acceptable manner and that any differences between the inferred and actual values for these fuel types would be expected to have a minor impact on the results. For these benchmarks, all neutronic and thermal hydraulic data were taken directly from the benchmarking suites for the approved STAIF and RAMONA5-FA codes, with no additional neutronic or thermal hydraulic data required.

In the same document, the licensee also provided a table of operating conditions and power distribution information for each of the linear reactor benchmarks. The benchmarks consist of stability tests performed at off-rated conditions during reactor startup (**[[**
]]) or a stability event from off-rated conditions **([[**
]]). Although these operating conditions may not strictly correspond to the conditions that would occur during a 2RPT or TTWB ATWS event at each plant, the tests encompass a reasonably wide range of power, flow rate, inlet subcooling, outlet quality, and axial power shapes (including highly bottom-peaked axial power profiles) similar to those expected during the initial oscillation growth phase of ATWS-I events. The oscillation DRs and frequencies calculated by the RAMONA5-FA ATWS-I code were within the acceptance criteria listed in Section 3.5.3.1 of this SE for all **[[ ]]** linear benchmark cases, which included both regional and corewide oscillation modes, and the results showed no discernible trend with respect to operating conditions, fuel types, or other plant-specific differences that exist among the **[[ ]]** benchmark cases. Therefore, the NRC staff finds that the linear stability benchmarks provide good additional assurance beyond the single-assembly KATHY stability benchmarking that the code is able to predict the onset and initial growth rate of oscillations that would occur during postulated ATWS-I events.

3.5.3.7 Benchmarking to Non-Linear Reactor Benchmarks

Section 6.7 of Attachment 16 to the LAR (Reference 1) describes the benchmarking performed with the RAMONA5-FA ATWS-I code for the following two nonlinear stability events:

- Oskarshamn turbine trip with non-linear oscillation
- BWR A FW temperature transient with non-linear oscillation

Unlike the linear reactor benchmarks, the nonlinear reactor benchmarks provide direct validation of the system response during events leading up to reactor instability, as well as validation of the onset timing and growth of oscillations up to relatively large-amplitude (approximately **[[** **]]** in both benchmarks) before oscillation suppression via scram.

Limited details were given in Attachment 14 to the LAR (Reference 1) for the boundary conditions and assumptions used in the RAMONA5-FA ATWS-I methodology for these benchmarks. The licensee supplemented the LAR (Reference 6) to provide additional details to assist the NRC staff in determining whether the events were analyzed in an acceptable manner. Using the same response provided by Framatome in response to RAI 7 (Reference 52) on the generic ATWS-I TR, the licensee provided information on the fuel types, fuel-specific data, boundary conditions, and other assumptions used for these two cases. The licensee indicated that **[[** **]]**. However, as was the case for the linear reactor benchmarks, some input data for **[[** **]]** were not directly available for all fuel in the core; in these cases, the additional input data were inferred by comparisons to similar **[[** **]]**. The NRC staff finds this acceptable for the same reasons as stated in the previous section.

The licensee also provided further information on the boundary conditions used in both models. For the Oskarshamn-2 benchmark, most initial conditions and boundary conditions were taken from the OECD/NRC Oskarshamn-2 BWR stability benchmark specifications (Reference 61). Notably, this included the FW temperature time-dependent behavior provided in the benchmark specifications in which the FW temperature was assumed to decrease earlier than the measured value to account for the heat-conduction-related time delay between the actual and measured FW temperature during the event. The FW flow rate was decreased from the measured data to ensure reasonable water level calculation in the RAMONA5-FA ATWS-I code, and two sets of runs were made for pump speed: one run using the measured pump speed versus time and a second run using a modified pump speed to more closely match the measured core flow rate. The NRC staff reviewed these assumptions and finds them to constitute a reasonable and acceptable representation of the Oskarshamn-2 instability event.

[[

]] The NRC staff finds this acceptable because it **[[**

]]

For the BWR-A benchmark, **[[**

]]

The licensee supplemented the LAR with the results for the two benchmarks as shown in Figures 6-24 through 6-30 in Appendix A to the licensee's RAI responses (Reference 6). In the Oskarshamn-2 case, the calculated core-average power matches the measured core-average power closely up to the point of instability, and the calculated oscillation onset time and oscillation frequency appear to match the measured values closely, while the calculated oscillation growth rate appears to be noticeably larger than measured, which is conservative. The two different assumptions used for pump speed affected the oscillation growth rate, but this growth rate was higher than measured in both cases.

[[

]] Because the **[[**
]] the NRC staff finds that **[[**

]] Furthermore, if **[[** **]]** it would be expected to
[[**]]** which is
conservative in terms of oscillation timing and PCT.

Therefore, the NRC staff finds that both nonlinear reactor benchmark cases demonstrate the accurate or possibly conservative prediction of oscillation onset time and growth rate during measured BWR instability events with relatively large oscillation amplitude.

3.5.4 <u>Uncertainty Analysis</u>

As opposed to analyses for design-basis events, which should explicitly account for modeling uncertainties to ensure that the safety criteria are met, ATWS analyses may use best-estimate or reasonably bounding modeling approaches to demonstrate acceptable consequences to the public under limiting ATWS events. No explicit requirement or guidance is given for analyzing uncertainties in the calculated results for these events. The rationale for this is that ATWS events, which are beyond design-basis events, have very low probabilities of occurrence compared to design-basis events.

Although ATWS analyses may use a best-estimate approach, some understanding of the impact of variations in specific parameters or modeling assumptions in relation to satisfying the ATWS acceptance criteria is important in order to properly evaluate the models and assist in determining acceptable input and modeling requirements for the given application. The licensee provided a PIRT in Section 4.15 of Attachment 14 to the LAR to assist in this process. As discussed in Section 3.5.2.1 of this SE, the licensee ranked various phenomena by their importance for three FoMs – oscillation inception, limit cycle amplitude, and post-dryout, which affect the ATWS-I event progression in different ways and contribute to the overall PCT in a given calculation. The NRC staff performed its review of the RAMONA5-FA ATWS-I

methodology partly based on this PIRT as a means of focusing the review preferentially on phenomena and corresponding models with higher importance. For example, the NRC staff reviewed information submitted by the licensee in Appendix A of its RAI responses (Reference 6), which used the same response provided by Framatome in response to RAI 4 and RAI 11 on the generic ATWS-I TR (Reference 52) to address, respectively, the validation of the gap model and sensitivity studies on the gap conductance values, in part because the **[[**
]] were dispositioned as parameters of high or medium importance for the oscillation inception and post-dryout FoMs. Additionally, although the gap model was applicable to the linear and nonlinear core benchmarks, it was not applicable to **[[**
]]. Therefore, numerical sensitivity analyses were particularly useful for this model to understand the model's impact on stability calculations. The NRC staff's evaluation of these RAIs is presented in Section 3.5.2.4.6 of this SE.

Several other highly-ranked phenomena such as total core power, total core flow, FW temperature, core size, and core design are determined or justified uniquely for the Brunswick application Thus, they were not subject to sensitivity studies.

Additional highly-ranked phenomena such as **[[**
]] are related to processes involving fluid transport, heat transfer, and neutronic coupling, which impact the stability behavior of the system. The models used to determine **[[**
]]. Their behavior under transient conditions, specifically, their impact on stability, was validated in an indirect manner through their impact on the KATHY and full-core stability benchmarking, which provide an integral validation of the stability-related dynamic processes as discussed in Sections 3.5.3.4 through 3.5.3.7 of this SE. Due to the extensive benchmarking of the stability predictions of the code under a wide range of operating conditions that provide confidence that the relevant phenomena are calculated accurately, additional sensitivity calculations were not requested by the NRC staff for these models.

The licensee supplemented the LAR (Reference 6) using the same response provided by Framatome in response to RAI 8 and RAI 9 on the generic ATWS-I TR (Reference 52) to provide justification that the core and vessel nodalization were sufficient to provide reasonable and accurate prediction of PCT during ATWS-I events. The discretization used in the numerical solution of the models impacts the transport of mass, momentum, and energy in the system, in particular, impacts numerical diffusion; therefore, it may have high importance on the calculated stability behavior. Discussion and evaluation of this information provided by the licensee is given in Sections 3.5.2.5.12 and 3.5.2.5.1 of this SE, respectively.

Additional sensitivity studies were provided by the licensee in the submittal, including sensitivities on **[[**
]] The sensitivities **[[**
]] are discussed in Sections 3.5.2.5.7 and 3.5.2.1, respectively, of this SE.

The sensitivity studies **[[**
]] with results shown in Table 7-1 of Attachment 14 to the

LAR. These sensitivities indicate **[[**

]] The NRC staff's experience shows that the rod node associated with initial failure-to-rewet is not necessarily the hottest (highest peaking factor) node in the core, and additional higher-power nodes that fail to rewet later in the event may cause large, rapid increases in PCT, potentially on the order of hundreds of degrees.

In addition to possible PCT increases associated with changes to the limiting PCT node location, the PCT at a given limiting node location is also expected to increase with increasing core inlet subcooling because this increases the core average power level. This behavior was observed in the time-dependent results for the sample problem provided in the submittal, as well as the additional sensitivity results provided by the licensee in its supplement to the LAR (Reference 6) in which it used the same response provided by Framatome in response to RAI 8 through RAI 12 on the generic ATWS-I TR (Reference 52). In these results, the PCT following failure to rewet appears to increase and decrease in tandem with the core inlet subcooling.

[[

]] the NRC staff attributes this primarily to **[[**

]]

The determination of core inlet subcooling is not straightforward and depends on the code's ability to accurately model the mixing of injected FW from the vessel FW spargers into the vessel downcomer liquid, or vapor, if the water level is low enough. Additionally, the code must accurately model the mass and energy transport of this fluid through the vessel downcomer and lower plenum in order to properly determine the core inlet subcooling as a function of time. This calculation further depends on the code's ability to accurately model the FW flow rate, which responds to changes in the steam flow rate exiting the vessel and is characterized by a time delay based on the balance-of-plant dynamics. As a result of these dynamic effects, the core inlet, and relatedly, the PCT, may continue to increase for a significant length of time **([[** **]])** after operator actions are performed, such that the timing and magnitude of peak PCT is determined by competing dynamic effects associated with FW flow rate, water level, and the mass and energy transport of fluid through the vessel.

Based on the sensitivity results provided in the submittal, as well as in responses to RAI 8 through RAI 12, the discretization assumptions, including vessel nodalization, core nodalization, and timestep size, as well as core modeling assumptions such as gap conductance, appear to have a **[[** **]]** effect on the core inlet temperature response for the Brunswick sample problem. These results highlight the importance of accurately modeling the vessel and recirculation loop, as well as the balance-of-plant dynamics on a plant-specific

basis, as these are expected to be the primary determinants of the core inlet temperature response during ATWS-I events. Modeling assumptions that impact the core response, such as core nodalization and gap conductance, may have some effect on core inlet temperature, but the more important effect of these parameters appears to be in impacting the stability behavior of the core itself and the timing of oscillation growth with respect to operator actions. The NRC staff expects that this conclusion likely also holds for cycle-specific changes because such changes would primarily impact the core behavior, by, for example, changes in the radial and axial power distribution, while the recirculation loop and balance-of-plant dynamics would typically not change between cycles. As discussed in Section 3.5.2.10 of this SE, any changes to the Brunswick configuration other than cycle-specific changes to the fuel in the core will require an evaluation to ensure that the ATWS-I analyses remain bounding or reanalysis of the ATWS-I event with the updated plant configuration. In addition, ATWS-I analyses will be justified to reasonably bound the behavior for future cycles when considering possible changes in oscillation onset timing and mode behavior that are expected to be caused primarily by differences in core fuel loading and operational changes, even when the ex-core plant configuration remains unchanged.

Even though NRC guidance for beyond design-basis accidents such as the ATWS-I event does not require uncertainties to be accounted for within the analysis conclusions, the licensee provided sensitivity analyses to demonstrate the relative sensitivity of the ATWS-I results to specific parameters that were not explicitly evaluated through code validation. As discussed above, most of the sensitivities were relatively modest, except for the existing **[[**
]] assumed in the analyses. The NRC staff finds that the information provided in the submittal, as supplemented in the RAI responses, was adequate to ensure that the important sensitivities are adequately addressed for each application of this methodology.

3.5.5 <u>Brunswick ATWS-I Calculations</u>

3.5.5.1 <u>Modeling of ATRIUM 11 Fuel</u>

ATRIUM 11 contains an 11x11 array of fuel rods compared to the 10x10 array in ATRIUM 10 and ATRIUM 10XM fuel. The **[[**

]].

Appendix E of ANP-3694P (Attachment 14 to the LAR (Reference 1)) presents modified fuel rod models specific to ATRIUM 11 fuel that account for Chromia doping of the UO_2 fuel pellets. The fuel thermal conductivity model was adapted from the approved RODEX4 model in ANP-10340P-A, Revision 0, “Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods” (Reference 13). **[[**
]]. Because these models are based on approved models for Chromia-doped fuel, the NRC staff finds these models acceptable for use in the Brunswick ATRIUM 11 ATWS-I analyses.

ANP-3694P (Attachment 14 to the LAR (Reference 1)) presents experimental validation data for ATRIUM 10 and ATRIUM 10XM fuel for void fraction, pressure drop, steady-state CPR, stability, and transient dryout/rewet phenomena. The ATRIUM 11 experimental validation in the same document includes **[[**

]] was presented in ANP-3703P (Attachment 15 to the LAR (Reference 1)) and ATRIUM 11 stability information was presented in ANP-3643P (Attachment 7 to the LAR (Reference 1)).

Appendix D of ANP-3694P presents the benchmarking of ATRIUM 11 fuel for the CPROM correlation. The process for experimental benchmarking and fitting of the CPROM coefficients was identical to the process presented in Appendix A of ANP-3694P and reviewed in Section 3.5.2.11 of this SE. The ATRIUM 11 experimental database consists of **[[ ]]** test points, which validate the ATRIUM 11 CPROM correlation for **[[ ]]**.

The broad set of experimental validation for ATRIUM 10 and ATRIUM 10XM was instrumental in the NRC staff's evaluation and approval of the ATWS-I models in ANP-3694P. Based on its review, the NRC staff has concluded that ATRIUM 11 is **[[ ]]** for ATRIUM 11 is adequate and no additional validation is required. This conclusion is based on the fact that ATRIUM 11 is **[[ ]]**. In Section 3.5.2.7.5 of this SE, the NRC staff determined that the CPROM-based dryout and rewet model provides a good representation of the physical phenomena occurring during post-dryout BWR oscillations, such that the dryout/rewet formulation is applicable to multiple different fuel types, as long as they have reasonably similar hydraulic and geometric features, as indicated above. The change from a 10x10 lattice to an 11x11 lattice does not represent a significant change in hydraulic and geometric characteristics because the outer diameter of the fuel rods is reduced such that the hydraulic cell dimensions are largely the same.

Because of the similarity between ATRIUM 11 and ATRIUM 10XM, the NRC staff finds that the models in ANP-3694P, including the CPROM-based dryout and rewet model, remain applicable to ATRIUM 11. Note that while the form of the CPROM correlation remains applicable for ATRIUM 11, the CPROM coefficients must still be fitted **[[ ]]** specific to ATRIUM 11. This fitting was performed in Appendix D of ANP-3694 using ATRIUM 11 specific data and the same approach discussed in Section 3.5.2.11 of this SE, and therefore, the NRC staff finds the methodology presented in ANP-3694P to be acceptable for use with ATRIUM 11 fuel.

The other area where a correlation used to model the ATRIUM 11 fuel assembly was **[[ ]]**, the NRC staff finds that the void correlation is acceptable for use with the ATRIUM 11 fuel assembly design.

3.5.5.2 Brunswick ATRIUM 11 ATWS-I Analyses

Appendix F of ANP-3694P (Attachment 14 to the LAR (Reference 1)) provides the ATWS-I results for Brunswick with ATRIUM 11 fuel. The licensee performed all required analyses based on the calculation procedure provided as part of its proposed ATWS-I analysis methodology. Per this procedure, the **[[ ]]** cycle exposures were analyzed with **[[ ]]**. The ATWS-I analyses were performed only for Brunswick, Unit 1, and not for Unit 2. The NRC staff finds this acceptable because the Unit 1 core has **[[ ]]** and is, therefore, expected to be more unstable than the Unit 2 core, resulting in more limiting PCT for Unit 1 during postulated ATWS-I events. Exclusion of Unit 2 from the stability and ATWS-I analyses is consistent with the Brunswick analysis of record.

In accordance with the ATWS-I analysis methodology, the **[[ ]]**. The licensee supplemented the LAR (Reference 6) to provide additional detail on **[[ ]]**. Using the same response provided by Framatome in response to RAI 7 on the generic ATWS-I TR (Reference 52), the licensee provided information that demonstrated that **[[ ]]**. The NRC staff finds this approach for the average rod to be reasonable and finds that the r and θ_0 adjustments performed for Brunswick provide an overall conservative impact on PCT. Therefore, the NRC staff finds the r and θ_0 adjustments to be acceptable.

The licensee assumed a 1.3 °F/second FW temperature reduction rate for the TTWBP event. The NRC staff issued RAI 1 (Reference 38) to request justification for this value, as well as clarification of what (if any) initial delay was assumed prior to FW temperature reduction. In the RAI response (Reference 6), the licensee indicated that 1.3 °F/second was taken from the analysis of record that considered rates of 0.5 °F/second and 1.3 °F/second. For that review, the NRC staff determined 0.5 °F/second to be acceptable for Brunswick. The use of 1.3 °F/second in the current application provides additional conservatism relative to 0.5 °F/second. The licensee clarified that no initial time delay was assumed following the trip, which is consistent with the analysis of record. No plant changes have occurred since then to cause the 0.5 °F/second and 1.3 °F/second rates to be nonconservative. Therefore, the NRC staff finds the use of a 1.3 °F/second FW temperature reduction rate with zero initial delay to be acceptable.

In RAI 3 (Reference 38), the NRC staff requested confirmation that the steam line and valve modeling options in the ATWS-I methodology accurately capture the expected Brunswick performance during ATWS-I events. In the RAI response (Reference 6), the licensee confirmed that the steam line and SRVs were modeled consistently with the expected plant performance,

with opening setpoints, closing setpoints, and delay times set to the licensing setpoints. Because the steam line and SRVs were modeled in accordance with the Brunswick configuration, the NRC staff finds these models to be implemented acceptably.

The NRC staff issued RAI 6 (Reference 38) to request justification that the selected settings and modeling options used in the ATWS-I analyses are appropriate, including core and vessel nodalization, time step control parameters, and noise parameters. In the RAI response (Reference 6), the licensee discussed that all benchmarks and analyses in ANP-3694P used consistent vessel and time step control parameters, and that **[[**
]] during the event in a consistent manner with the benchmarking. The NRC staff finds the selected settings and modeling options to be acceptable because these same settings were shown to provide acceptable agreement with measured data, as discussed in Section 3.5.3 of this SE.

3.5.5.3 ATWS-I Analysis Results

ATWS-I analysis results are given in Table F-1 of ANP-3694P (Attachment 14 to the LAR (Reference 1)). The limiting PCT occurred at **[[** **]]**. Based on the figures provided in ANP-3694P, **[[**

]].

As discussed in Section 3.5.2.10 of this SE, the **[[**

]]. Therefore, the core average power is dictated by the system response and, to a large extent, is unaffected by the core configuration or progression of oscillations.

By contrast, the location of PCT may be closely tied to oscillation progression, which depends on oscillation amplitude and may include complex interactions between multiple oscillatory modes. The PCT after failure to rewet is directly dependent on the power peaking of the local node, so the PCT will increase as increasingly highly-peaked nodes fail to rewet. In RAI 4 (Reference 38), the NRC requested justification that variations in neutron kinetics response and associated oscillatory behavior **[[** **]]** during the ATWS-I event. Such variations may occur for core designs other than the reference equilibrium cycle core design that was analyzed, or for cycles with different control rod patterns or operating strategies than were considered in the analysis.

In the RAI response (Reference 6), the licensee indicated that **[[**

]]. Therefore, the NRC staff finds that the ATWS-I analyses for the reference cycle provides nearly the most limiting possible PCT and that differences in core design will not lead to PCT exceeding 2,200 °F.

Based on the calculation procedure described in the ATWS-I analysis methodology described in the LAR, the licensee concluded that the 2RPT event is **[[**

]] The NRC staff issued RAI 5 (Reference 38) to request additional justification that the 2RPT ATWS-I event will remain non-limiting for Brunswick under current conditions, as well as for future plant design or operating changes, which may affect the stability behavior during ATWS.

In the RAI response (Reference 6), the licensee described **[[**

]], such that the TTWBP event will remain limiting relative to 2RPT when ATRIUM 11 fuel is considered. The NRC staff agrees that fuel differences between ATRIUM 10XM and ATRIUM 11 are not expected to change the outcome of the ATWS-I analyses to the extent that 2RPT would become limiting over TTWBP. Furthermore, the TTWBP analysis **[[**

]]

The licensee discussed that future design or operational changes made at Brunswick will need to be assessed for their impact on ATWS-I. This includes reevaluation of ATWS-I for any change that could increase the core inlet subcooling or change the SRV setpoints. The NRC staff accepts this response because any plant design and operation changes that may impact ATWS-I would potentially affect the licensing basis for Brunswick and would require proper evaluation and approval prior to their implementation.

ANP-3694P presents Brunswick results for a full core of ATRIUM 10XM fuel (ANP-3694P, Section 7) and a full core of ATRIUM 11 fuel (ANP-3694P, Appendix F). In RAI 8 (Reference 38), the NRC staff noted that the PCT results for the ATRIUM 11 core were **[[** **]]** than for the ATRIUM 10XM core. The NRC requested an explanation for this trend, given that ATRIUM 11 was shown to be **[[** **]]** than ATRIUM 10XM fuel. In the RAI response (Reference 6), the licensee explained that, for the ATRIUM 10XM analysis, **[[**

]] The NRC staff accepts this explanation and finds that it adequately explains the observed trend in calculated PCT for each fuel type.

As discussed in Section 3.5.2.4.6 of this SE, gap conductance may have a significant impact on the growth of oscillations and timing of failure to rewet that may impact whether regulatory limits are exceeded during ATWS-I events. Particular aspects or features of new fuel designs may notably impact the gap conductance. A gap conductance sensitivity study was performed in ANP-3694P (Reference 52) for ATRIUM 10XM fuel, but not for ATRIUM 11 fuel. Therefore, the NRC staff issued RAI 2 (Reference 38) to request gap sensitivity results for ATRIUM 11 and a discussion of how the gap conductance sensitivity will be addressed when fuel design changes occur in Brunswick. In the RAI response (Reference 6), the licensee provided a **[[** **]]** gap conductance sensitivity study. PCT varied by **[[** **]]** °C across these cases. **[[**

]] The licensee stated that as part of its proposed ATWS-I analysis methodology, for any fuel introductions at Brunswick beyond ATRIUM 11, the gap conductance will be justified to be sufficiently similar to ATRIUM 11 or a new gap conductance sensitivity will be performed. The NRC staff finds this acceptable because it will ensure that any potential impacts of new fuel designs on gap conductance will be appropriately addressed for the ATWS-I analyses.

3.5.6 <u>Confirmatory Calculations for ATWS-I</u>

The NRC Office of Nuclear Regulatory Research (RES) staff has analyzed ATWS-I transients using the TRACE code to support the NRC staff's evaluation. A detailed comparison of a reference base case between TRACE/PARCS and the licensee's analysis methodology can be found in a non-public report from RES to NRR (Reference 62), and a summary of the analysis and results is included below.

3.5.6.1 <u>Base Case Analysis</u>

The base case is a simulation of an ATWS-I event initiated by a turbine trip with turbine bypass available. As a result of the turbine trip, both recirculation pumps automatically trip, and the turbine bypass valves open. The turbine bypass capacity is insufficient to relieve all the steam generated in the reactor, leading to an increase in the RPV pressure until SRVs cycle to control the pressure. Eventually, the core inlet temperature decreases because extraction steam from the turbines to the FW heaters has been cut off. The decrease in temperature causes the reactor power to increase and eventually leads to the reactor becoming unstable. The event is mitigated by two existing key manual operator actions. First, operators will cease feed injection and begin to lower the reactor water level. This action has the effect of reducing the core inlet subcooling and reducing reactor power level. Second, operators will inject soluble boron through the standby liquid control system (SLCS), which has the effect of also reducing the reactor power.

In the TRACE simulation, the turbine trip is initiated after 10 seconds of steady state simulation. The closure of the turbine stop valve causes a sudden increase in the RPV dome pressure, as shown in Figure 3. After reaching a peak, the RPV pressure oscillates according to the SRV lift and seat pressures and remains between ~7.5 MPa and 8 MPa until 200 seconds. The TRACE model of the turbine bypass is relatively simple in that it does not include an electro-hydraulic controller to maintain pressure at a desired setpoint. Therefore, late in the transient, when the reactor power is decreased below the turbine bypass capacity, the RPV pressure begins to decrease in the TRACE calculation. In the TRACE model, the bypass is simulated as a valve that opens and no active control is simulated. However, in the actual plant, the turbine bypass system includes a pressure regulation controller to maintain a desired steam line pressure. This simplifying assumption only produces an error when the reactor power falls below the turbine bypass capacity; this occurs late in the transient, well after mitigating operator actions have shown an effect in decreasing reactor power and fuel temperature.

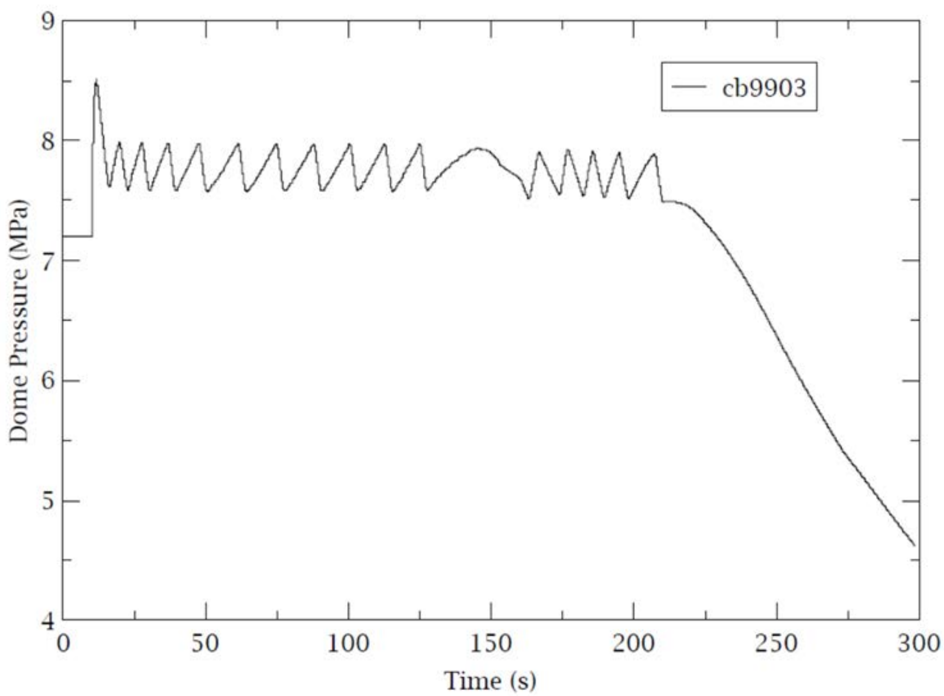

Figure 3: Reference Case Dome Pressure During TTWB ATWS-I

In response to the turbine trip, both recirculation pumps automatically trip. The pumps coast down and the reactor approaches a natural circulation condition. Figure 4 shows the core flow response. During the natural circulation phase, the reactor core flow drops to approximately 20 percent of rated core flow (%RCF). This flow rate appears to be underpredicted because TRACE overpredicts the pressure loss in the separators. Reference 7 provides a description of the TRACE separator models, in particular, how the losses in the separators are higher than the plant design values. A higher separator loss results in larger pressure losses around the loop between the core, separators, and downcomer. This higher loss results in a bias in the TRACE calculation where TRACE underpredicts the natural circulation flow rate. A flow rate between 25 and 30 %RCF is expected, but the current analysis results are conservative because the core flow rate is slightly underpredicted. The core flow is further reduced later in the transient in response to existing mitigating manual operator actions.

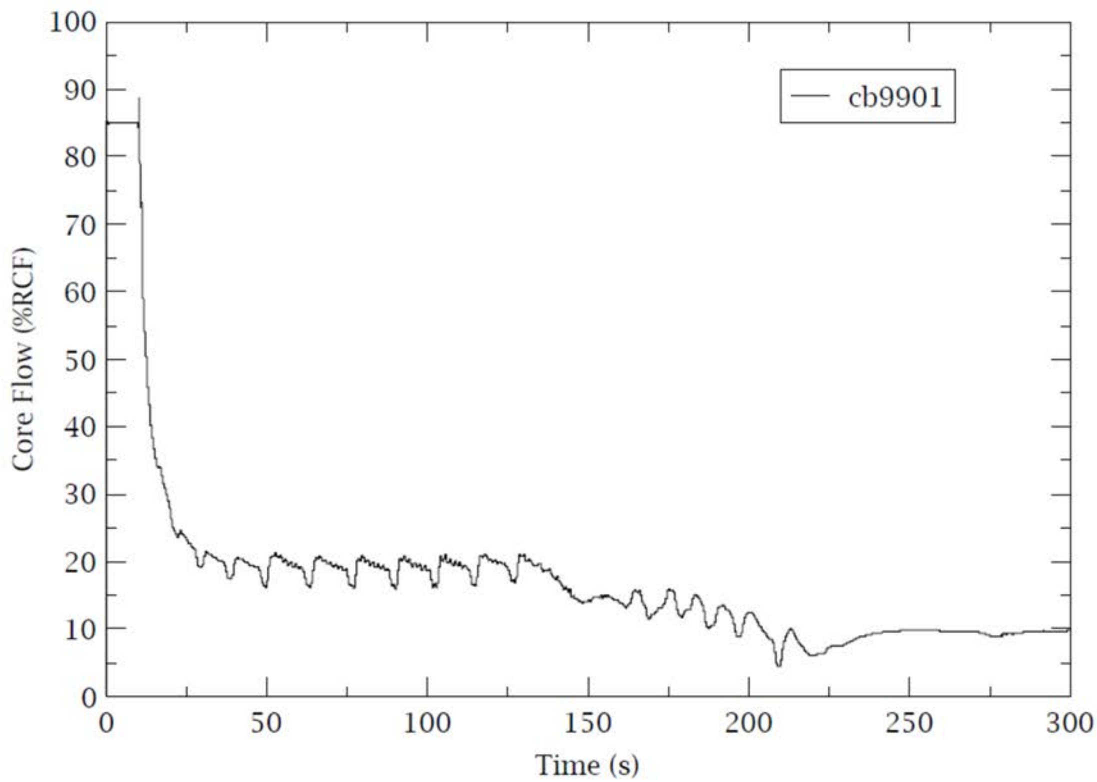

Figure 4: Reference Case Core Flow During TTWB ATWS-I

Figure 5 shows that after 130 seconds, the core flow rate decreases in response to a reduction in the reactor water level (see Figure 5). However, in the natural circulation phase, prior to the operator intervention to lower level, the FW temperature is decreasing. The decrease in FW temperature occurs because the turbine trip isolates the flow of extraction steam to the FW heater cascades. This reduction in feed temperature increases core inlet subcooling and power.

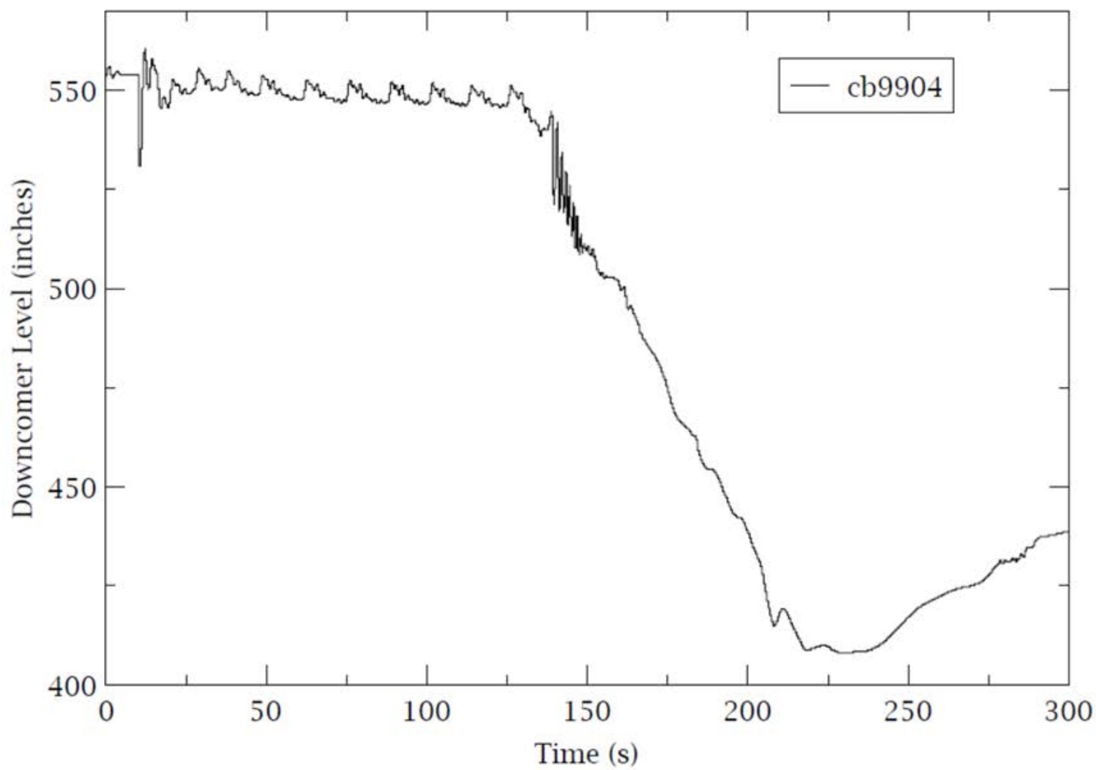

Figure 5: Reference Case Reactor Water Level During TTWB ATWS-I

Figure 6 shows the core inlet subcooling. Following the turbine trip, the subcooling steadily increases. There are some oscillations in the subcooling that occur because of variations in the RPV pressure during SRV cycling. The effect of the manual action to lower level is apparent in the late term response of the inlet subcooling (i.e., after 130 seconds), which shows a reversal in the subcooling trend and a significant reduction in the subcooling by 200 seconds.

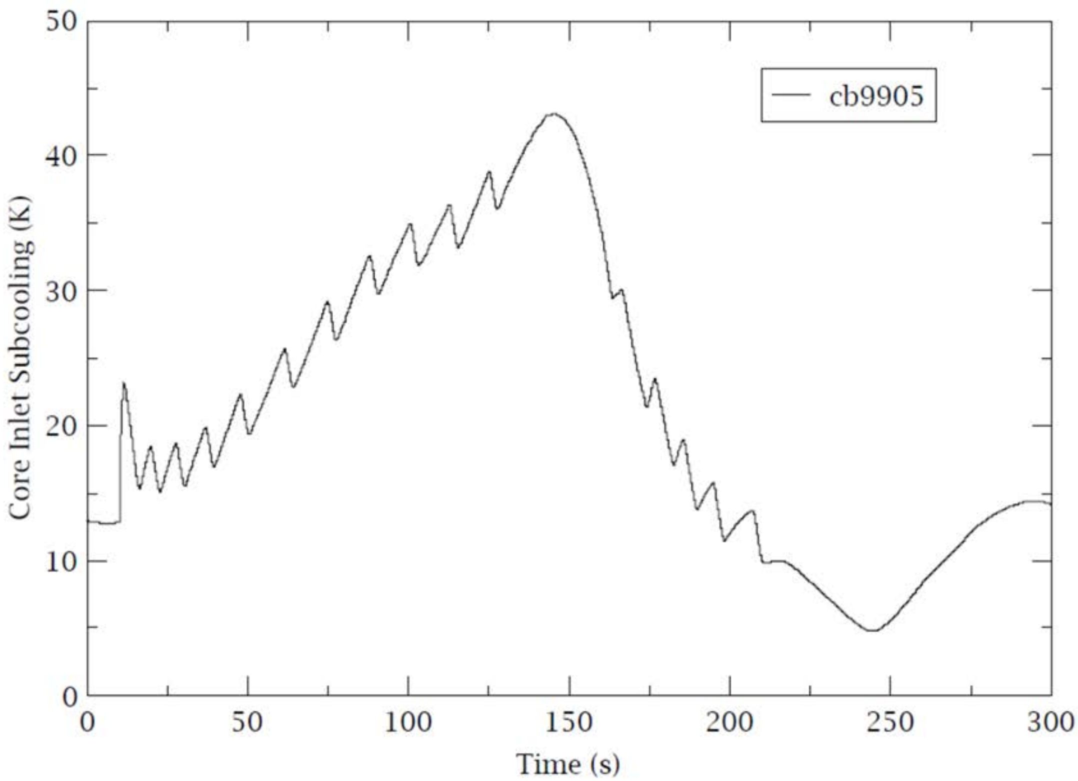

Figure 6: Reference Case Core Inlet Subcooling During TTWB ATWS-I

The increased subcooling causes the reactor power to increase prior to operator intervention. Figure 7 shows the transient reactor power response. When the turbine trips, this causes an increase in RPV pressure and void collapse in the core early in the transient, causing the reactor power to spike to nearly 300 percent of the current licensed thermal power (%CLTP). The power increase is arrested by a combination of void formation in the core and Doppler reactivity. The recirculation pump trip causes reactor core flow to decrease, and this causes the reactor power to decrease in kind (due to void reactivity feedback). After this initial phase, the reactor power increases slowly. Some oscillations are apparent in the core power. These oscillations are caused primarily by SRV cycling, but the core appears to be marginally stable in the corewide mode, leading to slowly decaying oscillations following perturbations from the SRVs opening or closing. Gross core power begins to decrease in response to manual operator actions after 2 minutes. While the corewide mode appears marginally stable, the plot of the total core power in Figure 7 does not illustrate how the reactor becomes unstable in the out-of-phase mode.

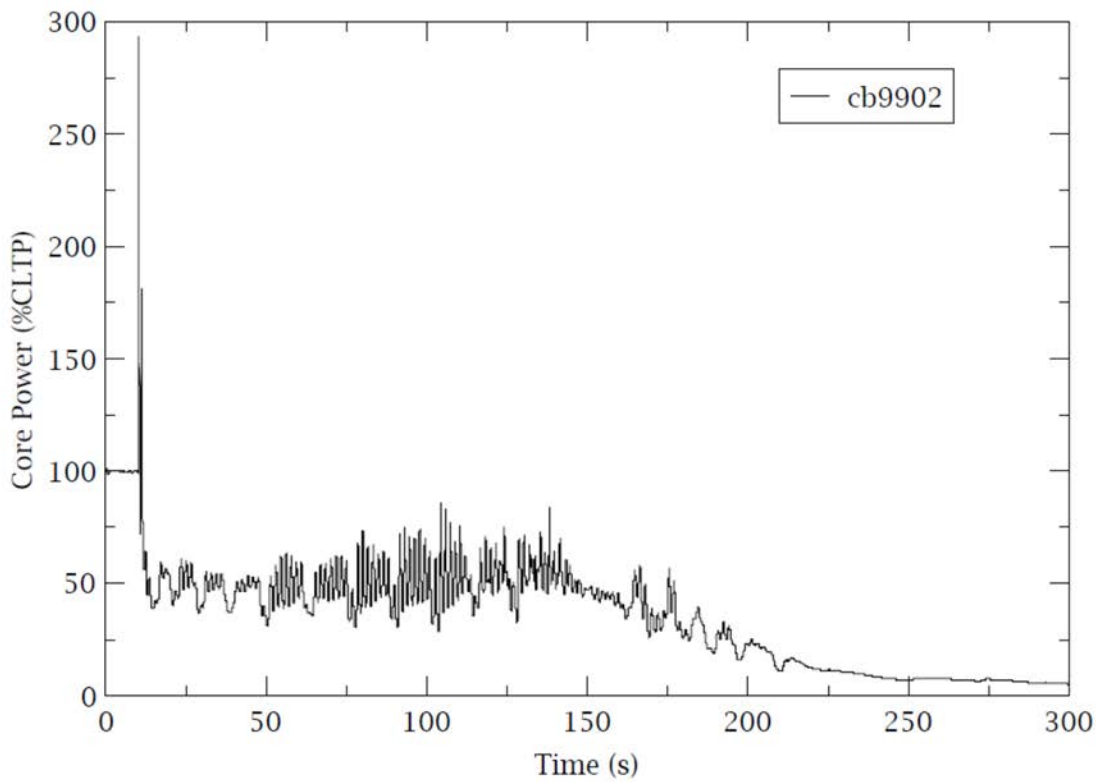

Figure 7: Reference Case Reactor Power During TTWB ATWS-I

Figure 8 shows the transient response of the first harmonic flux during the event. The result shows that while the corewide mode remains marginally stable, the first harmonic mode becomes unstable and begins to oscillate after approximately 75 seconds.

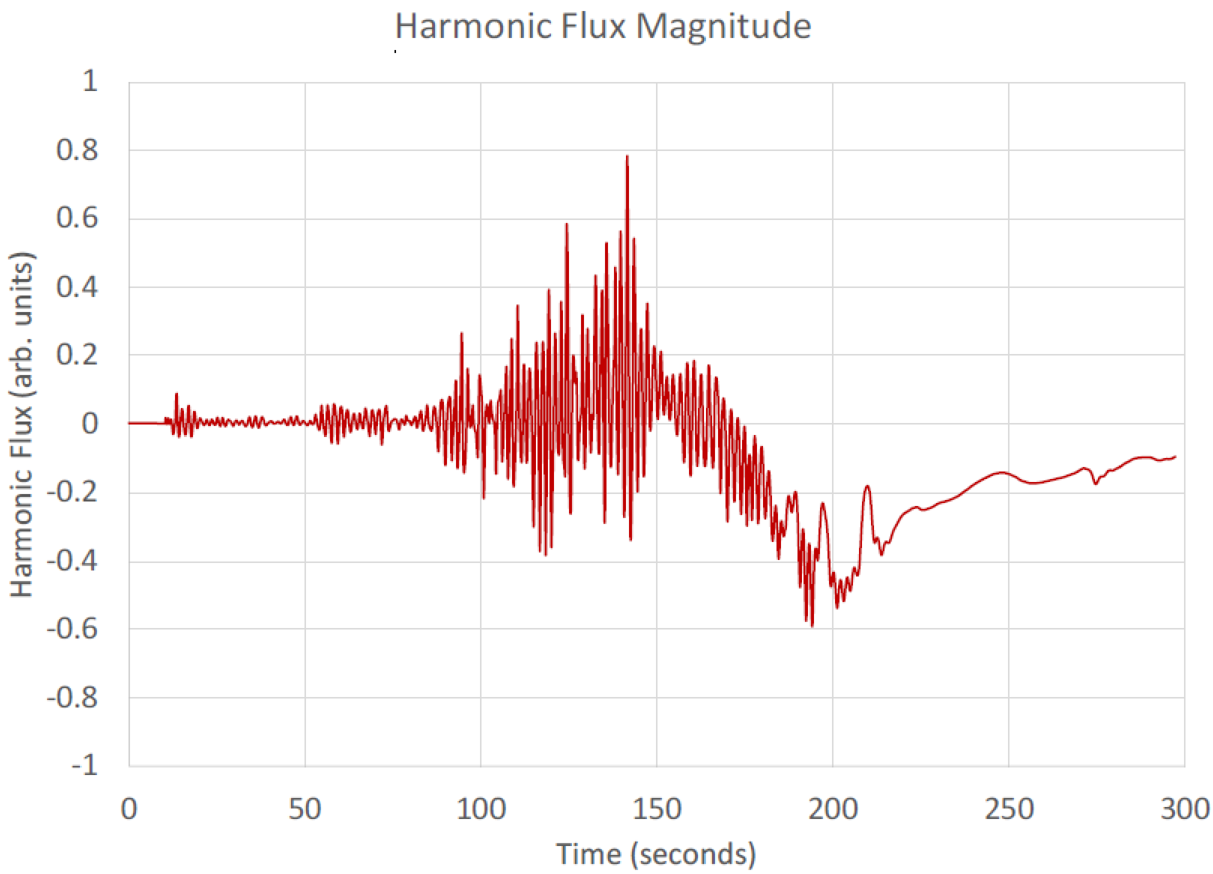

Figure 8: Reference Case Harmonic Flux Magnitude During TTWB ATWS-I

To further illustrate the out-of-phase nature of the oscillation, the NRC staff examined the channel flow rates for individual channels. In the NRC staff's analysis methodology, channel grouping is employed. However, several individual channels are selected as candidate hot channels, and these are modeled with singular channel components in TRACE to better track the thermal-hydraulic behavior in potentially limiting areas of the core.

Figure 9 illustrates the channel mapping used in this case. The numerical values in the figure represent the TRACE channel component number for each assembly in the core. Channels in the core that are lumped into a group share a common channel number. The channel grouping considers the core power distribution, harmonic peaking, fuel assembly type, and fuel assembly exposure.

								601	603	602	603	602	502	503	502	503	501								
							603	628	629	622	632	623	523	532	522	529	528	503							
				602	602	602	629	623	616	624	617	634	534	518	524	517	523	529	502	502	502				
			602	628	622	623	616	617	618	630	631	619	520	531	530	519	518	517	523	522	528	502			
		602	628	633	616	625	618	626	619	626	621	626	526	521	526	520	526	519	525	517	533	528	502		
		602	622	616	618	609	615	619	626	619	626	620	520	527	520	527	520	515	508	519	517	522	502		
		603	623	625	609	626	619	627	610	615	610	615	515	509	515	509	527	520	526	508	525	523	503		
	602	629	616	618	615	619	615	610	615	609	614	609	508	514	508	515	509	516	520	515	519	517	529	502	
602	628	623	617	626	619	627	610	615	609	611	605	614	514	504	512	508	515	510	527	520	526	518	523	528	502
602	622	616	618	619	626	610	615	610	614	605	611	608	507	511	504	514	509	516	509	527	520	519	517	522	502
602	622	624	630	626	619	615	610	615	609	613	607	613	513	507	513	508	515	509	515	520	526	530	524	522	502
602	632	617	630	619	626	609	614	609	614	607	613	607	506	513	507	514	508	515	508	526	520	530	518	532	502
602	622	624	618	626	618	612	606	614	608	611	604	613	513	504	511	507	514	505	512	519	526	519	525	522	502
702	722	724	719	726	719	713	706	715	708	712	704	713	813	804	811	807	814	805	812	819	826	819	825	822	802
702	732	718	730	720	726	709	715	709	715	707	714	707	806	813	807	814	808	815	808	826	820	830	818	832	802
702	722	724	730	726	720	716	710	716	709	714	707	714	813	806	813	808	815	809	815	820	826	830	824	822	802
702	722	717	719	720	726	710	716	710	715	705	712	708	807	811	804	814	809	816	809	827	820	819	817	822	802
702	728	723	718	726	720	727	711	716	709	713	705	715	814	804	811	808	815	810	827	820	826	818	823	828	802
	702	729	717	719	716	720	716	710	716	709	715	709	808	814	808	815	809	815	820	815	819	817	829	802	
		703	723	725	709	726	720	727	710	716	710	716	815	809	815	809	827	820	826	808	825	823	803		
		702	722	717	718	709	716	720	726	720	726	720	820	826	820	827	820	815	808	819	817	822	802		
		702	728	733	717	725	719	726	720	726	721	726	826	821	826	820	826	819	825	817	833	828	802		
			702	728	722	723	717	718	719	730	731	720	820	831	830	819	818	817	823	822	828	802			
				702	702	702	729	723	717	724	718	734	834	818	824	817	823	829	802	802	802				
							703	728	729	722	732	723	823	832	822	829	828	803							
								701	703	702	703	702	802	803	802	803	801								

Figure 9: Channel Grouping with Candidate Hot Channels Indicated

By identifying candidate hot channels based on the fundamental and harmonic mode power distributions, the NRC staff has high confidence that the individually modeled channels will contain the core hot spot in terms of the PCT. However, without first simulating the instability, it is not clear which of the candidate hot channels will be the location of the peak hot spot. Therefore, it is necessary to consider multiple candidate hot channels in the analysis.

Figure 10 below shows the active channel flows for channel 510 (candidate hot channel in the first, or northeast quadrant, of the core) and the flow for channel 711 (candidate hot channel in the third, or southwest quadrant).

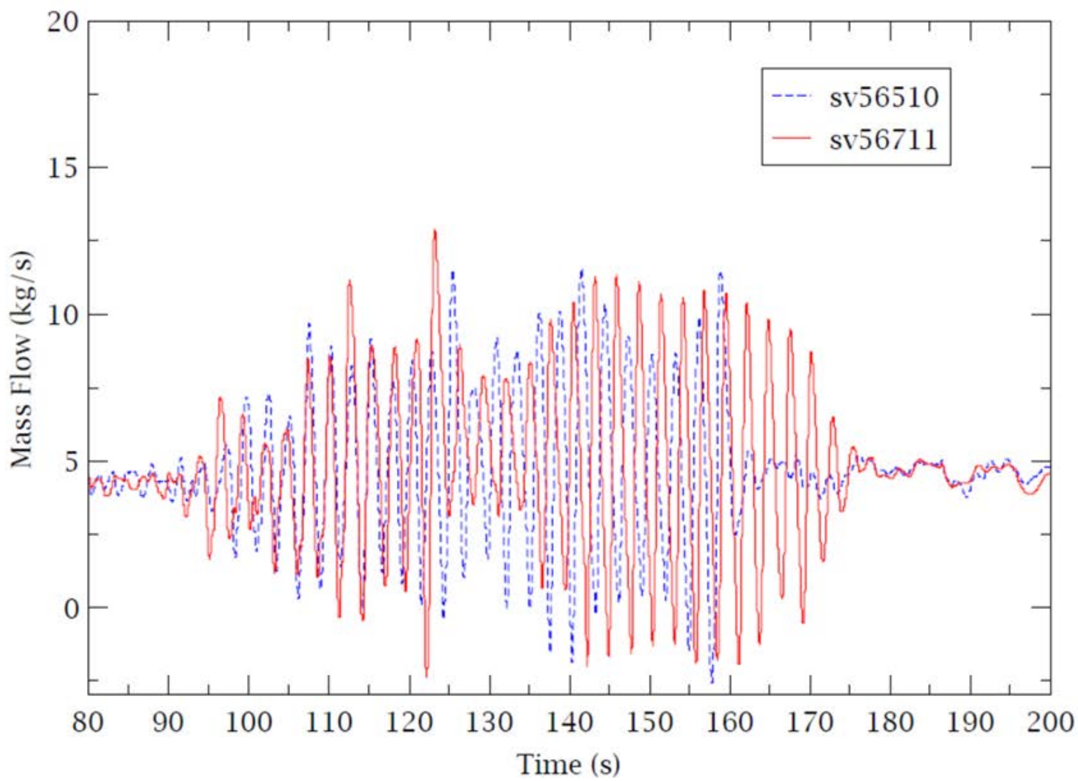

Figure 10: Reference Case Quadrant 1 and 3 Candidate Hot Channel Flows During TTWB ATWS-I

The out-of-phase nature of the oscillation is apparent after 120 seconds because the flow in one channel peaks during the low flow phase for the channel on the other side of the core. Figure 11 shows a similar comparison for channels 620 and 821 (second and fourth quadrants, respectively).

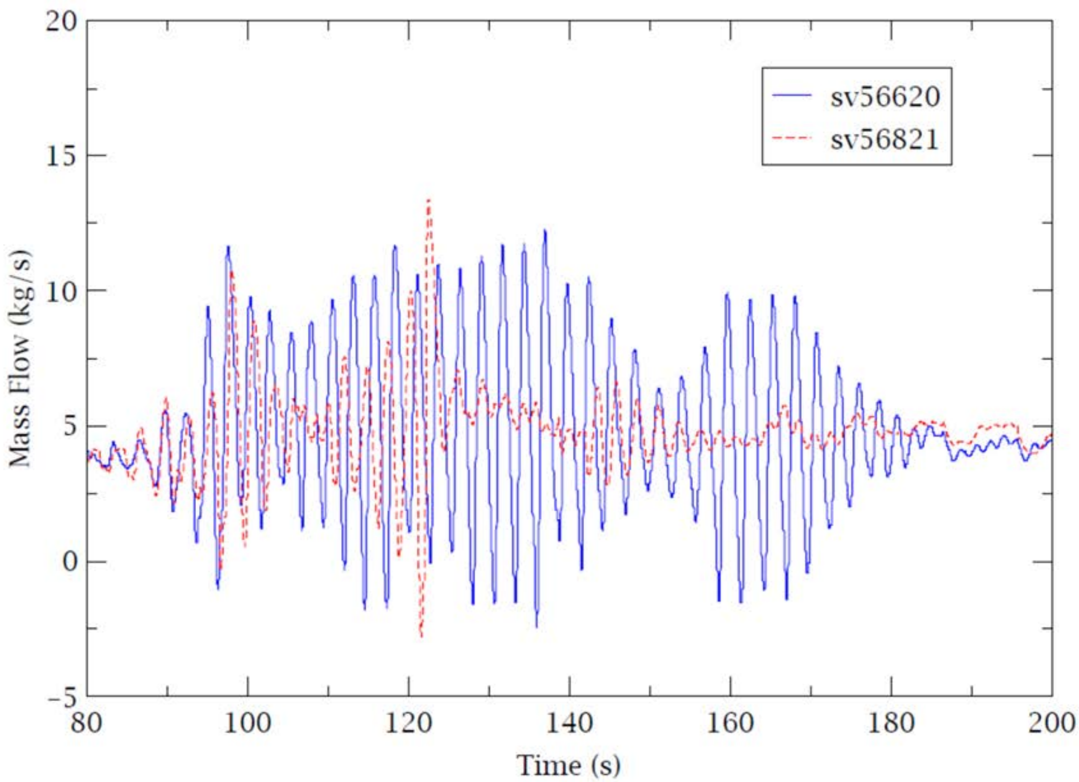

Figure 11: Reference Case Quadrant 2 and 4 Candidate Hot Channel Flows During TTWB ATWS-I

The oscillation is out-of-phase and, in fact, is a rotating mode of oscillation. Figure 12 below depicts the assembly power distribution at various discrete times around 140 seconds. This time range was selected because it is close to the timing of the maximum subcooling and the maximum oscillation magnitude.

Figure 12 below shows how the power peak rotates from the northwest corner, to the southwest, then southeast and so-on. The values provided in the figure indicate the radial power peaking factors for each assembly. The rotating mode was first studied using TRACE/PARCS by Wysocki, et al., and is well understood in terms of modal kinetics (Reference 63).

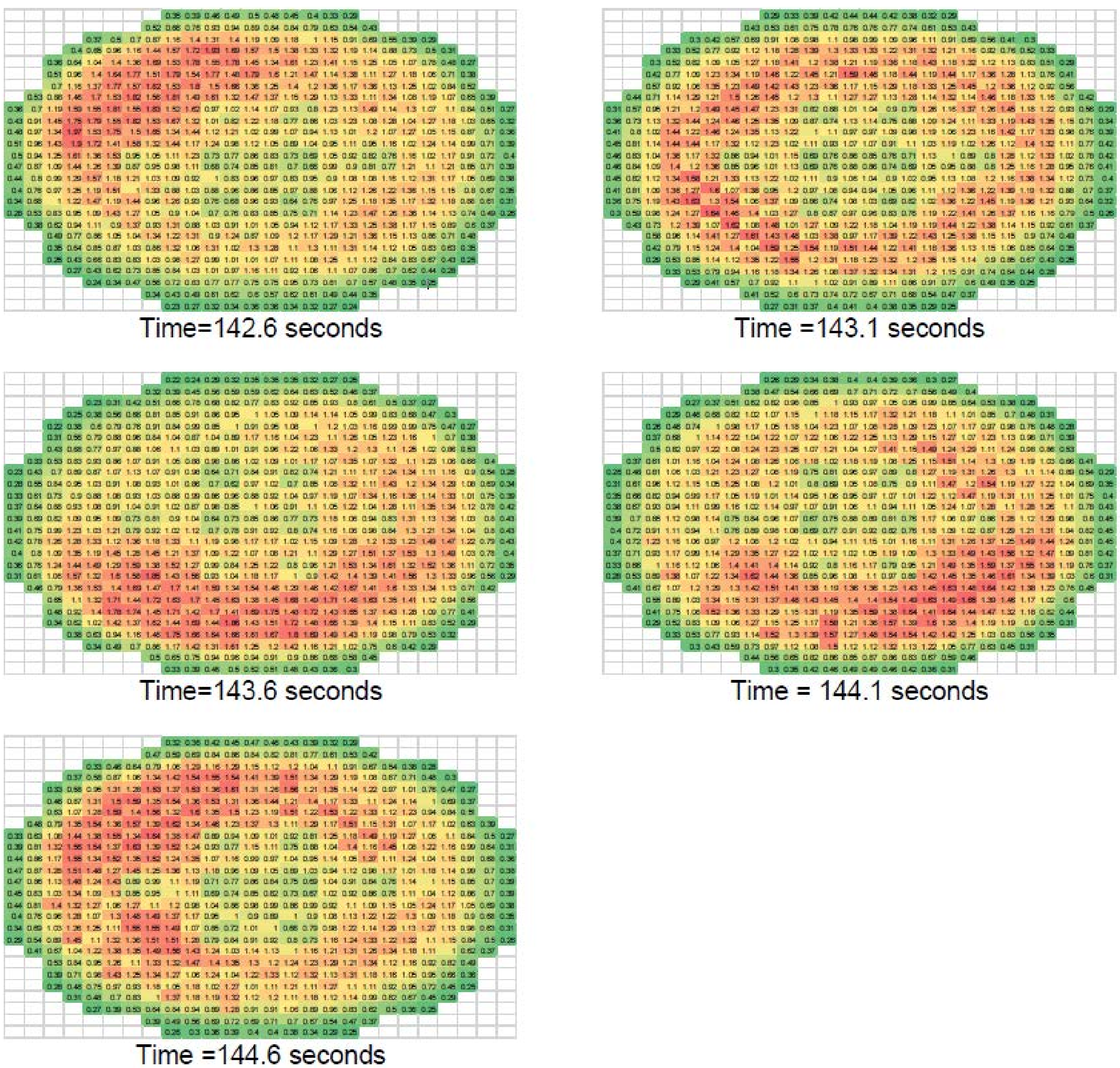

Figure 12: Assembly Power Distribution and Rotation

The PCT occurs in channel 821. Channel 821 is located in a region of the core with strong first harmonic flux peaking and could be expected to experience strong oscillations when the regional mode is excited. The NRC staff notes that the flow rates in Figure 10 and Figure 11 above are not the channel inlet flows. Rather, they represent the channel flows downstream of the water rod inlet holes. Therefore, because these flows are tracked at a higher axial elevation in the bundle, they underreport the magnitude of the flow oscillation experienced at the channel inlet. As a result, the NRC staff cannot directly compare the active flow predicted by TRACE with the channel inlet flow reported by the licensee. The channel inlet flow oscillation reported by the licensee (for the same downstream flow oscillation) will be of a higher magnitude due to flow incompressibility at the channel inlet. These plots are presented primarily to illustrate that the core becomes unstable in the out-of-phase mode and that the peak harmonic flux channel experiences large-amplitude flow oscillations.

TRACE predicts that the fuel goes into dryout during the initial flow reduction following the recirculation pump trip and that some of the fuel rods start to heatup. However, the highest temperatures occur later in the event once the reactor becomes unstable. Figure 13 shows the PCT, as well as the hottest cladding surface temperatures for candidate hot assemblies. The PCT represents the hottest cladding surface temperature anywhere in the core at any instant and does not represent a temperature history for any given location or specific bundle. The PCT plot is shown alongside plots of the peak cladding temperature for the hottest fuel rod within selected fuel assemblies. The figure shows that while channel 821 (depicted by the sv55821 curve) yields the highest overall PCT, the cladding in channel 821 does not experience as significant of a heatup until the reactor becomes unstable.

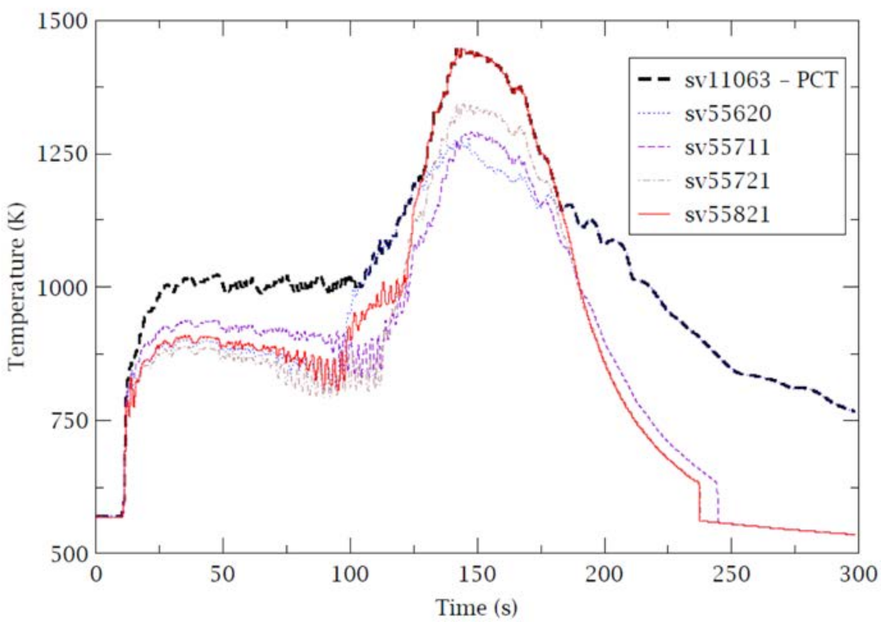

Figure 13: Reference Case Peak Cladding Temperature During TTWB ATWS-I

The NRC staff compared the TRACE/PARCS results to results provided by the licensee. The licensee's analysis was performed using RAMONA5-FA (Reference 1). Table 6: Summary of Comparison between TRACE/PARCS and RAMONA5-FA provides a high level summary of the comparison in terms of PCT and major plant parameters. The values provided in Table 6 are approximate, particularly the values shown for the licensee's analysis, because these numerical values were visually inferred from plots in (Reference 1), and the accuracy is limited by the resolution of these figures and the scales on these figures. TRACE predicts a **[[ ]].** This section discusses these differences below.

Table 6: Summary of Comparison between TRACE/PARCS and RAMONA5-FA

[[ ]]

When comparing the transient power predicted by the licensee's calculations for the turbine trip with bypass initiated (TTWB) ATWS-I, TRACE/PARCS predicts a **[[ ]]**. This could be due to small differences in the TRACE predicted pressures, which are slightly **[[ ]]** compared to the licensee's calculations, and these differences are reasonable. It is important to note that the reported figure is the core nuclear power, and a difference of **[[**

]], but the values are similar.

When comparing the licensee's predicted core flow response, the licensee predicted that natural circulation flow rate is **[[ ]]** than the TRACE predicted natural circulation flow. The licensee's analysis shows the core flow rate reaching about 25 %RCF compared to the NRC staff's calculation, which shows a flow of about 20 %RCF. The TRACE calculated flow is low due to a known bias in the TRACE calculation of the separator losses (Reference 64). Despite TRACE predicting a **[[ ]]** is in good agreement with the licensee's result. This is likely due to the **[[ ]]** predicted by TRACE compensating the slightly lower flow. Given the known bias in the TRACE calculation, the two results are in reasonable agreement. Furthermore, when comparing the licensee's prediction of the core inlet subcooling to the TRACE prediction, the results are in good agreement.

When comparing the result of the licensee's calculation of the hot channel flow, the NRC staff tracked similar flows but the flow rates cannot be directly compared. The licensee reports the channel inlet flow, but the TRACE calculation output is for the active channel flow, which is tracked downstream of the water rod inlet. The TRACE predicted active channel flows are shown in Figure 10 and Figure 11. The average flows **[[ ]]**. The NRC staff notes that the active channel flow is slightly smaller than the inlet flow because a small amount of flow is bypassed through the water rod.

The oscillation magnitude in the TRACE calculations appears **[[**

]]

TRACE predicts a **[[ ]]** onset of the instability (~80 seconds in TRACE compared to **[[ ]]** in the licensee's analysis), but this is most likely a consequence of TRACE predicting a **[[ ]]** during the natural circulation phase. As the core destabilizes due to the increasing core inlet subcooling, the TRACE calculation **[[**
]] in TRACE compared to the licensee's calculations. However, these differences are well understood and do not indicate any deficiency in the licensee's calculation. Rather, these differences indicate **[[**
]]

A comparison of the transient level predicted by the licensee's analysis was completed. It is not entirely clear where the licensee has referenced the water level in its calculation results; however, the results can be compared in terms of the relative change in the level. Both calculations agree in terms of the performance of the FW controller to maintain the level prior to the manual action to lower level. This result is expected because the NRC staff's TRACE model was recently updated to improve the prediction of the level during this phase of the transient (Reference 65). After 130 sec, the licensee predicts a **[[**

]], the NRC staff notes that this does not indicate any non-conservatism in the licensee's analysis because the differences in the calculation are related to a **[[** **]]**.

The licensee's predicted PCT was compared to the NRC staff's TRACE results. While both methods predict a significant fuel heatup during the unstable phase, TRACE predicts a **[[** **]]** PCT. TRACE predicts a PCT of 1,450 degree kelvin (°K) compared to the licensee's result of **[[** **]]**°K. There are several factors contributing to the **[[** **]]** PCT prediction in TRACE when compared to the licensee's result. The first is the difference in the **[[**

]]

The licensee's RAMONA5-FA based methodology uses a fundamentally different approach to simulating the **[[**

]]

Both calculations are in good agreement in terms of the prediction of the **[[**

]]

Early assessment of TRACE against oscillatory flow data indicates that TRACE is likely conservative in the treatment of heat transfer during such regime changes (see (Reference 66)). The comparison of TRACE to heat transfer coefficient data from related experiments indicates that TRACE can reasonably predict the heat transfer coefficient magnitude, but TRACE also predicts very rapid changes in the heat transfer coefficient when compared to the experiments. This explains why TRACE "locks" the heat transfer coefficient early on, and why the licensee's result may show **[[** **]]**. Therefore, this difference does not indicate any deficiency in the licensee's analysis. Given known differences in the analysis methods, the results are generally in good agreement.

3.5.6.2 Additional Cases Studied

A number of TRACE calculations were executed to confirm other results presented by the licensee, namely:

- The effect of the manual operator actions on PCT was investigated by assuming that operator intervention to reduce level is delayed by 30 or 60 seconds. The RES staff also considered an unmitigated case where no operator actions are simulated.

- The limiting initiating event was confirmed by evaluating an ATWS-I event that is initiated by a dual recirculation pump trip (2RPT) rather than a turbine trip with bypass available (TTWB).

- The effect of gap conductance was assessed by performing calculations where the gap conductance was both decreased and increased by 20 percent.

The details associated with the additional cases can be found in non-public reports from RES to NRR (Reference 68) and (Reference 69). While the magnitudes of the variations observed did not always match the results presented by the licensee, the trends are consistent, and the deviations in magnitude can be explained based on the inherent conservatisms in the TRACE model as discussed in the previous section.

In addition, some TRACE calculations were executed to study the impact of specific modeling sensitivities, as follows:

- The effect of the assumed minimum stable film boiling temperature (T_{min}) on the calculation results. In the sensitivity case, the T_{min} correlation is switched from the homogeneous nucleation temperature plus contact temperature model to the Groeneveld-Stewart model. This has the effect of increasing the T_{min} value by approximately 60 °K, which reduces the maximum PCT by 74 °K.

- The effect of hot rod-to-rod power peaking factor. The base case assumes a **[[**

  ]] in TRACE. This sensitivity case assumes a value of 1.3, which is consistent with PARCS calculations of the reference core cycle depletion and reflects a more realistic scenario. This has the effect of reducing the maximum PCT by 88 °K.

These results generally confirm that reducing known conservatisms in the TRACE model in an effort to avoid dryout at the beginning of the transient, while ultimately unsuccessful, does result in lower PCTs. As a result, the TRACE calculations provide reasonable assurance that the 1,478 °K acceptance criterion will not be exceeded for Brunswick.

3.5.6.3 Conclusion for the Confirmatory Calculations

The NRC staff used the TRACE/PARCS codes to simulate a TTWB ATWS-I for Brunswick with an equilibrium cycle of ATRIUM 11 fuel. The results of the NRC staff's calculations indicate that fuel damage during such an event is highly unlikely. The results of the NRC staff's calculations are in good agreement with the licensee's results when considering inherent conservatisms and

differences in the codes and models, supporting the licensee's claim that fuel damage would not occur during ATWS-I events for Brunswick.

3.5.7 Conclusions for ATWS-I

In the submittal, Brunswick presented a proposed methodology to analyze the ATWS-I event using the RAMONA5-FA code. The submittal presents a description of the ATWS-I event, the relevant phenomena, the applicable FoMs, and a ranking of the phenomena for any applicable FoMs. This information was reviewed and compared to similar information available to the NRC staff such as the PIRT documented in NUREG/CR-6743 (Reference 53) and confirmed to be consistent with previous approvals of ATWS-I or other stability related methodologies.

The application of the RAMONA5-FA code for the purpose of analyzing ATWS-I events involved the incorporation of several new models in the RAMONA5-FA code relative to what the NRC staff has previously reviewed and approved for LTSS analyses. Many of these models have been reviewed and approved by the NRC staff as part of a plant-specific ATWS-I methodology adopted at Monticello. The NRC staff reviewed the previously approved RAMONA5-FA models, the previously approved models from the Monticello ATWS-I application, and new models developed specifically for the purpose of the methodology. The NRC staff confirmed that the previously approved models and new models are applicable to analysis of the ATWS-I event.

The submittal also presents a procedure for analysis of the ATWS-I event, which **[[ ]]**. Since the intent of the proposed ATWS-I analysis methodology is to perform a single evaluation upon initial implementation at Brunswick without subsequent confirmatory analyses on a cycle-specific basis, the NRC staff carefully considered how different characteristics of future cycles might affect the results of a cycle-independent evaluation. In addition to changes in fuel assembly designs (including transition core designs), the NRC staff considered whether cycle or plant configuration changes might affect the limiting PCT or the margin to operator action timing. The licensee provided sufficient information, including the calculation procedure as part of the ATWS-I analysis methodology, to assure the NRC staff that the current and future implementation of the methodology at Brunswick would adequately address sensitivities of the coupled neutronic/thermal-hydraulic feedback to cycle-specific variations in the core neutronic or plant system response and the consistency of plant-specific models used for analysis of the ATWS-I event with the underlying validation and assessment of the methodology as described in the submittal and the RAI responses.

In order to demonstrate the capability of the RAMONA5-FA code to analyze the ATWS-I event, assessments were made against separate effects tests and integral benchmarks. Separate effects tests helped validate the RAMONA5-FA code for prediction of parameters important to the ATWS-I event, such as void fraction, pressure drop, single channel stability characteristics, and dryout/rewetting response during large-amplitude oscillations. The integral benchmarks provided confidence in the RAMONA5-FA code's ability to model full scale stability events. In some cases, sensitivity studies were used to demonstrate that the RAMONA5-FA ATWS-I methodology was either conservative or insensitive to variations in specific parameters. This provided assurance that relevant uncertainties in the ATWS-I analysis methodology and model parameters would not change the conclusions of an ATWS-I evaluation done in accordance with the submittal. Based on a general review of the tests, benchmarks, and sensitivity studies, the NRC staff finds that the methodology was appropriately confirmed to yield acceptable predictions for all parameters and phenomena important to the ATWS-I event.

In summary, the NRC staff finds that the assessment of the RAMONA5-FA code, as described in the submittal and responses to NRC staff RAIs, adequately demonstrates that RAMONA5-FA is suitable to analyze the ATWS-I event by demonstrating acceptable performance in each of the highly ranked phenomena. In addition, the NRC staff finds that the calculation procedure described in the submittal for performance of the ATWS-I analyses provides appropriate guidance to perform ATWS-I analyses that bound cycle-specific variations.

3.6 Stability Analysis Using Plant-Specific Best-Estimate Option III (BEO-III) Approach

Duke Energy proposed a new approach to perform the stability analysis for Brunswick using a plant-specific BEO-III approach developed by Framatome. The proposed approach involves a unique plant-specific aspect in that it incorporates use of the Confirmation Density Algorithm (CDA) developed by General Electric Hitachi.

3.6.1 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 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 that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed."

3.6.2 Technical Evaluation

ANP-3703P and DPC-NE-1009-P (Attachments 15 and 16 to (Reference 1)) describe the methodology proposed by Duke Energy to support the licensing basis LTSS for Brunswick, which the licensee referred to as the BEO-III with CDA approach. This methodology is intended to replace the current LTSS methodology for Brunswick, which the licensee stated to be Detect and Suppress Solution - Confirmation Density (DSS-CD).

In addition, the licensee stated that as a result of the change in analytical methods for ensuring stability, Note (f) to Table 3.3.1.1-1 of the Brunswick TSs is no longer necessary and will be deleted. Note (f) describes a condition placed upon arming DSS-CD that was applicable only during the first reactor startup and shutdown following implementation of that methodology. The NRC staff reviewed Note (f) and finds that implementation of the plant-specific BEO-III with CDA approach would obviate the need for Note (f) to Table 3.3.1.1-1.

3.6.2.1 Overview and Relationship to Previous LTSS Methodologies

The licensee's proposed plant-specific BEO-III methodology makes use of Framatome RAMONA5-FA code (Reference 70), which has been previously approved for LTSS calculations

as part of the Option III (Reference 71) and Enhanced Option-III (EO-III) methodologies (Reference 72). The Option III methodology determines the delta-CPR response during anticipated oscillations by performing an analysis consisting of three primary components:

- The first component consists of determining the MCPR margin that exists prior to the onset of oscillations. This is a plant- and cycle-specific determination that is based on the plant response to a two recirculation pump trip (2RPT), as well as during steady state operation at reduced flow conditions.

- The second component of the calculation determines to a 95/95 statistical tolerance limit the largest oscillation amplitude expected prior to oscillation suppression for a given plant configuration using analytically prescribed oscillation power range monitor (OPRM) response signals with assumed statistical distributions for oscillation growth rate, oscillation mode, and other relevant parameters.
- The third component of the calculation uses the DIVOM correlation to conservatively compute the delta-CPR response associated with this 95 percent probability with 95 percent confidence (95/95) oscillation amplitude. The DIVOM correlation is developed based on the MCPR response calculated by RAMONA5-FA during simulated oscillations of growing amplitude, starting from assumed conditions representative of the plant following a two recirculation pump trip.

This approach of dividing the calculation process into three separate components introduces significant conservatism into the calculation of OLMCPR values. For example, because the statistical analysis component does not use best-estimate RAMONA5-FA calculations to determine the core response during growing oscillations, the assumption is made that the oscillations grow with a constant DR from the time of oscillation inception until suppression. Depending on statistical sampling, the constant DR value can be well above 1.0. However, assuming a DR value significantly above 1.0 from the time of oscillation inception is conservative. In a realistic RPT event, the oscillation growth rate will begin at 1.0 at oscillation inception and gradually increase over time. This is due to the gradually decreasing core inlet temperature throughout the event, as well as changes in the recirculation pump driving flow that may continue into the early portion of the oscillations. These initially slower-growing oscillations increase the likelihood that sufficient successive oscillation counts will be recorded by the period-based detection algorithm (PBDA) prior to the oscillations exceeding the amplitude setpoint. Accordingly, the Option III and EO-III assumption of using a fixed oscillation DR leads to a conservatively high hot channel oscillation amplitude. Another conservatism lies in the process of calculating the DIVOM slope, which determines the MCPR response of fuel assemblies in the core under oscillatory conditions in a bounding (rather than best-estimate) manner.

The EO-III methodology employs the same process as Option III for determining the core MCPR response during anticipated oscillations. However, EO-III also calculates the limiting growth or DR for individual channel oscillations (ICOs) in the core. Such oscillations were found to be more prevalent at operating points farther into the unstable region (i.e., lower flow rates and higher power levels), and these oscillations could give a more limiting MCPR response than that associated with whole-core oscillations. The existence of ICOs simultaneously with whole-core oscillations invalidates the assumptions of the DIVOM relationship and is unsuitable under these conditions. Therefore, in conjunction with the normal DIVOM approach, EO-III implements a

scram region, known as the channel instability exclusion region, to ensure that the power will be suppressed before ICOs may develop.

BEO-III discards the three-step approach used in Option III and EO-III. Instead, BEO-III performs cycle-specific best-estimate RAMONA5-FA evaluations in which the entire event, including the initiating pump trip and subsequent growth of oscillations, is explicitly modeled. The event MCPR response and channel DR are then determined to a 95/95 tolerance limit to ensure adequate SLMCPR protection. These 95/95 values are determined by performing a set of statistical trials in which physical modeling parameters are randomly varied according to appropriate uncertainty distributions.

By explicitly modeling the plant and core response to the potentially limiting RPT events, explicitly treating uncertainties through a statistical process, and directly calculating the MCPR response from the oscillations that develop, many of the conservatisms inherent in the three-step approach of Option III and EO-III are avoided. Best-estimate assumptions are made for most of the modeling aspects of BEO-III; however, in some specific areas, conservative assumptions were made to ensure that the BEO-III predictions remain bounding with respect to the safety criteria.

Many of the underlying modeling aspects of the BEO-III methodology remain the same relative to Option III and EO-III. However, this is the first NRC review in which RAMONA5-FA is used within a statistical framework to determine the MCPR response and associated uncertainty during stability events. Therefore, the NRC staff focused its review on determining the acceptability of the new modeling features that were added to RAMONA5-FA, as well as the acceptability of the statistical approach to ensure that the safety limits are met during any anticipated oscillations at Brunswick.

3.6.2.2 Review of ANP-3703P, Section 2.0, "Regulatory Requirements Summary"

As discussed in Section 3.6.1 of this SE, GDC 10 and 12 of Appendix A to 10 CFR Part 50 require that SAFDLs not be exceeded under normal operation or AOOs. The relevant SAFDL for stability events is the SLMCPR.

The licensee identified two FoMs that were used to demonstrate compliance with GDC 10 and 12:

- Core MCPR at the time of oscillation suppression, referred to hereafter as the "core MCPR FoM"

- Verification that ICOs do not invalidate the assumption that the reactor protection system can detect and suppress the oscillations prior to violation of the SAFDLs, referred to hereafter as the "ICO FoM"

In ANP-3703P, Framatome evaluated the core MCPR based on simulated oscillation suppression times using the PBDA. However, as described in DPC-NE-1009-P (Attachment 16 to the LAR (Reference 1)), the licensing basis SLMCPR protection at Brunswick is provided by the CDA, which was implemented as part of DSS-CD methodology (Reference 73) during Brunswick's licensed operating domain extension to maximum extended load line limit analysis plus (MELLLA+ ™) (Reference 74). In the current LAR, Duke Energy proposes to implement

the plant-specific BEO-III methodology while retaining the CDA as the licensing basis stability algorithm. Because the CDA algorithm is proprietary to General Electric Hitachi, Duke Energy intends to employ a two-step process in which (1) Framatome supplies BEO-III calculations based on an implementation of the PBDA algorithm, and (2) Duke Energy subsequently performs post-processing analysis based on Framatome's calculated results to demonstrate SLMCPR protection with the CDA.

For the remainder of this evaluation, the term "core MCPR FoM" will be reserved for the 95/95 core MCPR values calculated by Duke Energy using oscillation suppression times based on the CDA. This reflects the direct use of these values as FoMs in determining licensing basis SLMCPR protection. Core MCPR values calculated by Framatome using the PBDA are discussed throughout the document, but it is to be understood that these are not directly evaluated against the regulatory requirements and are not considered FoMs as such.

The specific manner in which the core MCPR and ICO FoMs were assessed in the context of the statistical analysis is provided in Section 7.0 of ANP-3703P and in DPC-NE-1009-P (Attachments 15 and 16 to (Reference 1)), as evaluated below in Section 3.6.2.7.

3.6.2.3 Review of ANP-3703P, Section 3.0, "Scenario Identification"

The licensee identified a 2RPT from the minimum flow condition at rated power within the MELLLA+ ™ operating domain to be the limiting event for LTSS. This limiting event identification is consistent with previous plant-specific applications of Option III and EO-III. Pump trip events may lead to instability due to a large reduction in core flow rate combined with a relatively modest reduction in power, which moves the core toward the upper left corner of the power-flow operating map, an example of which is depicted in Figure 14. These conditions promote unstable oscillations.

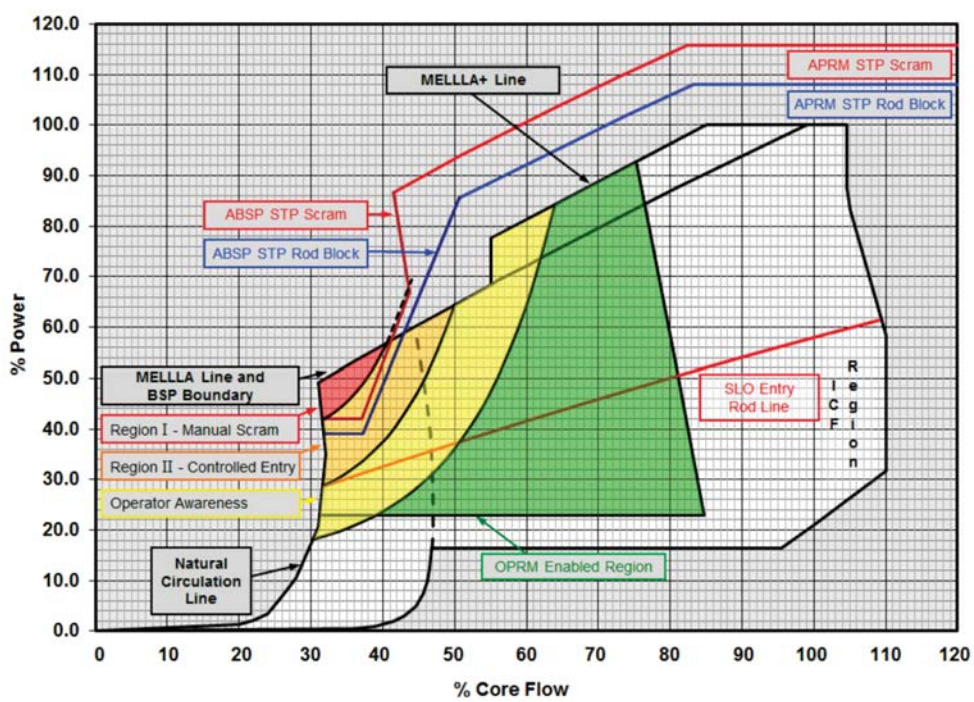

Figure 14: Example Power/Flow Operating Map for Brunswick

In particular, the 2RPT event from the lowest flow at rated power is expected to be the most limiting pump trip event because it starts from operation at the highest control rod line, which results in the highest power level, and therefore, most unstable condition following the recirculation pump trip.

However, there exists a possibility that other events may be limiting, depending on the specific conditions at the plant. The licensee also analyzed a 2RPT from the lowest flow point at rated core power within the MELLLA domain with the minimum allowed FW temperature under FW heater out-of-service (FWHOOS) conditions. A lower FW temperature gives higher core inlet subcooling, which is destabilizing. Note that FWHOOS is not allowed during MELLLA+ ™ operation. Therefore, this initial operating condition in the MELLLA domain may be more limiting than the operating condition in the MELLLA+ ™ domain due to the core inlet temperature difference.

The licensee also evaluated a 1RPT event starting from the highest power level under SLO conditions. This may be a limiting event because it results in flow at natural circulation conditions similar to the 2RPT event. However, the margin to the SLMCPR during the SLO 1RPT event is expected to be higher than the two-loop operation (TLO) 2RPT event due to the power dependence of the OLMCPR.

As stated in Section 7.2 of ANP-3703P (Attachment 15 to the LAR (Reference 1)), the licensee proposes to evaluate all three of these pump trip scenarios **[[ ]]** The NRC staff accepts the disposition that the limiting stability event will be one of these three events based on past experience and consistency with previous applications of Option III and EO-III. The licensee also proposes to **[[ ]]** The NRC staff's evaluation of this proposal is given in Section 3.6.2.7.2.

3.6.2.4 <u>Review of ANP-3703P, Section 4.0, "Evaluation Model Requirements"</u>

RAMONA5-FA is currently approved for DIVOM analyses within the Option III and EO-III methodologies. These DIVOM analyses involve calculation of the system stability response starting from natural circulation conditions after the pump trip has completed. These analyses must be able to accurately calculate the MCPR response as a function of oscillation amplitude as the oscillations grow. However, the magnitude of oscillations that occur before they are suppressed by a trip, is determined separately from the RAMONA5-FA calculations in these previous methodologies.

The BEO-III methodology is used to determine **[[ ]]** the MCPR response during unstable oscillations, as in the Option III and EO-III methodologies. Therefore, the evaluation model requirements[7] related to the growth of oscillations and associated MCPR response are the same for BEO-III as in these previous methodologies.

However, unlike the Option III and EO-III methodologies, the BEO-III RAMONA5-FA analyses start from normal operating conditions and explicitly model the recirculation pump trip and

[7] Note that the term "evaluation model requirements" is used in the sense specified in RG 1.203, which describes the evaluation model development and assessment process (i.e., EMDAP).

associated core inlet flow and temperature response. Therefore, accurate modeling of the time-dependent plant response following a recirculation pump trip is required for BEO-III as well.

BEO-III directly analyzes the time-dependent 3D power distribution provided by RAMONA5-FA in order to simulate the OPRM response and generation of the reactor trip signal. The PBDA was implemented into RAMONA5-FA for this application. However, the CDA is the licensing basis stability detection algorithm at Brunswick. Therefore, Duke Energy used the simulated OPRM data provided by RAMONA5-FA and determined the CDA trip time as a post-processing step using a customized code referred to as CDACALC. This gives rise to the additional evaluation model requirement that the OPRM calculation in RAMONA5-FA and the simulated CDA response determined by CDACALC must accurately reflect the actual implementations at Brunswick.

The licensee developed a phenomena identification and ranking table (PIRT) to determine which model uncertainties are important in determining the core MCPR FoM and the ICO FoM as defined in Section 2.0 of ANP-3703P (Attachment 15 to (Reference 1)), "Regulatory Requirements Summary." This table summarizes all the relevant phenomena and provides an importance ranking with respect to each FoM. The NRC staff evaluated the BEO-III'S PIRT in detail due to its importance in determining the evaluation model requirements for BEO-III, as well as in defining the uncertainty parameters included in the statistical uncertainty analysis performed for BEO-III.

Based on its review of the BEO-III'S PIRT, the NRC staff finds that the licensee identified all significant parameters that are relevant to the FoMs and that appropriate importance rankings were assigned to each of them. The licensee considered not only phenomena that impact the neutronic and thermal-hydraulic dynamics of the core during oscillations, but also phenomena that impact the plant and vessel response following a recirculation pump trip. The NRC staff determined these phenomena and their rankings to be consistent with the current state of understanding of BWR oscillations. In order to make this determination, the NRC staff reviewed PIRTs developed in 2001 and 2011 under the guidance of the NRC (Reference 53) and Section 5 of (Reference 54)), more recent NRC-published studies of ATWS-I scenarios (Reference 55) and (Reference 56), and other available sources of information from open literature or internal NRC experience based on reviewing ATWS-I methodologies.

The NRC staff also compared the BEO-III'S PIRT to the ATWS-I PIRT presented in ANP-10346P (Reference 50). Although the FoMs are not identical, the NRC staff expected that many of the same phenomena would be identified in both PIRTs due to the similarity of the two applications. This was found to be the case, as all relevant phenomena in the ATWS-I PIRT were considered in the BEO-III'S PIRT as well. Furthermore, the importance of these phenomena was indicated as the same or higher in BEO-III relative to ATWS-I, which is consistent with the NRC staff's expectations.

Similar to reviews of previous statistical analysis methods (e.g., ANP-10300P, which contains Framatome's AURORA-B evaluation model for AOOs (Reference 33)), the NRC staff considers that medium- and high-ranked phenomena should be included in the statistical analyses in order to sufficiently determine the impact of uncertainties on the FoMs. Although ANP-3703P proposed sampling for relevant high-ranked phenomena, medium parameters were originally excluded from the statistical analysis. The treatment of medium-ranked parameters in the statistical analysis is addressed in RAI 26, which is discussed below in Section 3.6.2.6.

The NRC staff further identified three phenomena that were dispositioned as being of low importance in the BEO-III PIRT, but which the NRC staff considered to have a potentially significant impact for stability. Additionally, in some cases, these parameters were included in the AURORA-B AOO statistical sampling for non-pressurization transients, which uses similar methods as BEO-III. These three phenomena are:

- **[[**

]]

The NRC staff issued RAI 29 to address whether the disposition of these parameters as low importance, and therefore, excluding them from the BEO-III statistical sampling, was appropriate for Brunswick.

[[

]], the NRC staff finds that the disposition of **[[** **]]** as low importance is acceptable, and it may be excluded from the **[[** **]]**.

[[

]] The NRC staff finds that this assumption is conservative **[[**

]]
Therefore, the NRC staff finds this treatment of **[[** **]]** to be acceptable and finds that no **[[** **]]** of this parameter is needed.

[[

]] To justify this position, the licensee **[[**

]] and the NRC staff agrees that this parameter is physically not expected to be of significant importance relative to other parameters. Therefore, the NRC staff finds the disposition of **[[** **]]** to be acceptable, and no **[[** **]]** of this parameter is needed.

Because of its consistency with the NRC staff's understanding of BWR stability and the similarity to previous stability PIRTs, the NRC staff finds the BEO-III'S PIRT presented in ANP-3703P to be acceptable for its application to Brunswick.

3.6.2.5 Review of ANP-3703P (Attachment 15 to (Reference 1)), Section 5.0, "Method Adaptations for BEO-III"

The version of the RAMONA5-FA code used for BEO-III is identical to that used in the approved EO-III and Option III methodologies, with several exceptions that are discussed and evaluated in the following sections of this evaluation.

3.6.2.5.1 Review of ANP-3703P, Section 5.1, "**[[** **]]** Fuel Rod Models"

The fuel rod model impacts the thermal energy stored in the fuel rod and the heat that reaches the cladding surface and coolant during thermal-hydraulic oscillations. Therefore, the BEO-III model must adequately determine the initial condition of the fuel rod, the change in fuel rod conditions following the initiating event (e.g., 2RPT), and the change in fuel rod conditions during growing oscillations up until oscillations are suppressed by a scram.

For the BEO-III methodology, **[[**

]] In Section 3.5.2.4 of this SE, the NRC staff concluded that the **[[** **]]** fuel rod model acceptably simulates fuel behavior under the full range of conditions expected for ATWS-I.

Limiting ATWS-I events, such as 2RPT, are identical to LTSS events except that the ATWS-I events are not terminated by a reactor scram. Therefore, the ATWS-I methodology must determine the fuel rod behavior under the same conditions as for BEO-III, as well as under larger-amplitude oscillations in the absence of scram. Therefore, the same evaluation given in Section 3.5.2.4 of this SE can be used to justify the fuel rod model in the BEO-III methodology. Additionally, the experimental benchmarking performed for BEO-III indicated no observable bias that would indicate a deficiency in the fuel rod modeling. For these reasons, the NRC staff finds that the **[[** **]]** fuel rod model is acceptable.

3.6.2.5.2 Review of ANP-3703P, Section 5.2, "Radial Power Deposition Distributions in Fuel Pellets"

The radial distribution of power deposition in the fuel pellets affects the fuel temperature distribution and the rate of heat reaching the cladding and coolant as a function of time during stability events. **[[**

]] Therefore, the NRC staff finds the radial power deposition distribution model, which was found to be acceptable for **[[** is acceptable for BEO-III as well.

3.6.2.5.3 Review of ANP-3703P, Section 5.3, "Period-Based Detection Algorithm Model"

Framatome implemented the PBDA included in the BWROG LTSS Option III solution within RAMONA5-FA to determine the time of scram during the simulated oscillations. However, the licensee has proposed to continue using the General Electric Hitachi proprietary CDA as the licensing basis LTSS algorithm for Brunswick. To support this, Duke Energy reanalyzed Framatome's RAMONA5-FA results using the CDA algorithm as implemented at the plant. Duke Energy's approach relies upon Framatome's conservative implementation of the PBDA, including selected setpoints, which resulted in the CDA trip occurring earlier than the PBDA trip in the large majority of cases. In particular, Duke Energy's approach uses the PBDA trip with conservative setpoints to serve as a convenient method of terminating RAMONA5-FA simulations. Beyond verifying that the PBDA trip would, in the large majority of cases, not interfere with the application of the CDA to the results of the RAMONA5-FA simulations, the details of Framatome's PBDA implementation in RAMONA5-FA do not impact Duke Energy's analysis assumptions or results. As discussed further below in Section 3.6.2.7.3, based upon its review, the NRC staff finds that (1) the conservatism in Framatome's PBDA implementation will ensure that the CDA trip occurs before the PBDA trip in a large majority of cases, and (2) the licensee will ensure that any exceptions are handled in an appropriate manner.

3.6.2.5.4 Review of ANP-3703P, Section 5.4, "Multi-Stage Analysis"

The plant-specific BEO-III methodology employs a "multi-stage analysis" approach to determine both the core MCPR and the ICO for a given statistical case. The multi-stage analysis is a key component of the plant-specific BEO-III methodology that has not previously been reviewed by the NRC staff. Because of its importance for determining the FoMs within the stability methodology, the multi-stage analysis was a focus of significant attention during the NRC staff's review. Details and staff evaluation of each stage of the multi-stage approach are provided below.

[[

]]

The NRC staff evaluated **[[ ]]** to determine its ability to adequately determine **[[ ]]** as a result of core oscillations during the limiting stability events. The NRC staff determined that the limiting stability events were simulated in a realistic manner, accounting for all important physics. **[[**

]] The NRC staff finds that these best-estimate calculations were performed in an acceptable manner and are suitable for use **[[**

]]

[[

]]

The NRC staff reviewed the **[[ ]]** and finds that it is an acceptable means of determining **[[**

]]

However, in RAI 25 (Reference 38), the NRC staff requested additional clarification on the **[[ ]]** analysis approach in order to determine whether all aspects of **[[ ]]** were performed in an acceptable manner. In RAI 25a, the NRC staff requested information on how the duration of **[[ ]]** is determined. In the RAI response, the licensee stated that **[[**

---

[8] **[[ ]]**

]]

The NRC staff finds this approach to be acceptable in principle because **[[**

]]

In RAI 25b (Reference 38), the NRC staff requested information on how the **[[ ]]** is calculated from RAMONA5-FA. In its response, the licensee stated that **[[**

]]

Therefore, the NRC staff finds the approach for calculating **[[ ]]** to be acceptable.

In RAI 25c (Reference 38), the NRC staff requested information on how the perturbation amplitude **[[ ]]** was determined. In the RAI response, the licensee stated that **[[**

[9] Although regulatory audit discussions are non-binding, for general context, **[[ ]]** was discussed as **[[ ]]** in the audit that occurred on March 20–21, 2019 (Reference 32).

]] The NRC staff finds that **[[** **]]** Based on the additional information provided by the licensee, the NRC staff finds that the initial perturbation amplitude was defined in an acceptable manner such that **[[** **]]**

In RAI 25d (Reference 38), the NRC staff requested an explanation for why **[[** **]]** In the RAI response, the licensee stated that **[[** **]]** The NRC staff finds this to be acceptable because **[[** **]]** Therefore, the apparent discrepancy **[[** **]]** does not adversely impact **[[** **]]**

In RAI 25e (Reference 38), the NRC staff requested a plot of core pressure drop **[[** **]]** Thus, the NRC staff finds that **[[** **]]** is acceptable.

Based on the evaluation of the LAR and additional information supplemented by the licensee, the NRC staff finds that the core and ICO MCPR responses are adequately determined by the multistage analysis process.

3.6.2.6 <u>Review of ANP-3703P (Attachment 15 to (Reference 1)), Section 6.0, "Code Validation and Model Uncertainties"</u>

Section 6.0 of ANP-3703P describes the determination of neutronic and thermal-hydraulic modeling uncertainties applicable to the BEO-III statistical analysis, as well as the benchmarking of these models to measured data.

3.6.2.6.1 Model Uncertainties

Table 7-3 of ANP-3703P lists the parameters that were statistically sampled in the BEO-III licensing analyses. Only high-importance parameters were included in the analyses provided in ANP-3703P. However, in its evaluation of the AURORA-B AOO evaluation model in ANP-10300P, which used a similar statistical approach as BEO-III, in response to an RAI from the NRC staff, Framatome also included medium-ranked parameters in the statistical sampling. This was because the combined effect of the medium-ranked parameters on the final 95/95 result was considered large enough to warrant their inclusion, even if the impact of individual medium-ranked parameters may be relatively small. The NRC staff determined that the same reasoning should apply to BEO-III, and therefore, RAI 26 was issued to request that the licensee include the medium-ranked parameters defined in the BEO-III'S PIRT within the statistical sampling or justify their exclusion. This section contains the NRC staff's evaluation of both the original set of parameters in ANP-3703P, as well as additional parameters addressed in RAI 26.

Parameters listed under **[[** **]]** were assigned uncertainties based on **[[** **]]** (Reference 33). The modeling uncertainty of each parameter is determined based on comparison to measured data, **[[** **]]** Because of these considerations, the NRC staff finds that these parameter uncertainties **[[** **]]** are acceptable for use in BEO-III.

The approach for determining parameter uncertainties in the BEO-III methodology includes **[[** **]]** The NRC staff reviewed the new uncertainty methods and determined that they remain within the spirit of the approved methods in AURORA-B AOO. **[[** **]]** Therefore, the NRC staff finds the methods used to determine uncertainties for all sampled parameters to be acceptable. This includes high-ranked parameters included in ANP-3703P, as well as medium-ranked parameters included in RAI 26.

[[**]]** uncertainties were derived based on experimental void fraction data from the FRIGG and KATHY facilities. The FRIGG experiments included legacy geometric designs, while the KATHY experiments included benchmarking of ATRIUM-10 and ATRIUM 10XM fuel bundles. The **[[** **]]** uncertainty was determined based on experimental pressure drop data from KATHY for ATRIUM-10, ATRIUM 10XM, and ATRIUM 11 fuel. The ATRIUM-10 and ATRIUM 10XM designs include part length fuel rods, mixing vane grids, and prototypic axial/radial power distributions, which are reasonably representative of the design features in

ATRIUM 11. The licensee demonstrated the high degree of thermal-hydraulic compatibility between ATRIUM 10XM and ATRIUM 11 in ANP-3643P (Attachment 7 to the LAR (Reference 1)). The NRC staff notes that bundle thermal-hydraulic parameters, including pressure drop and void fraction distributions, depend primarily on bulk quantities such as bundle hydraulic diameter and are relatively insensitive to mild variations in the configuration of flow paths within the bundle. Therefore, the NRC staff finds it acceptable that the thermal-hydraulic uncertainties for ATRIUM 11 fuel were based largely on experimental data for ATRIUM-10 and ATRIUM 10XM rather than ATRIUM 11 exclusively.

[[

]] The NRC staff finds this approach to be acceptable because **[[**
]]

[[
]] Realistic modeling of reactor noise is important for stability calculations because it strongly affects the onset time and initial magnitude of oscillations as the core becomes unstable. The model used to define this random noise, including the values of parameters used to define the noise amplitude, as well as its temporal characteristics, **[[**
]] it provides a realistic representation of the actual noise present in Brunswick in terms of the distribution of amplitude and frequency ranges within the noise signal. Additionally, the noise parameters used for the BEO-III analyses were consistent with those used for the validation cases, which provides confidence that the BEO-III analyses will produce accurate results consistent with the good experimental agreement demonstrated in ANP-3703P. These points were discussed in the evaluation of RAI 6 for the **[[**
]]

However, the random nature of noise means that the results will differ depending on **[[**
]] This may impact oscillation onset timing to some degree, but the most significant effect is the possibility of CDA resets due to the chaotic effects of the applied noise. Such resets can significantly impact the CDA trip time in each statistical trial, and therefore, impact the final 95/95 FoMs. This chaotic effect is not a shortcoming of the model but a realistic representation of actual CDA behavior in the plant.

To ensure that the final MCPR determination accounts for this noise-induced variability, the licensee has **[[**

]] Therefore, the NRC staff finds the inclusion of **[[**
]] to be acceptable. The NRC staff has determined that **[[**

]]

In the RAI 26 response, the licensee proposed to include certain medium-ranked phenomena in the statistical sampling approach. These items are reflected in Table 7. However, the licensee proposed to omit other medium-ranked phenomena from the statistical sampling:

- **[[**

]]

The exclusion of these phenomena is discussed in the following paragraphs.

[[

]]

The NRC staff reviewed **[[** **]]** modeling approach and finds it acceptable that MICROBURN-B2 contains sufficient modeling fidelity to accurately predict **[[**

]] The NRC staff finds that this impact is small enough that this parameter may be excluded from the BEO-III **[[** **]]** without significant adverse impact on the final FoMs.

[[

]]

In its response to RAI 26, the licensee applied **[[** **]]** The NRC staff finds that **[[** **]]** In the RAI 26 response, the licensee indicated that **[[** **]]** Therefore, the NRC staff finds that **[[** **]]** can be acceptably excluded from the **[[** **]]** The NRC staff finds that excluding this phenomenon from the **[[** **]]** due to the system configuration at Brunswick is acceptable.

Also, in the RAI 26 response, the licensee provided updated Brunswick BEO-III results, including the medium-ranked parameters in Table 7. The addition of these medium-ranked parameters led to **[[** **]]** Note that this is the intermediate 95/95 MCPR result using the PBDA, which used a significantly higher amplitude setpoint than the licensing basis CDA analyses performed by Duke Energy. The CDA analyses discussed later in this evaluation resulted in a **[[** **]]** margin to the SLMCPR for the same event (limiting 2RPT from the MELLLA+ ™ domain). This was the result using high-ranked parameters only. The CDA analyses are based on the same RAMONA5-FA calculations provided with the PBDA, and the two stability algorithms are expected to result in similar trends with respect to parameter sensitivities. Therefore, the impact of including relevant medium-ranked parameters on the final 95/95 FoMs with the CDA is expected to remain well within the large existing margin to the SLMCPR for the analyzed cycle-specific condition.

Table 7: Modeling Parameters Included in the BEO-III Statistical Analysis

Category	Parameter	PIRT Importance

3.6.2.6.2 Impact of Core Oscillation Mode

The NRC staff issued RAI 27(1)[10] to address concerns related to the core oscillation mode (in-phase or out-of-phase) and the statistical methodology. The motivation for the request was to help gain an understanding of the physical behavior of the reactor, assess the code predictions against *a priori* expectations, and assure that the uncertainty characterizations are appropriate.

[10] The RAIs issued by the NRC (Reference 38) include two RAIs marked as "RAI-27." To avoid confusion within this evaluation, these RAIs will be referred to as "RAI-27(1)" and "RAI-27(2)," based on the order in which they appear in the issued RAIs.

In RAI 27(1)a, the NRC staff asked whether the BEO-III methodology is capable of characterizing the oscillation mode observed during the RAMONA5-FA simulations. In the RAI response, the licensee stated that **[[**

]] Therefore, the NRC staff finds that this method would be a reasonable diagnostic tool to determine the dominant oscillation mode across various RAMONA5-FA trials (e.g., to confirm that code predictions match expected plant behavior).

In RAI 27(1)b, the NRC staff requested clarification for whether any parameters considered in the statistical analysis may have a significantly different impact on predicted FoMs depending on the core oscillation mode. The licensee stated that **[[**

]]

Based on plots provided in response to RAI 26, the licensee stated that **[[**

]]

In RAI 27(1)c, the NRC staff requested an assessment of which oscillation modes were observed for the BEO-III Brunswick analysis and to what degree this impacted the sensitivity of these parameters. In the RAI response, the licensee **[[**

]]

These results and discussion provided the NRC staff with an increased understanding of the impact of core oscillation mode on the magnitude and spread of MCPR response across the statistical trials. This is in contrast to the Option III and EO-III methodologies, which relied on the conservative assumption of out-of-phase oscillations, even when such oscillations might not be dominant at particular exposure points in the cycle. Nevertheless, the observed behavior is consistent with the NRC staff's expectations and demonstrates the ability of the BEO-III methodology to provide a statistically informed prediction of the dominant mode behavior and associated MCPR response, accounting for input uncertainties and cycle exposure. Thus, the NRC staff finds the licensee's approach to address the core oscillation mode is acceptable.

3.6.2.6.3 <u>Additional Code Validation</u>

The validation of RAMONA5-FA against KATHY void fraction and pressure drop data is discussed and evaluated in the previous section. Additional experimental benchmarking was performed against measured stability data. These include KATHY stability tests, KATHY dryout/rewet tests, linear reactor stability benchmarks, and a nonlinear reactor stability benchmark. The tests encompass a wide range of conditions that provide sufficient coverage of the expected core conditions during anticipated oscillations at Brunswick.

These stability benchmarks were also performed for the plant-specific ATWS-I methodology. The benchmarks include experimental validation for the onset of oscillations, growth of oscillations, occurrence of dryout, and post-dryout behavior. In Section 3.5.3 of this SE, the NRC staff concluded that the RAMONA5-FA ATWS-I code demonstrated close agreement with

the measured data and that this benchmarking was sufficient to justify the use of the RAMONA5-FA ATWS-I code for ATWS-I applications.

The BEO-III methodology analyzes the same physical phenomena as the ATWS-I methodology, with the exception of not treating post-dryout behavior. The NRC staff reviewed the benchmarking results for the BEO-III version of RAMONA5-FA and determined that the agreement with measured data was comparable to what was observed for RAMONA5-FA ATWS-I and that the agreement remains acceptable. Statistical trials were also performed to determine upper 95/95 bounds for DRs, frequencies, and other results across these benchmarks when relevant statistical parameters were considered. The statistical perturbations led to a reasonable degree of variation in the calculated results, and in the majority of cases the 95/95 DR results bounded the experimental data, which is expected. Overall, RAMONA5-FA tended to predict **[[** **]]** However, as discussed in Section 3.5.3 of this SE, this apparent bias in **[[** **]]** did not lead to an unacceptable discrepancy in **[[** **]]**. The NRC staff finds that the licensee's BEO-III methodology, including treatment of uncertainties, is acceptable because its modeling result for the stability response and dryout occurrence during anticipated instability events is consistent with the measured data.

3.6.2.6.4 <u>Timestep Size and Nodalization</u>

Spatial and temporal discretization may impact the stability behavior predicted by system thermal-hydraulic codes such as RAMONA5-FA. It is often found that increasing the timestep size leads to increased oscillation DRs, regardless of oscillation mode due to reduction in numerical damping. Increasing the number of axial nodes in the core may have a similar effect by reducing the numerical damping, as well as increasing the spatial resolution. However, increasing the number of axial nodes in the vessel is only expected to have a significant effect on numerical damping for in-phase modes. This is because the total core flow rate, and therefore, the flow rate in the vessel nodes, is essentially constant during out-of-phase oscillations. In either case, vessel nodalization may also impact the core inlet subcooling by affecting the transport of fluid energy through the vessel as the FW temperature decreases during the event.

The NRC staff issued RAI 24 to request sensitivity studies on timestep size and vessel nodalization. The intent of this RAI was to obtain assurance that potential changes in discretization would not have an undue impact on calculated FoMs or change the sensitivities to statistical parameters.

In the RAI response, **[[**

]] No clear trend was observed with respect to timestep size.

For the vessel nodalization study, **[[**

]]

The range of the impact on core MCPR and ICO results is relatively small across the wide variations in timestep size and nodalization used. More significantly, no clear trend was observed in the 95/95 results as a function of timestep size. For vessel nodalization, a possible trend was observed in the BEO-III MELLLA+ TM sensitivity results, with more limiting MCPR results at finer vessel nodalizations. However, the linear reactor benchmark cases showed conflicting trends with respect to oscillation growth rate. Therefore, no clear trend or only a weak trend was observed for vessel nodalization overall. This lack of clear or strong trends, combined with the good overall agreement with measured data discussed above, provides sufficient justification for the NRC staff to conclude that the "base" vessel nodalization and timestep size parameter values used in ANP-3703P are acceptable for the Brunswick BEO-III analyses.

Core nodalization could potentially impact the BEO-III results as well for similar numerical damping considerations as mentioned above, as well as an impact related to resolving void fraction gradients in the bottom portion of the channel. A sensitivity study on core nodalization was not requested by the NRC staff for BEO-III. However, such a study was performed for the ATWS-I methodology, **[[** **]]**. For ATWS-I, a trend of **[[** **]]** was observed. However, the "base nodalization" of **[[** **]]** axial core nodes was found to be acceptable due to the good agreement it provided with the measured data, whereas **[[** **]]**. Because **[[** **]]**, the NRC staff expects a similar trend would be observed for BEO-III and the same conclusions would apply. Furthermore, the vessel nodalization study, in particular, the timestep size study performed for BEO-III, would be expected to impact the solution in a similar way as a core nodalization study, at least in terms of the impact on numerical diffusion. Therefore, the NRC staff finds sufficient justification to conclude that the base axial nodalization of **[[** **]]** nodes for the BEO-III methodology is acceptable.

3.6.2.7 <u>Review of ANP-3703P, Section 7.0, "BEO-III Cycle Analyses"</u>

3.6.2.7.1 <u>Review of ANP-3703P, Section 7.1, "Statistical Methodology"</u>

The impact of code uncertainties on the 95/95 core MCPR and ICO results was evaluated by the licensee using a statistical process based on non-parametric order statistics. This is a well-established Monte Carlo-based statistical method, and implementations of this method have been approved by the NRC staff in the past, for example, in the AURORA-B AOO TR (Reference 75). This method involves the following steps:

1. selection of a set of model parameters that is expected to provide the largest impact on the 95/95 results,
2. determination of applicable uncertainty values for these variables,
3. execution of a series of statistical trials using random perturbations of these variables within RAMONA5-FA, and
4. determination of the 95/95 results for the FoMs derived from these calculations.

The selection of largest-impact parameters was performed based on the BEO-III'S PIRT provided in Section 4.2 of ANP-3703P. The licensee defined high probability as **[[** **]]** at least 95 percent of the population with

95 percent or greater confidence (95/95). The NRC staff has accepted use of the 95/95 criterion in numerous past reviews as providing sufficient confidence that safety limits and other regulatory criteria are satisfied.

In practice, the 95/95 value for each FoM is determined by sorting the FoM results from all statistical trials at a given exposure point and event condition. Then, the N_{th} most limiting FoM value is selected, where *N* is the acceptance number corresponding to a simultaneous upper tolerance limit with at least 95 percent probability coverage at a 95 percent confidence level for the predetermined statistical sample size. For BEO-III, the consequences of the limiting stability event(s) are determined to be acceptable if **[[ ]]** with 95 percent probability at 95 percent confidence. This means that if either the core MCPR or ICO criterion is not satisfied in a given statistical trial, that trial is considered a failed case. The licensee noted that the required sample size for a given acceptance number is dependent upon the number of parameters being treated simultaneously. The NRC staff finds the statistical approach proposed for the Brunswick-specific BEO-III methodology appropriately ensures **[[ ]]**

Based on its review of the LAR, the NRC staff finds that the same overall statistical approach proposed in BEO-III was previously used in the approved AURORA-B AOO TR (Reference 33). This statistical approach based on non-parametric order statistics provides a broad framework for determining the impact of code uncertainties on relevant FoMs, independent of the actual modeling details and FoMs specific to each application. Furthermore, the licensee proposed administrative controls, similar to those imposed by the NRC staff in its review of the AURORA-B AOO TR, to ensure the fidelity of the statistical analysis. These controls include selection of the number of statistical trials prior to the initiation of statistical calculations and maintaining auditable records demonstrating that the statistical analysis has been performed in an unbiased manner. Therefore, the NRC staff finds the proposed use of non-parametric order statistics to be acceptable for use in BEO-III, provided that the method was implemented appropriately to the LTSS analyses.

To determine the appropriateness of the LTSS implementation, the NRC staff verified that the individual RAMONA5-FA calculations were performed in an acceptable manner. The calculations realistically modeled the system response during the entire event progression from the initiating pump trip until oscillation suppression and the most limiting potential stability events were considered, as discussed in Section 3.6.2.3 of this SE. Furthermore, input assumptions, including the timestep size and nodalization, were found to be acceptable, as discussed in Section 3.6.2.6 of this SE.

Additionally, the NRC staff determined that the FoMs – both the core MCPR FoM and the ICO FoM, were selected appropriately within the BEO-III framework to ensure compliance with GDCs 10 and 12. In the absence of individual channel oscillations, the core MCPR FoM determines the limiting MCPR response in the core during oscillations. The ICO FoM is used to ensure that any ICOs that may occur during such events will not lead to a more limiting MCPR response, and therefore, challenge the SLMCPR. Core oscillations and ICOs are the two fundamental types of oscillatory phenomena in BWRs that may challenge the SLMCPR during anticipated stability events. The inclusion of these two FoMs allows the methodology to provide adequate assurance that the safety criteria are met for all anticipated oscillation types at Brunswick.

In summary, the licensee proposed an acceptable non-parametric order statistics process, applied this process to suitable LTSS calculations, determined statistical parameters and uncertainties appropriately, and established acceptable FoMs to ensure that relevant safety limits are not violated. Thus, the NRC staff finds the statistical methodology proposed by the licensee is acceptable, provided that an appropriate calculation procedure is used to apply it to cycle-specific calculations at Brunswick. The calculation procedure is evaluated in the following section to confirm this condition is satisfied by the licensee.

3.6.2.7.2 Review of ANP-3703P, Section 7.2, "BEO-III Calculation Procedure"

Section 7.2 of ANP-3703P defines a calculation procedure that will be used on a cycle-specific basis to determine that stability events will not challenge the SLMCPR. Framatome obtained Brunswick ATRIUM 11 equilibrium cycle results using this procedure, based on reactor trip timings using the PBDA. These results are shown in ANP-3703P, Section 8.0.

Framatome provided the RAMONA5-FA outputs from these Brunswick ATRIUM 11 equilibrium cycle analyses to Duke Energy, who performed additional calculation procedure steps to demonstrate licensing basis SLMCPR protection with the CDA at Brunswick. Therefore, the NRC staff evaluated both components of the calculation procedure. The review of ANP-3703P, Section 7.2, is given in this section. The review of Duke Energy's additional procedure, defined in DPC-NE-1009-P, is given in Section 3.6.2.7.3 of this SE.

Definition of Statepoints

The ANP-3703P, Section 7.2, calculation procedure defines the statepoints to be analyzed.
[[

]] The NRC staff finds the definition of exposure points to be acceptable
[[
]]

[[
]] This is consistent with the previous methodologies and the NRC staff finds it remains acceptable for BEO-III at Brunswick.

The procedure proposes that three events **[[**
]]

1. a two-pump trip from rated power at the lowest licensed core flow with nominal rated subcooling (MELLLA+ ™ event),

2. a two-pump trip from rated power at the lowest licensed core flow that allows FWHOOS, with increased subcooling corresponding to the minimum allowed FW temperature (MELLLA FWHOOS event), and
3. a single-pump trip from the highest power under SLO, with nominal subcooling (SLO event).

[[

]]

The NRC staff expects the most limiting event in terms of final MCPR margin to be one of these three events, and inclusion of these events **[[** **]]** is consistent with Option III and EO-III. In general, oscillations will grow faster, and therefore, may exhibit the largest delta-MCPR response at the time of trip, at higher rod lines. The MELLLA+ ™ event provides the highest allowable rod line at rated power and is likely to be the most limiting event. However, the growth rate of oscillations also increases with core inlet subcooling, so the MELLLA FWHOOS event shall be analyzed as well. The SLO event is included because this case provides a smaller decrease in flow rate during the event, and therefore, a smaller initial increase in MCPR margin, relative to the TLO operating points. This may compensate for the slower oscillation growth rate expected for this case.

The NRC staff finds **[[**

]] The stability characteristics and dynamic system response may change somewhat across typical reload cycles, but at least to a reasonable degree, such changes would be expected to have a similar impact on the results for all three events. **[[**

]] to assure compliance with GDC 10 and 12, the licensee must ensure, with significant margin, that any cycle-specific differences will not **[[** **]]** relative to the most recent analysis **[[** **]]**

The NRC staff issued RAI 23 (Reference 38) to request description and justification for the process used to determine whether the analysis remains bounding when actual cycle operation deviates significantly from the intended cycle design. In the RAI response (Reference 6), the licensee discussed the existing process that is in place to address such deviations in actual cycle operation. Under this process, Duke Energy would notify Framatome of potentially significant modifications to cycle operation, and Framatome would either determine that the impact of the change is minor or that **[[** **]]** must be rerun to ensure that the original analysis remains bounding. This determination is based largely on engineering judgement and prior experience.

To support this determination, the licensee regularly performs projections of plant operation from the current exposure to the end-of-full-power cycle exposure to ensure that the projected axial power distribution at end of full power remains bounding relative to the actual distribution. This check is also required for significant deviations in control rod pattern from what was

assumed. An up-to-date core follow and depletion calculation is provided to Framatome and used to determine important core parameters. These parameters are then assessed for each event to determine if their impact is large enough to invalidate the original analysis.

In the event that actual cycle operation is not expected to be protected by established operating limits, Framatome uses historic operating data and the projected depletion to EOC to establish new appropriate operating limits. The NRC staff reviewed the information presented and finds this process for addressing unanticipated operating cycle changes to be reasonable and consistent with general industry practice. However, the representativeness of the specific historical operating data and depletion projections that may be used to address unanticipated operating cycle changes in future cycles is beyond the scope of the present review. In accordance with Generic Letter 88-16 and subject to the provisions of 10 CFR 50.59, licensees typically perform cycle-specific core reload analyses without prior NRC staff review. By the same token, modifications to cycle-specific reload analyses to address unanticipated operating cycle changes may also be performed without prior NRC staff review if the provisions of 10 CFR 50.59 are satisfied. The NRC staff notes that changes made by licensees under the 10 CFR 50.59 process are subject to oversight through the NRC's inspection program. Therefore, the NRC staff finds that the licensee will appropriately address unanticipated changes and the cycle-specific BEO-III analyses will remain bounding or will be updated to appropriately account for unanticipated variations in cycle operation.

In addition, ANP-3703P, Section 7.2, specifies that the RAMONA5-FA jet pump model **[[**

]]. The NRC staff finds this acceptable because lower core flow rates promote more unstable oscillations; hence, this approach conservatively accounts for the differences between **[[**

]]

<u>Confirmation of SLMCPR Protection</u>

Under the procedure in ANP-3703P, Section 7.2, as amended by the response to RAI 25a, the OLMCPR is confirmed to protect the SLMCPR if **[[**

]] the methodology finds that the existing OLMCPR is adequate to protect against postulated core oscillations. Otherwise, the OLMCPR must be modified **[[** **]]** to protect the SLMCPR, or additional actions such as modification of the cycle design are required.

ICOs are significantly more likely for pump trips starting from the MELLLA+ ™ domain, as these oscillations typically only occur deeper into the unstable region (upper left corner) of the power-flow map (see Figure 14) relative to core oscillations. In the generic EO-III methodology, which is approved for EFWs (e.g., the MELLLA+ ™ operating domain), ICOs were precluded by establishing a channel instability exclusion region. This was done because ICOs lead to a breakdown of the relationship between delta-MCPR and oscillation magnitude (i.e., DIVOM), which forms a central component of that methodology. Thus, it was determined that the methodology could not be guaranteed to protect the SLMCPR in the presence of ICOs.

Hypothetically, a similar philosophy for ICOs could have been adopted in BEO-III by
[[

]]

The SLMCPR must be protected in the presence of full-core oscillations, ICOs, or both modes at once. **[[**

]] This could occur, hypothetically, by constructive interference depending on the timing and location of both modes in the core. The NRC staff issued RAI 30 to address this concern.

In the RAI 30 response (Reference 6), the licensee stated that **[[**

]]

Such behavior was demonstrated in a sample study provided in Section 5.4 of ANP-3703P, **[[**

]] This was an isolated study rather than a comprehensive examination across multiple operating points; however, the NRC staff's previous experience with simulations of multiple simultaneous oscillation modes does confirm the presence of nonlinear mode interactions and the ability of more unstable modes to suppress less unstable modes under certain conditions; for example, the regional core mode may suppress the global core mode and vice versa. Interference may also occur between full-core oscillations and ICOs; however, if the two modes are similarly unstable or even if the ICO is slightly more unstable than the full-core mode, it seems likely that the full-core oscillations may still tend to suppress the ICO. This is because in the unstable channel, the influence of the full-core mode may be supported and strengthened to some degree by the oscillations occurring in neighboring bundles and throughout the core. An additional consideration is that ICOs are only expected to become unstable after core

oscillations have already grown to a significant amplitude, which allows the full-core mode to more readily suppress any ICOs that may otherwise develop.

The NRC staff finds the licensee's operational requirement that **[[ ]]** to be an acceptable means of ensuring that the ICO mode is not strong enough to decouple from the core oscillation mode, because the presence of core oscillations will suppress ICOs in such a way that the ICOs will not adversely impact the combined MCPR response when this operational requirement is maintained. Thus, the NRC staff finds that the calculation procedure in ANP-3703P, Section 7.2, provides an acceptable means of demonstrating SLMCPR protection during all anticipated full-core and individual channel oscillations at Brunswick, provided that the process for adapting this procedure with the CDA does not invalidate any of the above assumptions and conclusions. The evaluation of this proviso is given in the next section.

3.6.2.7.3 <u>Review of DPC-NE-1009-P, "Brunswick Nuclear Plant Implementation of Best-Estimate Enhanced Option-III"</u>

The procedure described by the licensee in DPC-NE-1009-P to determine licensing basis SLMCPR protection at Brunswick is evaluated in this section. As discussed above, this additional procedure was necessary because Framatome provides BEO-III analysis results to Duke Energy based on an implementation of the PBDA, whereas the licensing basis stability algorithm at Brunswick is the CDA that is proprietary to General Electric Hitachi.

The main steps in this procedure, which will be used for each cycle reload analysis, are summarized as follows:

1. determine the BEO-III cases to analyze with the CDA,
2. post-process the RAMONA5-FA output for the selected cases to determine the CDA trip time, if such a trip occurs,
3. determine the 95/95 F/I MCPR response at each statepoint based on the CDA trip times,
4. determine the minimum required OLMCPR based on the 95/95 F/I MCPR result, and
5. confirm that this required OLMCPR is bounded by other transients.

The licensee's proposed implementation of these steps is evaluated below.

It is important to note that in the RAMONA5-FA results provided by Framatome, the calculations are terminated at the time of oscillation suppression via the PBDA. Therefore, in the event that a CDA trip does not occur prior to the PBDA trip, the licensee cannot determine the exact CDA trip time and associated MCPR response. However, the PBDA settings used by Framatome in the sample equilibrium cycle analyses, including a higher amplitude setpoint, caused the PBDA trip to occur later than the CDA trip in the large majority of cases. The PBDA trip occurred before a CDA trip in only 10 of **[[ ]]** statistical cases analyzed with the CDA for the Brunswick equilibrium cycle.

Certain assumptions used by the licensee rely on the expectation that the CDA will trip earlier than the PBDA. The delta-MCPR due to oscillations increases over time as the oscillations grow and as the core inlet subcooling increases. Therefore, the MCPR response with PBDA will bound the MCPR response with CDA due to CDA's earlier trip time in most cases. However, the licensee also considered that the CDA will not always generate a trip earlier than the PBDA,

and an evaluation of the handling of these potentially limiting "suspect cases" is given later in this section.

[[**]]** use the core conditions at the time of suppression with PBDA. **[[**

]]. Instead, Duke Energy performed its ICO FoM assessment using **[[**

]]. The licensee used this consideration as part of its determination for whether the ICO FoM is satisfied.

<u>Determination of Limiting BEO-III Cases</u>

The licensee eliminated particular cases from reanalysis with the CDA based on a determination that these cases would not be among the most limiting CDA cases above the 95/95 threshold for either of the two FoMs. This means that they were determined to have no impact on the final 95/95 results with the CDA.

First, the licensee eliminated all 2RPT cases starting from the 100 percent power/85 percent flow MELLLA+ ™ point that did not result in an MCPR decrease when using the PBDA. In pump trip events, the MCPR typically increases initially due to the decreased power following the trip. The MCPR then decreases as the oscillations grow in magnitude. The most limiting statistical cases are those in which the MCPR decrease due to oscillations significantly outweighs the initial MCPR increase, such that the initial operating margin (i.e., assumed to be the OLMCPR minus the SLMCPR) may potentially be exceeded. Cases in which the oscillatory MCPR decrease is less than the initial MCPR increase are not challenging to the SLMCPR because of the margin that exists between the OLMCPR and the SLMCPR. Such cases are among the least-limiting cases in the statistical analyses.

The possibility exists that the CDA could have tripped later than the PBDA in a small minority of these eliminated cases if they had been analyzed with the CDA. This could lead to some of these cases actually exhibiting a net decrease in MCPR margin with the CDA, as opposed to a net increase with the PBDA. However, both algorithms use a similar underlying period-based approach to oscillation detection, and the CDA would be expected to generate a trip not long after the PBDA trip in these rare cases. This would result in an only slightly more limiting MCPR response with the CDA as compared to the PBDA. Because of the large MCPR margin exhibited by this set of cases (as predicted by the PBDA), a hypothetical small reduction in MCPR margin for a small number of these cases would not be expected to produce any limiting cases when compared to many other more unstable cases in the BEO-III analyses.

The licensee provided evidence indicating that these cases will not likely challenge the ICO FoM either. The licensee provided a plot (Figure 15) of all statistical trials computed by Framatome using the PBDA, sorted by the F/I core MCPR value. **[[**

]]

Therefore, the NRC staff finds that the cases with no decrease in MCPR when using PBDA will not be limiting with the CDA and may be excluded from the CDA analyses.

[[

Case Number

Figure 15: F/I MCPR **[[ ]]**
(DPC-NE-1009-P, Figure 3-2)

]]

A number of SLO-1RPT cases were also eliminated from the CDA analysis for the analyzed ATRIUM 11 equilibrium cycle at Brunswick. Because the initial flow rate is lower in SLO than in TLO, the MCPR increase due to flow reduction is less for SLO than for TLO. However, the OLMCPR is higher in SLO than in TLO (**[[ ]]** versus **[[ ]]** for the sample Brunswick equilibrium cycle calculations). Although the majority of SLO cases exhibited an MCPR decrease during the event, a **[[ ]]** MCPR margin or greater was observed in all statistical cases across all statepoints for SLO. For comparison, the MELLLA+™ 2RPT event had a significant number of cases that exhibited an MCPR margin of less than **[[ ]]**, with the most limiting margin being **[[ ]]**. This indicates that the SLO event is significantly less limiting than the MELLLA+™ 2RPT event for the analyzed ATRIUM 11 equilibrium cycle at Brunswick.

Although the PBDA results indicate that the SLO event is not likely to be limiting when the CDA is applied, the licensee conservatively established a minimum percentage of the limiting SLO cases from across the operating cycle that must be analyzed with the CDA. This minimum percentage is determined as the ratio of 10 times the order statistic selected for the application to the required sample size corresponding to the selected order statistic. Considering the cycle-specific analysis described in DPC-NE-1009-P, which involved a sample size of **[[ ]]**, the **[[ ]]** most limiting statistical trial is used to determine the 95/95 upper tolerance limit at each statepoint, and the licensee required that **[[ ]]** of the most limiting SLO trials from across the cycle (as determined by the PBDA analysis) be analyzed with CDA. As stated

above, the NRC staff expects that the most limiting PBDA cases are likely to be some of the most limiting CDA cases as well, and vice versa. Therefore, analyzing 10 times as many cases as the order statistic provides reasonable assurance that a sufficient number of potentially limiting CDA cases will be analyzed. The decision by the licensee to select the most limiting cases on a full-cycle basis rather than a per-statepoint basis is appropriate because it ensures that a larger number of cases will be analyzed at the most limiting cycle exposure points. In light of the PBDA results that indicate that the SLO event is not likely to be limiting for this cycle at Brunswick in terms of the 95/95 core MCPR FoM, the NRC staff finds the proposed treatment of the SLO event to be acceptable.

The licensee justified that the ICO FoM is not likely to be limiting for SLO. This is based on **[[**

]] Similar to the discussion above, a small number of the excluded SLO cases may have potentially yielded a later CDA trip than the PBDA trip, but in these cases, the CDA trip is expected to occur soon after the PBDA trip. In such a scenario, the channel DR would be only slightly higher when analyzed based on the CDA trip time instead of the PBDA trip time. However, such channel DRs would be expected to remain well below 1.0, due to **[[**

]] Therefore, the NRC staff finds that analysis of SLO cases using the CDA is not necessary in cases where a given reload analysis demonstrates sufficiently large margin using a PBDA-based trip.

For the MELLLA FWHOOS analyses with the PBDA, the large majority of cases **[[**

]]

For the analyzed ATRIUM 11 equilibrium cycle, the licensee concluded that the FWHOOS event at MELLLA **[[** **]]** even when the CDA is used. Therefore, the licensee did not analyze any FWHOOS cases with the CDA for the equilibrium Brunswick cycle. **[[**

]]

The NRC staff notes that the limiting event when using PBDA will not necessarily remain the limiting event when using the CDA, and additional justification is required to eliminate an event from analysis with CDA. **[[**

]] Thus, the NRC staff finds it acceptable to eliminate all FWHOOS cases from analysis with CDA for a given reload analysis, provided that the PBDA-based analyses demonstrate sufficient margin relative to the other event(s) analyzed.

The NRC staff issued RAI 28 to request additional justification for how it will be ensured that any cases excluded from analysis with CDA will not be expected to challenge the limiting 95/95

FoMs, given the potential differences in stability behavior for these cycles relative to the equilibrium cycle. In the RAI response, the licensee stated this justification based on the event. For the MELLLA+ ™ 2RPT event, the analysis of all cases with a F/I MCPR less than 1.0 with CDA will automatically ensure that the most limiting cases are considered, as the number of cases analyzed will adjust to the relative degree of stability of the cycle. Cycles with more limiting PBDA MCPR responses on average will have more cases analyzed with CDA, and vice versa. The NRC staff finds this acceptable because when a large number of cases is analyzed with CDA, the likelihood of capturing the most limiting cases (up to the order statistic) is high. On the other hand, when few cases are analyzed with CDA, the possibility exists for one or more individual limiting cases to be not analyzed, but such a scenario would only arise for relatively stable cycles, and the likelihood of these cycles challenging the safety limits is small.

The previous section discussed that **[[**

]] If these events are analyzed, the minimum population requirement (i.e., the ratio of 10 times the order statistic to the corresponding required sample size) is intended to ensure a sufficiently large sampling population that the most limiting CDA cases will be analyzed, and any switching of order between PBDA and CDA will be of no significant consequence.

As discussed above, this minimum population requirement was proposed as a means of avoiding excessive analyses of cases that are unlikely to be limiting. This potential relaxation may be applied contingent upon a finding for a given cycle that the SLO or FWHOOS event is significantly less limiting overall than the MELLLA+ ™ event in the PBDA analyses. However, if it cannot be justified that **[[**
]], or the PBDA results suggest that either event may be nearly as limiting as the MELLLA+ ™ event, then the justification for the minimum population requirement becomes invalid. In this case, a more comprehensive analysis using the CDA should be performed, for example, using the requirement that all cases for the given event with F/I MCPR less than 1.0 be analyzed.

<u>Handling of CDA Suspect Cases</u>

The technical basis for the process used to select potentially limiting cases to analyze with the CDA relies in part on the assumption that the CDA will trip before the PBDA in the large majority of cases. Due to the particular PBDA settings used by Framatome in its analyses, the PBDA trip occurred earlier than a CDA trip in only 10 of **[[ ]]** statistical cases analyzed with the CDA for the Brunswick equilibrium cycle. These statistical cases encompass a significant range of stability behavior, including oscillation growth ratio, as well as both in- and out-of-phase oscillations. This provides a strong indication that the CDA will continue to trip earlier than the PBDA for the large majority of cases, regardless of anticipated future cycle variations such as operational history and fuel loading. Nevertheless, the NRC staff expects that a small number of future analysis cases may result in the CDA trip occurring after the PBDA trip, which must be dispositioned by the licensee.

For "suspect cases" in which a CDA trip does not occur before the PBDA trip, the licensee proposed to either consider these as failed cases or have the PBDA analyses re-performed using modified PBDA setpoint values to give later PBDA trip times. Considering these cases as failed cases is conservative because it treats the cases as if the FoM acceptance criteria were

violated, regardless of the core MCPR or ICO margin that existed before the simulation terminated. Therefore, the NRC staff finds this approach to be acceptable. In the case of the Brunswick equilibrium cycle, the licensee used this approach for the 10 suspect cases (4 for the MELLLA+ ™ event and 6 for the SLO event), and the 95/95 MCPR and ICO FoMs remained within the acceptable limits.

The second approach of re-performing the PBDA analyses with adjustments to the algorithm to give a later trip time is also acceptable. This is because the PBDA analyses are only used to provide bounding MCPR and ICO response values for the CDA analysis, and the bounding nature of the PBDA results is assured, as long as the CDA trip occurs before the PBDA trip. (Note that no adjustment to the **[[ ]]** or sampled parameters is permitted since resampling may bias the statistical results.) Therefore, either of the two approaches, or a combination of both, provides an acceptable and conservative means of handling suspect cases.

Sample Brunswick Equilibrium Cycle Results

Sample BEO-III results for the Brunswick ATRIUM 11 equilibrium cycle were provided for illustration purposes in DPC-NE-1009-P (Attachments 15 and 16 to (Reference 1)). Analyses using the same calculation procedure will be performed for each cycle to confirm the acceptability of the cycle-specific OLMCPR values. These sample analyses were performed using only the highly ranked parameters given in the BEO-III PIRT; however, reload analyses should use the full set of high- and medium-ranked parameters as shown in Table 7.

Using the 95/95 F/I MCPR values from the BEO-III analyses, the minimum stability OLMCPRs were **[[ ]]** for the MELLLA+ ™ event and **[[ ]]** for the SLO event. This provides a **[[ ]]** and **[[ ]]** MCPR margin relative to the actual OLMCPR values expected for MELLLA+ ™ and SLO, respectively. As expected, these analyses using the licensing basis CDA algorithm produced larger MCPR margins than the PBDA analyses with relaxed setpoints, which yielded MCPR margins of **[[ ]]** for MELLLA+ ™ and **[[ ]]** for SLO for the same set of statistical parameters.[11] This is true even with the conservative handling of CDA suspect cases discussed above.

The 95/95 ICO evaluation identified **[[ ]]** MELLLA+ ™ cases with a channel DR greater than 1.0. For **[[ ]]** cases, the **[[**]] remained bounded by the **[[ ]]**, which satisfies the acceptance criterion with respect to the ICO FoM.

Minimum Oscillation Period

The Brunswick CDA implementation includes a time period lower limit (T_{min}) of 1.0 seconds. Use of this value for T_{min} was originally requested in the Brunswick MELLLA+ ™ LAR (Reference 76) and approved by the NRC staff in the Brunswick MELLLA+ ™ SE (Reference 37). The value of T_{min} establishes the lowest oscillation period the algorithm will detect as being a thermal-hydraulic oscillation. Therefore, the period for all anticipated

[11] These values were obtained using only high-ranked parameters in the PBDA analyses. As shown in the RAI-26 response, the MCPR margin was **[[ ]]** for MELLLA+ ™ when both high- and medium-ranked parameters were included. The SLO analyses with the larger set of parameters were not performed as part of the RAI-26 response.

oscillations at Brunswick must remain above 1.0 second in order for the LTSS methodology to provide SLMCPR protection.

In RAI 27(2),[12] the NRC staff requested analyses of a 1RPT event starting from a 100 percent power/85 percent flow condition to ensure that predicted oscillation period values remain above the T_{min} of 1.0 second in the Brunswick BEO-III analyses. The results presented in the RAI response demonstrate **[[**

]] These analyses confirmed that the oscillation period will remain within the required range for all anticipated oscillations at Brunswick. Therefore, the NRC staff finds that the T_{min} value of 1.0 second continues to be acceptable for Brunswick.

Backup Stability Protection

In the event that the OPRM system is unavailable, backup stability protection (BSP) is used to ensure that core oscillations that may violate the safety limits will not occur. The BSP requirements, including requirements for both automatic and manual BSP, are included in the TSs at Brunswick. The BEO-III methodology does not change the TSs related to BSP, and the introduction of ATRIUM 11 fuel does not impact the ability of the BSP to perform its intended function. Therefore, the NRC staff finds that the current BSP implementation at Brunswick remains acceptable.

3.6.3 Stability Analysis Conclusions

Based upon its review, the NRC staff finds that the plant-specific BEO-III calculation procedure in ANP-3703P with the PBDA, combined with the calculation procedure in DPC-NE-1009-P with the CDA, provides an acceptable means of determining licensing basis SLMCPR protection during anticipated stability events at Brunswick. As discussed in the foregoing evaluation, the NRC staff's conclusion relies upon the licensee adhering to the following conditions:

1. As discussed above in Section 3.6.2.5.4, **[[**

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2. Cycle-specific statistical analyses address all parameters shown in Table 7 of this SE in accordance with the uncertainty distributions defined in ANP-3703P and the licensee's response to RAI 26.

3. Cycle-specific analyses **[[** **]]**

4. **[[**

[12] The RAIs issued by the NRC (Reference 38) include two RAIs marked as "RAI-27." To avoid confusion within this evaluation, these RAIs will be referred to as "RAI-27(1)" and "RAI-27(2)," based on the order in which they appear in the issued RAIs.

]] is permissible, in terms of compliance with GDC 10 and 12, provided that the licensee ensures, with significant margin, that **[[**
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5. In performing cycle-specific analyses with the CDA, a reduced analysis using the minimum population requirement described in Section 3.6.2.7.3 that may permit the exclusion of some trials with a F/I MCPR less than 1.0 is appropriate only for specific events that have been demonstrated to be non-limiting with significant margin in the cycle-specific BEO-III analysis using the PBDA. Otherwise, a more comprehensive analysis be performed using the CDA, for example, using the requirement to analyze all cases for the given event(s) with F/I MCPR less than 1.0.

6. Selected settings and modeling options, including core and vessel nodalization, time step control parameters, and noise parameters, shall be defined consistently with the characteristics of the plant and the validation basis provided in ANP-3703P.

On October 9, 2019, the NRC staff requested addition information (Reference 77) for the licensee to (1) propose a means of implementing the above conditions in a regulatorily binding manner, (2) propose alternative measures that would accomplish the underlying purpose of the above conditions, or (3) justify that the above conditions are not necessary to assure acceptable implementation of the plant-specific BEO-III methodology with the CDA at Brunswick. In response, the licensee proposed license conditions for Brunswick, Units 1 and 2, which would implement the above conditions as follows (Reference 8):

> When determining the core operating limits, the Licensee shall apply the conditions outlined in the NRC's Request for Additional Information dated October 9, 2019, when applying ANP-3703P, BEO-III Analysis Methodology for Brunswick Using RAMONA5-FA, and DPC-NE-1009-P, Brunswick Nuclear Plant Implementation of Best-estimate Enhanced Option-III (i.e., Technical Specification 5.6.5.b.19 and 5.6.5.b.22, respectively).

The six proposed license conditions above for Brunswick, Units 1 and 2, would be implemented in a regulatorily binding manner, which the NRC staff's review found necessary to assure acceptability of the plant-specific BEO-III with CDA approach. Therefore, the NRC staff finds the licensee's proposed implementation of the plant-specific BEO-III calculation procedure in ANP-3703P with the PBDA, combined with the calculation procedure in DPC-NE-1009-P with the CDA (Attachments 15 and 16 to the LAR (Reference 1), respectively), to be acceptable support the TS change LAR for Brunswick.

3.7 Control Rod Drop Accident

3.7.1 Regulatory Evaluation

GDC 13 and 28 are the applicable criteria (see Section 2.0 of this SE) for the CRDA event. GDC 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. 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. In addition, Brunswick 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.

The acceptance criteria for CRDA events to satisfy GDC 28 and 10 CFR 50.67 are currently defined in Chapter 15 of the SRP, which also references SRP Section 4.2. In particular, SRP Section 4.2 (Reference 26) provides an extensive discussion of acceptance criteria related to high temperature cladding failure, pellet clad mechanical interaction induced cladding failure, core coolability, and fission product inventory determination for dose assessment purposes. RG 1.183 and RG 1.195 are also referenced for further guidance related to fission product inventories.

However, the NRC staff is currently developing new guidance for reactivity insertion accident acceptance criteria that will supersede SRP Section 4.2. The draft guidance document, draft guide (DG)-1327, has not yet completed the process to become a final regulatory guide. The licensee indicated (Reference 1) that it intends to adopt the DG-1327 criteria for use in analysis of the CRDA event. The NRC staff does not expect the specified acceptance criteria to change significantly, and the technical basis for use of the DG-1327 criteria is more robustly supported by recent research than the CRDA acceptance criteria that Brunswick is currently licensed under. Therefore, the NRC staff considered the basis for application of the DG-1327 criteria at Brunswick in lieu of Brunswick's current licensing basis or the current guidance in SRP Section 4.2.

3.7.2 Technical Evaluation

In ANP-3174P (Attachment 17 to the LAR (Reference 1)), the licensee provided information and some sample calculations demonstrating how the Framatome CRDA analysis methodology described in ANP-10333P-A (Reference 75) will be applied at Brunswick to evaluate each cycle. The sample calculations were based on the equilibrium core design, but cycle-specific calculations will be performed to support each reload. A comparison of the information provided by the licensee against ANP-10333P-A shows that the licensee demonstrated an acceptable application of the methodology to evaluate the CRDA event for the Brunswick equilibrium core design, with a few plant-specific nuances as discussed below. In the LAR, the licensee also provided information that allowed the NRC staff to confirm that all of the limitations and conditions for ANP-10333P-A were met for the Brunswick application.

In addition to finding that the information provided by the licensee shows that it will correctly apply the Framatome CRDA methodology at Brunswick, the NRC staff makes the following additional findings and observations specific to Brunswick:

- In Appendix B to ANP-3174P, the CHF correlation used for the CRDA calculations is discussed. The range of applicability for the fuel-specific CHF correlations for ATRIUM 11 does not extend to the cold startup conditions that the CRDA analyses are performed at. Instead, the licensee used the **[[ ]]** CHF correlation, which is a generic CHF correlation with a broad range of applicability supported by data. The **[[ ]]** CHF correlation was also approved for use in the Framatome LOCA analysis methodology. This correlation is generally known to be conservative for modern fuel designs, and the only effect on the results from the CRDA analyses is a small change in the total enthalpy due to the fact that most of the heat generation during a CRDA happens so rapidly that the conditions are essentially adiabatic; hence, boiling transition does not occur until after most of the heat has been deposited. Therefore, the

NRC staff finds the **[[ ]]** CHF correlation to be acceptable for use for this purpose.

- The CRDA demonstration calculations utilize the fuel rod failure criteria from DG-1327, which has not yet completed the process towards becoming a final regulatory guide. However, the NRC review and approval of ANP-10333P-A indicates that the methodology is acceptable for use with either the current CRDA acceptance criteria in Appendix B to SRP 4.2 or the new proposed criteria in DG-1327. Furthermore, the NRC has published the technical and regulatory basis for the new acceptance criteria (Reference 78). A review of this information indicates that sufficient evidence exists to support use of the fuel failure threshold curves from DG-1327; thus, the NRC staff finds the proposal to use the DG-1327 guidance in the manner described in ANP-3174P to be acceptable. The NRC staff also noted that the ATRIUM 11 fuel to be loaded at Brunswick utilizes stress relief annealed (SRA) cladding and the current ATRIUM 10XM fuel utilizes SRA cladding with a liner. Therefore, the DG-1327 curves are directly applicable. The most recent version of DG-1327 clarifies that the data supporting the curves for recrystallized annealed (RXA) cladding is solely based on lined cladding, so they may not be applicable to fuel that utilizes unlined RXA cladding. Therefore, if the licensee intends to load fuel with unlined RXA cladding, it would be expected to identify appropriate fuel failure threshold curves. The SRA curves from DG-1327 are acceptable for use with lined and unlined cladding.

- Appendix A of ANP-3174P 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 a limitation and condition requiring the licensee to confirm the applicability of the curve to several local characteristics that may be present in the core being analyzed. This information was presented for the equilibrium core, but the licensee would need to confirm that the evaluation boundary curve is also applicable to ATRIUM 10XM fuel prior to use for analysis of the transition cores.

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 an artificial 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. 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.

3.7.3 CRDA Conclusions

The NRC staff reviewed the information in the licensee's submittal pertaining to the analysis of the CRDA event for Brunswick, Units 1 and 2 (Reference 1). The NRC staff's review was further supported by a regulatory audit (Reference 32), which was used to confirm information included in docketed submittals. 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.

3.8 Impact of Code Error

Near the end of the NRC staff evaluation, a code error was identified by Framatome, which potentially impacts the Brunswick submittal. The error includes **[[**

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The licensee addressed identified the error using the appropriate corrective action process and submitted supplemental information in the LAR supplement (Reference 10). Any additional information that may be developed in the future regarding this issue would be subject to existing regulatory requirements associated with changes and errors discovered in evaluation models, such as those in 10 CFR 50.46 and Criterion 16 of Appendix B to 10 CFR Part 50. Thus, the NRC staff reviewed the supplement and finds there is no adverse impact on the conclusions documented in this safety evaluation because there is adequate margin.

3.9 Technical Evaluation Conclusions

The NRC staff reviewed the licensee's analyses related to the effect of the proposed amendments for Brunswick Steam Electric Plant, Units 1 and 2, 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. The NRC staff further reviewed the licensee's proposed changes to TS 5.6.5.b regarding removal of certain methodologies that will no longer be used, addition of the methodologies approved for use in this SE, and deletion of a footnote that is no longer applicable from Table 3.3.1.1-1, which support adoption of the intended Framatome analysis methodologies as well as the license conditions listed in Section 3.6.3 of this SE and confirmed that they are appropriate and necessary to ensure safe operations. Based on the discussion contained in this SE, the NRC staff finds that the proposed amendments for Brunswick, Units 1 and 2, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the appropriate official for the State of North Carolina was notified of the NRC's proposed issuance of the amendments on December 16, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (84 FR 492, dated January 30, 2019). 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

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8.0 NOMENCLATURE

Acronym	Definition
ΔCPR	Change in Critical Power Ratio
ΔCPR	Transient Change in CPR for a Fuel Assembly
ΔMCPR	Transient Change in MCPR
∆H	transient change in enthalpy
∆H_p	prompt enthalpy rise
∆H_tot	tot total enthalpy rise
2PT	2 pump trip
2RPT	Two Recirculation Pump Trip
ABSP	automated backup stability protection
ABWR	advanced boiling water reactor
ACE	Critical Power Correlation
ADS	Automatic Depressurization System
ADSVOOS	ADS valve out of service
AE	Absolute Error
AFC	Automatic Flow Control
ANF	Advanced Nuclear Fuels
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
APF	Axial Peaking Factor
APF	Axial Peaking Factor
APRM	Average Power Range Monitor
AQL	Acceptable Quality Limit
AREVA NP	AREVA NP Inc.
ARI	Alternate Rod Insertion
ARO	all control rods out
ARQ	Acceptance Review Question
ASME	American Society of Mechanical Engineers
ASTM	ASTM (American Society for Testing and Materials)
ATWS	Anticipated Transient without Scram
ATWS-I	anticipated transient without scram with instability
ATWSP	ATWS Peak Pressure Event with Main Steam Isolation Valve Closure
ATWS-RPT	anticipated transient without scram recirculation pump trip
B&W	Babcock & Wilcox
BATT	Battery
BDHT	Blowdown Heat Transfer
BEO-III	best estimate enhanced option III
BOC	beginning of cycle
BOHL	Beginning of Heat Length
BOL	Beginning of Life
BPWS	banked position withdrawal sequence
BQ	Beta-quench
BR3	Belgian Reactor 3
BSP	Backup Stability Protection
BT	Boiling Transition
BWR	Boiling Water Reactor
BWROG	BWR Owner's Group
CB	Core Bypass

CCD	component calculational device
CCFL	Counter Current Flow Limitation
CCTF	Cylindrical Core Test Facility
CDA	confirmation density algorithm
CEA	Commissariat a l'Energie Atomique
CET	component effects test
CFD	Computational Fluid Dynamics.
CFR	code of federal regulations
CHDR	Channel Decay Ratio
CHF	Critical Heat Flux
CHFR	Critical Heat Flux Ratio
CHFRMIN	Critical Heat Flux Ratio Minimum
CMWR	core average metal-water reaction
CoA	Continuity of Assessment
COLR	core operating limits report
CP	Critical Power
CPR	Critical Power Ratio
CPROM	Critical Power Reduced Order Model
CR	Condition Report
CRDA	control rod drop accident
CRWE	Control Rod Withdrawal Error
CSAU	Code Scaling Applicability and Uncertainty
CSD	Cycle Specific DIVOM
CWSR	Cold-Worked Stress-Relieved
CZP	cold zero power
D&S	Detect and Suppress
DC	Direct Current
DEG	Double-Ended Guillotine
DG	diesel generator
DIVOM	Delta over Initial Versus Oscillation Magnitude
DNB	Departure from Nucleate Boiling
DR	Decay Ratio
ECCS	Emergency Core Cooling System
ECD	Equivalent Circle Diameter
ECPR	Experimental Critical Power Ratio, defined to be the ACE/ATRIUM 10XM calculated critical power divided by the experimentally measured critical power
EFPD	effective full-power days
EFPH	effective full-power hours
EFW	Expanded Flow Window
EHPG	Enlarged Halden Program Group
EM	evaluation model
EMDAP	evaluation model development and assessment process
ENC	Exxon Nuclear Company
EOB	end of blowdown
EOC	End of Cycle
EOCLB	end-of-cycle licensing basis
EOFP	End of Full Power
EOHL	End of Heated Length
EOI	Emergency Operating Instructions

EO-III	Enhanced Option III
EOL	End of Life
EOOS	Equipment Out Of Service
EOP	Emergency Operating Procedure
EPFOD	Extended Power/Flow Operating Domain
EPMA	Electron Probe Micro Analysis
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EXAFS	Extended X-ray Absorption Fine Structure
FANP	Framatome ANP
FCTF	Fuel Cooling Test Facility
FDL	Fuel Design Limit
FFTR	Final Feedwater Temperature Reduction
FGR	Fission Gas Release
FHA	fuel handling accident
FHOOS	Feedwater Heater Out Of Service
FIST	full integral simulation test
FMCRD	fine motion control rod driver
FMM	Fuel Management Manual
FOM	Figure of Merit
FSAR	Final Safety Analysis Report
FTR	Feedwater Temperature Reduction
FWCF	Feedwater Controller Failure
FWHOOS	Feedwater Heater Out of Service
Gd	Gadolinia
GDC	General Design Criteria
GE	General Electric Company
GSF	generic shape function
GT	Guide Tube
HBS	High Burnup Structure
HC	Hot cell
HCOM	Hot Channel Oscillation Magnitude
HEM	Homogeneous Equilibrium Model
HFCL	high flow control line
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HTC	Heat Transfer Coefficient
HTFS	Heat Transfer and Fluid Flow Service
ICA	Interim Corrective Action
ICF	Increased Core Flow
ICO	individual channel oscillations
ICP-MS	Inductively Couples Plasma-Mass Spectrometry
IET	integral effects test
IFA	Instrumented Fuel Assembly
IGSW	Intergranular Gaseous Swelling
IMCPR	Initial Minimum CPR
IN	Information Notice
INEL	Idaho National Engineering Laboratory
ISP	International Standard Problem
JP	Jet Pump

KATHY	KArlstein Thermal HYdraulic test facility
K-I	Katuoka-Ishii Correlation
K-S	Kolmogorov–Smirnov
KWU	Kraftwerk Union AG
LAR	License Amendment Request
LBLOCA	Large Break Loss of Coolant Accident
LCA	Limit Cycle Amplitude
LCL	Lower Confidence Limit
LFA	Lead Fuel Assembly
LFWH	Loss of Feedwater Heater
LHGR	Linear Heat Generation Rate
LHGRFACf	Flow Dependent LHGR Multiplier
LHGRFACp	power-dependent linear heat generation rate multipliers
LHS	Left Hand Side
LOCA	Loss Of Coolant Accident
LOFT	Loss of Fluid Test
LOOP	Loss of Offsite Power
LP	Lower Plenum
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPF	Local Peaking Factor
LPRM	Local Power Range Monitor
LRNB	Load Reject with No Bypass
LTL	Lower Tolerance Limit
LTP	Low Temperature Process
LTR	Licensing Topical Report
LUC	Lead use channel
LUT	Look Up Table
LWR	light water reactor
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MCPRf	Flow-Dependent Minimum Critical Power Ratio
MCPROL	Minimum Critical Power Ration Operating Limit
MCPRp	Power-Dependent Minimum Critical Power Ratio
MELLLA	maximum extended load line limit analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MEOD	Maximum Extended Operating Domain
MLI	Mean Linear Intercept
MLO	Maximum Local Oxidation
MOC	Middle of Cycle
MPa	Mega Pascal
MSIV	Main Steam Isolation Valve
MSIVF	Main Steam Isolation Valve Closure - High Flux Scram Event
MSIVIS	main steam isolation valve in-service
MSIVOOS	main steam isolation valve out-of-service
MWd/kgU	MegaWatt days per kilogram Uranium
MWR	Metal-Water Reaction
NAF	Neutron Absorber Fuel
NCL	natural circulation line
ND	Neutron Detector

NEM	Nodal Expansion Method
NEOC	Near End of Cycle
NFIR	Nuclear Fuel Industry Research
NFT	Nuclear Fuel Type
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
NSE	Normal Spectral Emissivity
NSRR	Nuclear Safety Research Reactor
NSS	nominal scram speed
OI	Oscillation Inception
OLMCPR	Operating Limit Minimum Critical Power Ratio
OOS	Out Of Service
OPRM	Oscillation Power Range Monitor
P	P anisotropy parameter
PA	Postulated Accident
PAPT	Protection Against the Power Transient
PB	Peach Bottom
PBDA	Period Based Detection Algorithm
Pbypass	power below which direct scram on TSV/TCV closure is bypassed
PCF	Power Conversion Factor
PCI	Pellet-Cladding Interaction
PCMI	pellet clad mechanical interaction
PCT	Peak Clad Temperature
PDO	Post Dryout
PHE	peak hot excess reactivity
PHTF	Portable Hydraulic Test Facility
PIE	Post Irradiation Examination
PIRT	phenomena identification and ranking table
PLHR	Part Length Heater Rod
PLR	Part Length Rod
PLU	power load unbalance
PLUOOS	Power Load Unbalance System Out of Service
PNL	Pacific Northwest Laboratory
PPD	Plant Parameters Document
PPR	Pin Power Reconstruction
PPS	Plant Protection System
PRFDS	pressure regulator failure downscale
PRFO	Pressure Regulator Failed Open with no Scram
PROOS	pressure regulator out-of-service
PTD	plant transient data
PWR	pressurized water reactor
QA	Quality Assurance
QAP	Quality Assurance Program
QIA	Quantitative Image Analysis
R	R anisotropy parameter
RAI	Request for additional information
RBM	(control) rod block monitor
RCIC	Reactor Core Isolation Cooling
RDF	recirculation drive flow
RDIV	Recirculation Discharge Isolation Valve

RE	Relative Error
RHR	residual heat removal
RHS	Right Hand Side
RIA	reactivity insertion accident
RIP	Reactor Internal Pump
RLBLOCA	realistic large break loss of coolant accident
RMS	Root Mean Square
ROM	Reduced Order Model
RP	Recirculation Pump
RPF	Radial Peaking Factor
RPS	reactor protection system
RPT	Recirculation Pump Trip
RSAR	reload safety analysis report
RTL	Ramp Terminal Level
RTP	rated thermal power
RX	Recrystallized
RXA	Recrystallized Annealed
SAD	amplitude discriminator setpoint
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss of Coolant Accident
SCIP	Studsvik Cladding Improvement Program
SDE	Statistical Design of Experiments
SE	safety evaluation
SE	Standard Error
SEO	Side Entry Orifice
SER	Safety Evaluation Report
SET	separate effects test
SF	Single Failure
SFEE	Steady Flow Energy Equation
SF-BATT	single failure of battery (DC) power
SF-HPCI	single failure of the HPCI system
SF-LPCI	single failure of an LPCI injection valve
SHTC	Spray Heat Transfer Coefficient
SIP	Scenario Identification Process
SL	Steam Line
SLC	standby liquid control
SLCS	Standby Liquid Control System
SLMCPR	safety limit minimum critical power ratio
SLO	Single Loop Operation
SP	Standpipe
SPC	Siemens Power Corporation
SPCB	Critical Power Correlation
SPT	Stability Protection Trip
SRA	Stress-Relief Annealed
SRP	standard review plan
SRSS	Square Root of the Sum of the Squares
SRV	Safety Relief Valve
SRVOOS	safety/relief valve out-of-service
SS	steady state
SS	Steam Separators

SSTF	Steam Sector Test Facility
STP	simulated thermal power
T/C	Thermocouple
TAF	Top of Active Fuel
TBV	Turbine Bypass Valve
TBVOOS	turbine bypass valves out-of-service
TCD	thermal conductivity degradation
TCV	Turbine Control Valve
TD	Theoretical Density
TER	Technical Evaluation Report
THTF	Thermal Hydraulic Test Facility
TIP	Traversing Incore Probe
TLO	two-loop operation
TLTA	Two Loop Test Apparatus
TM	Thermal Margin
TMMIN	Thermal Margin Minimum
TR	Topical Report
TS	Technical Specification
TSSS	technical specifications scram speed
TSV	Turbine Stop Valve
TT1	Turbine Trip Test 1
TTNB	Turbine Trip No Bypass
TTWB	Turbine Trip With Bypass
TUI	Trans-Uranium Institute
UFSAR	updated final safety analysis report
UO2	Uranium Dioxide
UP	Upper Plenum
UPTF	Upper Plenum Test Facility
USNRC	U.S. Nuclear Regulatory Commission
UTL	Upper Tolerance Limit
UTP	Upper Tie Plate
V&V	Verification and Validation
XANES	X-ray Absorption Near-Edge Structure Spectroscopy
Z4B™	AREVA Proprietary Zirconium Alloy
Zry-2	Zircaloy-2 Alloy
Zry-4	Zircaloy-4 Alloy

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Date: March 6, 2020

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 299 AND 327 TO REVISE TECHNICAL SPECIFICATION 5.6.5b TO ALLOW APPLICATION OF ADVANCED FRAMATOME ATRIUM 11 FUEL METHODOLOGIES (EPID L-2018-LLA-0273) DATED MARCH 06, 2020

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