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

January 13, 2023

Mr. Christopher P. Domingos
Site Vice President
Northern States Power Company - Minnesota
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT NO. 210 RE: REVISED METHODOLOGIES FOR DETERMINING THE CORE OPERATING LIMITS (EPID L-2021-LLA-0144)

Dear Mr. Domingos:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 210 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the technical specifications (TSs) in response to your application dated July 29, 2021, as supplemented by letter dated June 6, 2022.

The amendment revises TS 5.6.3, "Core Operating Limits Report (COLR)," to allow the application of advanced Framatome, Inc., methodologies for determining the core operating limits in support of the loading of the Framatome, Inc. ATRIUM 11 fuel type at Monticello Nuclear Generating Plant. The amendment also revises TS 3.3.3.1 "Reactor Protection System (RPS) Instrumentation," to remove reference to Enhanced Option III which will no longer be used.

Enclosure 2 to this letter contains sensitive unclassified non-safeguards information. When separated from Enclosure 2, this document is DECONTROLLED.

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

Sincerely,

/RA/

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure:
1. Amendment No. 210 to DPR-22
2. Proprietary Safety Evaluation
3. Non-Proprietary Safety Evaluation

cc: Listserv

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 210
Renewed License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (NSPM) dated July 29, 2021, as supplemented by letter dated June 6, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 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.

Enclosure 1

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-22 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 210, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to start up from the spring 2023 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: January 13, 2023

ATTACHMENT TO LICENSE AMENDMENT NO. 210

MONTICELLO NUCLEAR GENERATING PLANT

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Renewed Facility Operating License

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
Page 3	Page 3

Technical Specifications

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

REMOVE	INSERT
3.3.1.1-2	3.3.1.1-2
3.3.1.1-4	3.3.1.1-4
3.3.1.1-10	3.3.1.1-10
5.6-2	5.6-2
5.6-3	5.6-3
5.6-4	5.6-4

2. Pursuant to the Act and 10 CFR Part 70, NSPM to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel);

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

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

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

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. <u>Maximum Power Level</u>

   NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 2004 megawatts (thermal).

2. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 210, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. <u>Physical Protection</u>

   NSPM shall implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	--------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ------------------------------------ Place associated trip system in trip.	12 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
B. -----------NOTE------------ Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ------------------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
	<u>OR</u>		
	B.2	Place one trip system in trip.	6 hours <u>OR</u> In accordance with the Risk Informed Completion Time Program
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
	AND		
	I.2	-------------NOTE---------------- LCO 3.0.4 is not applicable ------------------------------------- Restore required channels to OPERABLE.	120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1	Reduce THERMAL POWER to < 20% RTP.	4 hours

Table 3.3.1.1-1 (page 2 of 4)
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
c. Neutron Flux – High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11(f)(g) SR 3.3.1.1.15	≤ 122% RTP
d. Inop.	1, 2	3(c)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
e. 2-Out-Of-4 Voter	1, 2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	NA
f. OPRM Upscale	≥ 20% RTP	3(c)	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.16	As specified in COLR
3. Reactor Vessel Steam Dome Pressure – High	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 1075 psig

(c) Each APRM / OPRM channel provides inputs to both trip systems.

(f) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative with respect to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(g) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The NTSP and the methodology used to determine the NTSP are specified in the Technical Requirements Manual.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Control Rod Block Instrumentation Allowable Value for the Table 3.3.2.1-1 Rod Block Monitor Functions 1.a, 1.b, and 1.c and associated Applicability RTP levels;

5. Reactor Protection System Instrumentation Delta W value for Table 3.3.1.1-1, Function 2.b, APRM Simulated Thermal Power – High, Note b; and

6. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the Reactor Protection System Instrumentation Period Based Detection Algorithm OPRM Upscale trip setpoints associated with Table 3.3.1.1-1 Function 2.f.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

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

2. NEDO-31960-A, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology", with Supplement 1, dated November 1995

3. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996

4. (Deleted)

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

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

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

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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

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

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

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

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

16. EMF-2361(P)(A) Revision 0, “EXEM BWR-2000 ECCS Evaluation Model,” May 2001

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

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

19. BAW-10247P-A Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” February 2008

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

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

21. ANP-10307P-A Revision 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” AREVA NP, Inc., June 2011

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

23. ANP-10344P-A, Revision 0, “Framatome Best-estimate Enhanced Option III Methodology,” Framatome Inc., March 2021

24. ANP-3857P Revision 2, “Design Limits for Framatome Critical Power Correlations,” Framatome, Inc., July 2020

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

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

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

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

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

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.

ENCLOSURE 3

NON-PROPRIETARY SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 210

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

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

PROPRIETARY SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 210

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated July 29, 2021 (Reference 1), as supplemented by letter dated June 6, 2022 (Reference 2), Xcel Energy (the licensee), submitted a license amendment request (LAR) for Monticello Nuclear Generating Plant (MNGP), to allow application of new methodologies necessary to support a planned transition to ATRIUM 11 fuel.

The supplemental letter dated June 6, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 5, 2021 (86 FR 55016).

2.0 REGULATORY EVALUATION

2.1 Proposed Changes

The proposed technical specification (TS) changes are given in section 2.4 of the LAR. The LAR proposed the following TS changes:

- TS 3.3.1.1:
  - Remove reference to Function 2.g in Required Action A.2 and in Condition B.
  - Remove Action J and relabel following Action K accordingly.
- TS Table 3.3.1.1-1
  - Remove reference to Function 2.g.

- TS 5.6.3, item a.6,
    - Remove "and the EFW – High Setpoints associated with Table 3.3.1.1-1 Function 2.g."
- The LAR proposes to delete the following methodologies from TS 5.6.3.b:
    - XN-NF-84-105(P)(A) Volume 1 and Volume 1, Supplements 1 and 2, XCOBR/A-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis, Exxon Nuclear Company, February 1987.
    - ANF-913(P)(A) Volume 1, Revision 1, and Volume 1, Supplements 2, 3 and 4, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, Advanced Nuclear Fuels Corporation, August 1990.
    - Engineering Evaluation EC 25987, "Calculation Framework for the Extended Flow Window Stability (EFWS) Setpoints," as docketed in Xcel Energy letter to NRC L-MT-15-065, dated September 29, 2015
- The LAR proposes to add the following methodologies from TS 5.6.3.b:
    - BAW-10247P-A, Supplement 2P-A, Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods, Framatome Inc., August 2018
    - ANP-10340P-A, Revision 0, Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods, Framatome Inc., May 2018
    - ANP-10335P-A, Revision 0, ACE/ATRIUM 11 Critical Power Correlation, Framatome Inc., May 2018
    - ANP-10333P-A, Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA), Framatome Inc., March 2018
    - ANP-10300P-A, Revision 1, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios, Framatome Inc., January 2018
    - ANP-10332P-A, Revision 0, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios, Framatome Inc., March 2019
    - ANP-10344P-A, Revision 0, "Framatome Best-estimate Enhanced Option III Methodology," Framatome Inc., March 2021.

2.2 Applicable Regulations

The regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), section 50.36, "Technical specifications." This regulation requires that the TSs include items in five specific categories. These categories include: (1)

safety limits, limiting safety system settings and limiting control settings, (2) limiting conditions for operation (LCOs), surveillance requirements (SRs), (4) design features, and (5) administrative controls.
Regulation 10 CFR 50.36(c)(5), "administrative controls," are stated to be "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." This also includes the programs established by the licensee and listed in the administrative controls section of the TS for the licensee to operate the facility in a safe manner.

Section 50.46, "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors," of 10 CFR is not directly applicable to the ATWS-1 (anticipated transient without scram (ATWS-1)) event because it is intended to address postulated LOCAs (loss-of-coolant accidents) rather than ATWS events. However, this regulation does present a set of acceptance criteria for ensuring adequate cooling of fuel such that significant fuel failures do not occur.

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

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

The MNGP applicable principal design criteria (PDCs) predate 10 CFR 50, appendix A, general design criteria (GDCs), as well as the 1967 Atomic Energy Commission (AEC) Draft GDCs. An evaluation comparing the MNGP applicable PDCs to the AEC Draft GDCs is presented in MNGP Updated Safety Analysis Report (USAR), appendix E (ML21070A105). The NRC staff reviewed the LAR to evaluate the applicability of the Framatome methodologies to MNGP and confirm that their use is within the NRC-approved range of the parameters necessary to support a planned transition to ATRIUM 11 fuel and to verify that the results of the analyses are in compliance with the regulatory requirements. The NRC staff reviewed the LAR to confirm that the intent of the requirements in the equivalent 10 CFR, part 50, appendix A, GDCs given below are met.

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

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

- GDC 13, "Instrumentation and control," requiring that instrumentation be provided to monitor variables and systems over their anticipated ranges to assure adequate safety and that appropriate controls be provided to maintain these variables and systems within prescribed operating ranges.

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

- GDC 20, "Protection system functions," requiring that the protection system 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 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.

- GDC 26, "Reactivity control system redundancy and capability," requiring that two independent reactivity control systems of different design principles be provided, one of which can hold the reactor core subcritical under cold conditions.

- GDC 27, "Combined reactivity control system capability," requiring that the reactivity control systems 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.

- GDC 28, "Reactivity limits," requiring that the reactivity control systems 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 RPV [reactor pressure vessel] internals to impair significantly the capability to cool the core.

- GDC 35, "Emergency core cooling," requiring that a system to provide abundant emergency core cooling is provided 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.

Regulation 10 CFR, part 50, appendix K, consists of two parts:

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

2.3 Applicable Guidance

NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear

Power Plants: LWR Edition" (SRP) Chapter 15.8, "Anticipated Transients Without Scram" (Reference 32), establishes acceptance criteria for ATWS events.

SRP, Section 4.2, "Fuel System Design"; Section 4.3, "Nuclear Design"; and Section 4.4, "Thermal and Hydraulic Design," provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and the thermal and hydraulic design of the core.

Regulatory Guide (RG) 1.236, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," (Reference 28) details acceptable methods and procedures to use when analyzing a postulated control rod drop accident.

RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 29) and RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," (Reference 30) provide guidance for evaluation of accident dose radiological consequence criteria.

3.0 TECHNICAL EVALUATION

3.1 Technical Specification Changes Evaluation

The LAR proposes to revise the current long-term reactor stability solution Enhanced Option-III (EO-III) methodology to the NRC-approved ANP-10344P-A, Revision 0, (Reference 14). Based on this change, the LAR proposed removing the extended flow window (EFW) Stability-High function (Function 2.g) contained within TSs 3.3.1.1 and 5.6.3 because it is not part of the best estimate option (BEO)-III stability methodology. The NRC staff finds these changes acceptable because based on the new NRC approved BEO-III methodology for the long-term stability solution, the EFW stability-high function (Function 2.g) is not needed.

The NRC staff finds the methodologies proposed to be deleted from TS 5.6.3.b acceptable as they are no longer applicable and are being replaced with advanced methodologies. The NRC staff finds that the topical reports (TRs) proposed to be added in TS 5.6.3.b are NRC-approved and, therefore, their addition to the TS core operating limit report (COLR) is acceptable.

3.2 Anticipated Transient Without Scram-Instability (ATWS-I) (ANP-3933P)

The evaluation of the ATWS-I is discussed in ANP-3933P (Reference 4).

3.2.1 Regulatory Evaluation

Including the related GDC described in section 2.2 of this safety evaluation (SE), the following regulatory requirements apply to the ATWS-I evaluation.

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

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

The SRP is the primary regulatory guidance document used by the NRC staff to support review of this LAR. SRP, chapter 15.8, establishes acceptance criteria for ATWS events. SRP 15.8 includes additional GDC beyond those discussed in section 2.2 of this SE; 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 this review.

3.2.2 Technical Evaluation

The ATWS-I transients are simulated using the ANP-10346P-A Revision 0, (Reference 15) RAMONA5-FA computer code. The ATWS-I Analysis methodology using the RAMONA5-FA code was approved by the NRC in October 2019. The scope of the NRC staff's analysis covers the MNGP extended power uprate (EPU)/EFW operating domain with an equilibrium ATRIUM 11 core. The turbine trip with bypass and two recirculation pump trip transients are evaluated.

<u>RAMONA5-FA ATWS-I Methodology</u>

Framatome's generic methodology for ATWS-I calculations completed using RAMONA5-FA are found in ANP-10346. The LAR stated that all calculations were performed in compliance with the conditions of ANP-10346. RAMONA5-FA is used to evaluate the fuel specific portion of the event which confirms the limiting peak cladding temperature is below 2200 degrees Fahrenheit (°F).

<u>CPROM Correlation</u>

The RAMONA5-FA transient model uses the critical power reduced order model (CPROM) correlation for dryout. This correlation was presented in ANP-10346. The CPROM correlation was benchmarked with the ATRIUM 10XM fuel and extended for use with ATRIUM 11 fuel by using the procedure described in Section A.4 of ANP-10346. The advantage of this correlation is that it is well-suited for transient models which include cyclical dryout and rewet with possible failure to rewet.

[[

]] In this data, the mean critical power ratio is **[[ ]]** and the standard deviation of the calculated versus measured critical power for the entire database is **[[ ]].**

The bounds of applicability are determined by the data available to benchmark the correlation. For ATRIUM 11, the LAR states, the **[[**

]]

Turbine Trip with Bypass

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

The analysis for the ATRIUM 11 fuel transition was completed with the same input as the ATWS-I analysis of record. The transient is assumed to initiate from rated power at the EFW boundary. The feedwater temperature decrease is assumed to start at 10 seconds after the valve closure and the temperature is assumed to decrease with a 30 second time constant until the final temperature is reached. The time critical operator action for water level reduction is assumed to begin at 90 seconds. Sensitivity calculations were completed at **[[**

]] The worst-case scenario for peak cladding temperature occurs at **[[**

]] The peak cladding temperature remains below the limit in all calculated scenarios.

Two Recirculation Pump Trip

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

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

]]

3.2.3 Evaluation of ANP-10346 Limitations and Conditions

Limitation and Condition 1

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

Evaluation

ANP-3933P stated that the gap conductance sensitivity study for the limiting event, **[[ ]]** was performed. The overall impact was shown to be within the available margin for ATRIUM 11 fuel with a gap conductance variation of **[[ ]]**.

Limitation and Condition 2

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

Evaluation

ANP-3933P states that steps to ensure the calculation will bound future core designs have been made. ANP-3933P states that the analysis **[[**

]]

Limitation and Condition 3

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

Evaluation

As discussed in Limitation and Condition 2, a bounding analysis was completed.

Limitation and Condition 4

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

Evaluation

As discussed in Limitation and Condition 2, a bounding analysis was completed.

Limitation and Condition 5

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

Evaluation

Analyses were performed for both events and determined that **[[**
]].

Limitation and Condition 6

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

Evaluation

The steam line and valve models were completed using plant geometry and setpoints, allowing for an accurate representation.

Limitation and Condition 7

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

Evaluation

The nodalization was consistent with the nodalization of the benchmarks and sample problems in the original ANP-10346. A study was performed to demonstrate the nodalization reasonably represents the plant.

3.2.4 Conclusions – ATWS-I Analysis

The licensee evaluated the ATWS-I event consistent with NRC-approved ANP-10346 and addressed all applicable limitations and conditions. Therefore, the NRC staff finds the MNGP ATWS-I evaluation for ATRIUM 11 fuel, acceptable.

3.3 Control Rod Drop Accident (CRDA) (ANP-3929P)

The evaluation of the CRDA is discussed in ANP-3929P (Reference 5).

3.3.1 Regulatory Evaluation

GDCs 13 and 28 (see section 2.2 of this SE) along with 10 CFR 50.67 are applicable for the evaluation of CRDA events. GDC 13 addresses the availability of instrumentation to monitor associated systems and variables to assure that adequate safety and appropriate controls keep them within the prescribed operating ranges. GDC 13 applies by ensuring that the limiting system operating parameters and other controls in place (i.e., rod withdrawal limitations) are sufficient to ensure that the CRDA acceptance criteria are not exceeded during a CRDA event. This is satisfied by ensuring that the initial conditions represented in the CRDA analyses are sufficiently representative of the most conservative conditions allowed by the aforementioned controls. Application of GDC 28 requires that the postulated reactivity accident does not impart sufficient damage to impair core cooling capacity and does not damage the coolant pressure boundary greater than local yielding.

MNGP 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 methods and procedures described in RG 1.236, are acceptable when analyzing a postulated control rod drop accident. The licensee used RG 1.236 as regulatory guidance criteria for the CRDA event.

RG 1.236 also references RG 1.183 and RG 1.195 for evaluation of accident dose radiological consequence criteria.

3.3.2 Technical Evaluation

The applicability of AURORA-B CRDA methodology to ATRIUM 11 fuel in discussed in Section 6.4 of ANP-3924P (Reference 3). A summary of the application of the AURORA-B CRDA methodology (ANP-10333P-A (Reference 16)) to a MNGP equilibrium cycle along with sample calculations are provided in ANP-3929P. The methodology presented in ANP-3929P includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic (T-H) and fuel temperature feedback to demonstrate consistency with the guidance in RG 1.236. The methodology demonstration involves sensitivity studies and determination of an evaluation boundary for ATRIUM 11 fuel, with similar process to be followed for the transition cores.

The NRC staff notes that comparison of the methodology provided in ANP-3929P against ANP-10333P-A to evaluate the CRDA event for the MNGP equilibrium core design demonstrated an acceptable application of the methodology. Section 8 of the ANP-3929P provides information on limitations and conditions for ANP-10333P-A. Based on this discussion, the NRC staff finds that limitations and conditions of ANP-10333P-A were met.

In addition to finding that the information provided in the LAR, as supplemented, shows that the Framatome AURORA-B CRDA methodology will correctly be applied at MNGP, the NRC staff makes the following additional findings and observations specific to this LAR:

- The AURORA-B CRDA analysis methodology described in ANP-10333P-A was reviewed and approved by the NRC before the issuance of RG 1.236. Use of a nodal three-dimensional kinetics solution with both T-H and fuel temperature feedback is consistent with RG 1.236.

- With regards to the hydrogen model, RG 1.236 specifies that the threshold curves for evaluation of pellet-cladding mechanical interaction failure should include the hydrogen within the oxide layer. **[[**

  ]] Based on data presented in ANP-3929P, the NRC staff finds **[[**
  acceptable for evaluation of RG 1.236 threshold curves for evaluating pellet-cladding mechanical interaction failure.

- To assess radiological consequences of the postulated CRDA, the licensee determined total release fraction (TOTR) using Licensing Basis Release Fraction (LBRF) from RG 1.183 conservatively applied as the steady state release fractions (SSRF), with the transient fission gas release (TFGR) as described in RG 1.236. Three nuclide-specific multipliers (GMUL) are established in DG-1327 to be applied to the TFGR term.

  The NRC staff plans to move TFGR models and nuclide-specific multipliers to RG 1.183 and RG 1.195 but retained TFGR fractions in Appendix B of RG 1.236 until RG 1.183 and RG 1.195 can be updated. Nuclide-specific multipliers were not reproduced in Appendix B of RG 1.236 but were included in the proposed revision 1 to RG 1.183 (DG-1389 (Reference 42)), which was noticed for public comment on April 21, 2022. Based on comparison of value used in ANP-3929P with NRC guidance, the NRC staff considers the nuclide weights used by the licensee acceptable.

- Limitation and Condition 31 as part of the ANP-10333P-A states that the licensee should confirm the applicability of the evaluation boundary curve to several local characteristics of the fuel being evaluated. Appendix A of ANP-3929P describes the process used to establish an evaluation boundary curve to simplify the calculations. The local characteristics of fuel used to establish the evaluation boundary with respect to design core conditions are presented for the ATRIUM 11 core. The licensee would need to confirm that the evaluation boundary curve is also applicable to current 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. The licensee derived the acceptance criteria from the approved methodology for CRDA analysis and demonstrated the determination for whether fuel failures would occur. The radiological consequences evaluation was considered based on an artificial fuel failure scenario to demonstrate that performed calculations and evaluations are in a manner consistent with the approved methodology and demonstrated acceptance criteria are met. Based on this, the NRC staff finds that the proposed adoption of the CRDA analysis methods as part of the transition to ATRIUM 11 fuel is acceptable.

3.3.3 Evaluation of ANP-10300P-A Limitations and Conditions for CRDA

The AURORA-B methodology has limitations and conditions listed in section 5.0 of the NRC staff's SE for ANP-10300P-A, Revision 1 (Reference 17). The statements addressing the limitations and conditions related to CRDA analysis are provided in ANP-3929P, Section 8.0. The NRC staff evaluation of the limitation and conditions is given below.

Limitation and Condition 1

> AURORA-B may not be used to perform analyses that result in one or more of its CCDs [component calculational devices] (S-RELAP5, MB2-K, MICROBURN-B2, RODEX4) operating outside the limits of approval specified in their respective TRs, SEs, and plant-specific LARs. In the case of MB2-K, MB2-K is subject to the same limitations and conditions as MICROBURN-B2. *(This is Conditions 1 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

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

Limitation and Condition 14

> The scope of the NRC staff's approval for AURORA-B does not include the ABWR [advanced boiling water reactor] design. *(This is condition 14 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

This limitation and condition is not applicable because MNGP is not an ABWR design.

Limitation and Condition 20

> The implementation of any new methodology within the AURORA-B EM [evaluation model] (i.e., replacement of an existing CCD) is not acceptable unless the AURORA-B EM with the new methodology incorporated into it has received NRC review and approval.

> An existing NRC-approved methodology cannot be implemented within the AURORA-B EM without NRC review of the updated EM. *(This is a revised version of Condition 20 of the SE for the base AURORA-B TR, rewritten to be specific to the CRDA application. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

The NRC staff finds this limitation and condition is satisfied because the evaluation model described in the base AURORA-B and AURORA-B CRDA TRs will be implemented with no CCD being replaced as described in the TR.

<u>Limitation and Condition 21</u>

> NRC-approved changes that revise or extend the capabilities of the individual CCDs comprising the AURORA-B EM may not be incorporated into the EM without prior NRC approval. *(This is Condition 21 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

The NRC staff finds this limitation and condition is satisfied because the Framatome regulatory procedures require use of NRC-approved methodologies within the applicability defined for that methodology. The LAR provided additional detail on use of **[[** **]]** Critical Heat Flux (CHF) for the cold atmospheric conditions. **[[** **]]** Section 6.4 of ANP-3824P discusses the CHF correlation used for the CRDA calculations. The LAR **[[** **]]** CHF correlation. The LAR states that the **[[** **]]** Therefore, the NRC staff finds use of the **[[** **]]** CHF correlation to be acceptable for use for this purpose.

<u>Limitation and Condition 22</u>

> As discussed in section 3.3.1.5 and section 4.0 of this SE, the SPCB and ACE [Framatome's advanced critical power correlation] CPR [critical power ratio] correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EM. Other CPR correlations (existing and new) that would be used with the AURORA-B EM must be reviewed and approved by the NRC or must be developed with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens

> Power Corporation's Critical Power Correlations to Co-Resident Fuel" (Reference 50).
> Furthermore, if transient thermal-hydraulic simulations are performed in the process of applying AREVA CPR correlations to co-resident fuel, these calculations should use the AURORA-B methodology. *(This is Condition 22 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

The NRC staff finds this limitation and condition is satisfied because the Framatome regulatory procedures require use of NRC approved methodologies. The applicability of the ACE/ATRIUM 11 correlation for use in the AURORA-B AOO methodology is described in NRC-approved TR ANP-10335P-A, Revision 0 (Reference 19).

<u>Limitation and Condition 23</u>

> Except when prohibited elsewhere, the AURORA-B EM may be used with new or revised fuel designs without prior NRC approval provided that the new or revised fuel designs are substantially similar to those fuel designs already approved for use in the AURORA-B EM (i.e., thermal energy is conducted through a cylindrical ceramic fuel pellet surrounded by metal cladding, flow in the fuel channels develops into a predominantly vertical annular flow regime, etc.). New fuel designs exhibiting a large deviation from these behaviors will require NRC review and approval prior to their implementation in AURORA-B. *(This is Condition 23 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

The NRC staff finds this limitation and condition is satisfied because as stated in ANP-3929P Framatome has no fuel designs that exhibit a large deviation from the behaviors described in this limitation and condition. ANP-3929P states that if a fuel design is developed that is significantly different, this fuel design will be submitted to the NRC for approval.

<u>Limitation and Condition 24</u>

> Changes may be made to the AURORA-B EM in the **[[**
>
> **]]** areas discussed in section 4.0 of this SE without prior NRC approval. *(This is Condition 24 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

The LAR stated that this condition is met through the use of the Framatome software development procedures. The NRC staff finds this limitation and condition is satisfied.

<u>Limitation and Condition 25</u>

> The parallelization of individual CCDs may be performed without prior NRC approval as discussed in section 4.0 of this SE. *(This is Condition 25 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

The LAR stated that no confirmation is required for this condition. The NRC staff finds this limitation and condition is satisfied since this is performed through the use of the Framatome software development procedures.

<u>Limitation and Condition 26</u>

> AREVA must continue to use existing regulatory processes for any code modifications made in the **[[**
>
> **]]**
> areas discussed in section 4.0 of this SE. *(This is Condition 26 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

Evaluation

The LAR stated that this condition is met through the use of the Framatome software development procedures, which includes 10 CFR 50.59 licensing considerations**.** The NRC staff finds this limitation and condition is satisfied.

<u>Limitation and Condition 27</u>

> The control rod model at each location in the core used for CRDA analyses with the AURORA-B EM shall use a control rod geometry and composition that is verified to bound the control rod worth for the physical control rod used in the location, for all axial elevations.

Evaluation

The LAR stated that **[[**

]] The NRC staff finds this limitation and condition is satisfied based on this conservatism.

<u>Limitation and Condition 28</u>

> Licensees utilizing AURORA-B to perform CRDA analyses using the methodology described in this TR shall confirm that the recommended maximum rod velocity of 3.11 ft/s is conservative for their control rods.

Evaluation

ANP-3929P stated that the rod velocity at MNGP remains bounded by the recommended maximum rod velocity of 3.11 ft/s. Based on this confirmation, the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 29

> If the check to verify that the total enthalpy is limiting at 10 percent core flow CZP conditions by **[[** **]]** fails, AREVA [Framatome] shall perform a more comprehensive evaluation to verify that they have identified the limiting initial conditions for that plant. This evaluation should consider a range of flow values and corresponding plant-specific minimum temperatures that is sufficiently broad to clearly identify the combination of initial conditions which maximizes the total enthalpy for the limiting rod.

Evaluation

ANP-3929P stated that **[[** **]]** for determining the total enthalpy with ATRIUM 11**.** Based on the ANP-3929P statement, the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 30

> When individual control rods are evaluated using the CRDA analysis methodology, if necessary, alternate distributions of inoperable rods should be utilized to ensure inclusion of at least one evaluation within each group of 4 quadrant symmetric control rods that maximizes the change in face- and/or diagonally-adjacent uncontrolled cells as a result of the candidate control rod withdrawal.

Evaluation

ANP-3929P stated that **[[** **]]** The LAR stated that **[[** **]]** The LAR stated that **[[** **]]** The NRC staff finds the approach used acceptable and this limitation and condition is satisfied.

Limitation and Condition 31

> The evaluation boundary curve used to determine candidate control rods for further evaluation based on their static rod worth must be verified to bound the

> following local characteristics of the fuel being evaluated: design pin peaking factors, fuel assembly design, location in or adjacent to the outermost ring of control rods, and average burnup for the 16 fuel assemblies surrounding the rod of interest.

Evaluation

Appendix A of ANP-3929P describes the process used to establish an evaluation boundary curve to simplify the calculations. The local characteristics of fuel used to establish the evaluation boundary with respect to design core conditions are presented for the ATRIUM 11 core. Based on the evaluation performed, the NRC staff finds the limitation and condition is satisfied.

<u>Limitation and Condition 32</u>

> If the highest worth rod at a given core statepoint results in a total enthalpy that is higher than the minimum high temperature failure threshold (i.e., lowest threshold for all rod internal pressures), additional rods must be considered for evaluation. This may be done by evaluating the next highest worth rods at the core statepoint of interest until the minimum high temperature failure threshold is met, or by using an approach analogous to the evaluation boundary curve for the PCMI failure threshold (as subject to condition 29).

Evaluation

ANP-3929P stated that the highest control rod worth did result in total enthalpy that led to exceeding of the minimum high temperature failure threshold. Section 4.1 of ANP-3929 addressed this condition. Based on the evaluation of additional control rods performed, the NRC staff finds the limitation and condition is satisfied.

<u>Limitation and Condition 33</u>

> If the methodology described in ANP-10333 is used to analyze the CRDA event with a fuel assembly design that has a different fuel rod geometry and/or manufacturing tolerances than the one used as a basis for the sensitivity study on gap width, the sensitivity study shall be repeated for the new fuel assembly design, using bounding values consistent with the uncertainty range for **[[**
>
> **]]** limiting increase in the peak total enthalpy, the total uncertainty shall be increased accordingly for total enthalpies calculated based on the new fuel assembly design.

Evaluation

ANP-3929P stated that gap sensitivity studies were performed with a bounding value for the uncertainty range of **[[** **]]** and the resulting peak total enthalpy **[[** **]]** Based on the results from the sensitivity studies the NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 34

> The uncertainty designated in the CRDA TR of **[[ ]]** for the enthalpy rises calculated using the CRDA analysis methodology may not be reduced without prior NRC approval.

Evaluation

ANP-3929P stated that an uncertainty of **[[ ]]** percent was used. Thus, the NRC staff find this limitation and condition is satisfied.

3.3.4 Conclusions – CRDA Analysis

The NRC staff reviewed the information in the LAR pertaining to the analysis of the CRDA event for MNGP. Based upon its review, the NRC staff finds that the proposal to implement the CRDA analysis methodology using the AURORA-B CRDA evaluation model is acceptable, satisfies all limitations and conditions, and is in compliance with the applicable regulatory requirements.

3.4 Transient Demonstration (ANP-3925P)

The evaluation of the transient analysis demonstration is discussed in ANP-3925P (Reference 6).

3.4.1 Regulatory Evaluation

In addition to the related GDCs 10 and 15, described in section 2.0 of this SE, the following regulatory requirement applies to the AOO/ATWS evaluation.

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

3.4.2 Technical Evaluation

Anticipated Operational Occurrences (AOOs)

The NRC staff evaluation of the AOOs consisted of the following items presented in ANP-3925P:

- Applicability of the AURORA-B AOO methodology.
- AURORA-B AOO methodology implementation to MNGP USAR Chapter 14 events.
- Demonstration of applicability of the AURORA-B methodology for selected AOOs.
- Disposition of the Limitations and Conditions in the Safety Evaluation for ANP-10300P-A, Revision 1.

Applicability of AURORA-B AOO Methodology

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

- S-RELAP5 - This code provides the transient T-H, thermal conduction, control systems, and special process capabilities (i.e., valves, jet pumps, steam separator, critical power correlations, etc.) necessary to simulate a BWR plant.
- MB2-K - This code provides the core neutronic response.
- RODEX4 – This code provides the thermal-mechanical response of the individual fuel rods. A subset of routines from this code are used to evaluate the transient thermal-mechanical fuel rod (including fuel/clad gap) properties as a function of temperature and rod internal pressure. The fuel rod properties are used by S-RELAP5 when solving the transient thermal conduction equations in lieu of standard S-RELAP5 material property tables.

The above three codes together as a system make up a multi-physics evaluation model to predict the relevant AOO characteristics. The methodology is based on **[[**
]] The uncertainty analysis in the AURORA-B AOO methodology, bound the 95 percent worst-case result with 95 percent confidence. Table 3.6 of the SE for the AURORA-B AOO methodology contains the uncertainty parameters used for the uncertainty analysis. **[[**

]]
Deterministic analysis using conservative initial and boundary conditions is performed for operational transients and TS requirements.

The NRC staff review is to ensure that the AURORA-B AOO methodology is applicable for the analysis of AOOs for MNGP. As described in section 3.1 (Applicability of Framatome BWR Methods to MNGP with ATRIUM 11 Fuel) of the SE for the AURORA-B AOO TR ANP-10300P-A, the methodology is applicable, in part, to BWR/3 through BWR/6 plants with conditions extending to EPU in EFW domain. Since MNGP is a BWR/4 plant, the methodology is applicable to MNGP in this respect. In addition, as discussed in section 4.3 of the SE for ANP-10332P-A, Revision 0 (Reference 18), and in limitation and condition number 27 of this SE, ATRIUM 11 is identified as one of the existing fuel designs that was considered in the review. Therefore, the NRC staff finds that the AURORA-B AOO methodology is applicable to the ATRIUM 11 fuel design.

AURORA-B AOO Methodology Implementation to MNGP USAR Chapter 14 Events

Table 1 below provides the disposition of all events and accidents described in USAR, Chapter 14 which were provided in ANP-3925P. The disposition consists of whether the event or accident will be analyzed at the initial reload of ATRIUM 11 fuel only, or at each reload cycle using the AURORA-B AOO methodology, or if no further analysis is required. ANP-3925P, Table 3.1, provided reasons for each disposition. Table 1 below shows the NRC staff evaluation of the dispositions of USAR, chapter 14, events provided in ANP-3925, Table 3.1.

Table 1- NRC Staff Evaluation of ANP-3925P Disposition of USAR, Chapter 14, Events

USAR Section	**Event/Analysis**	**LAR Disposition**	**NRC Staff Evaluation**
14.4.1	Generator Load Rejection Without Bypass	Address for initial reload.	The NRC staff finds the disposition acceptable because the turbine trip without bypass is similar to this transient as stated in USAR Section 14.4.5. The turbine trip without bypass transient will be addressed at each reload as mentioned below in the disposition of USAR Section 14.4.5 event.
14.4.2	Loss of Feedwater (FW) Heating (LFWH)	Address each reload.	The NRC staff finds the disposition acceptable because of the following: • ANP-3925P stated that the change in FW temperature takes place in less than 80 seconds after LFWH and therefore the generic analysis is not applicable. • This event is potentially limiting and therefore it is appropriate to address at each reload.
14.4.3	Rod Withdrawal Error – Low Power	No further analysis required.	The NRC staff finds the disposition acceptable because this event is bounded by the transient rod withdrawal-at power.
14.4.3	Rod Withdrawal Error – At Power	Address each reload.	The NRC staff finds the disposition acceptable because this is a potentially limiting event and it is conservative to analyze at each reload.
14.4.4	Feedwater Controller Failure – Maximum Demand	Address each reload.	The NRC staff finds the disposition acceptable because this is a potentially limiting event and it is appropriate to analyze at each reload.
14.4.5	Turbine Trip Without Bypass	Address each reload.	The NRC staff finds the disposition acceptable because this is a potentially limiting event and it is appropriate to analyze at each reload.

USAR Section	Event/Analysis	LAR Disposition	NRC Staff Evaluation
14.5.1	Vessel Pressure American Society of Mechanical Engineers (ASME) Code Compliance Model	Address each reload.	The NRC staff finds the disposition acceptable because it is appropriate to analyze at each reload.
14.5.2	Standby Liquid Control System Shutdown Margin	Address each reload.	The NRC staff finds the disposition acceptable because it is appropriate to analyze at each reload.
14.5.3	Stuck Rod Cold Shutdown Margin	Address each reload.	The NRC staff finds the disposition acceptable because it is appropriate to analyze at each reload.
14.6	Plant Stability Analysis	Address each reload.	The NRC staff finds the disposition acceptable because it is appropriate to analyze at each reload.
14.7.1	CRDA Evaluation	Address each reload.	The NRC staff finds the disposition acceptable it is appropriate to analyze at each reload.
14.7.2	LOCA	Address for initial reload.	The objective of the LOCA analysis is to demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46. The LOCA is analyzed in conjunction with the ECCS performance evaluation. The fuel parameters Peak Linear Heat Generation Rate (PLHGR) and Maximum Average Planar Linear Heat Generation Rates (MAPLHGRs) for the fuel are inputs to the ECCS performance evaluations. The NRC staff finds the disposition acceptable because LOCA analysis is used to determine the MAPLHGR limits which are independent of cycle specific assembly designs.
14.7.3	Main Steam Line Break Accident Analysis	No further analysis required.	The NRC finds the disposition acceptable because the consequences of this accident does not depend on the fuel design.

USAR Section	Event/Analysis	LAR Disposition	NRC Staff Evaluation
14.7.4	Fuel Loading Error Accident	Address each reload.	The NRC staff finds the disposition acceptable because it is appropriate to analyze at each reload.
14.7.5	One Recirculation Pump Seizure Accident Analysis	Address each reload.	The NRC staff finds the disposition is acceptable because the seizure of the recirculation pump in a single-loop operation is a potentially limiting event as compared to the non-limiting one recirculation pump seizure event in a two-loop operation.
14.7.6	Refueling Accident Analysis	Address for initial reload.	The NRC staff finds the disposition acceptable because the refueling accident is independent of the fuel design.
14.7.7	Accident Atmospheric Dispersion Coefficients	No further analysis required.	The NRC staff finds the disposition acceptable because this event is independent of the fuel design.
14.7.8	Core Source Term Inventory	Address each reload.	The NRC staff finds the disposition acceptable because the alternate source term (AST) analysis is required with the ATRIUM 11 fuel.
14.8	Anticipated Transients Without Scram (ATWS)	Address each reload.	The NRC staff finds the disposition acceptable because of the following: • As stated in ANP-3924P, **[[ ]].** • Peak cladding temperature and oxidation are bounded by the LOCA analysis results. • ATWS-I is addressed in ANP-3933P. The NRC staff evaluation is given in Section 3.3 of this SE.

USAR Section	Event/Analysis	LAR Disposition	NRC Staff Evaluation
14.10.1	Adequate Core Cooling for Transients with a Single Failure	No further analysis required.	The NRC staff finds the disposition acceptable because this event is independent of the fuel design.
14A Section 5.0	GE SIL [service information letter] 502 (Revision 1) Single Turbine Control Valve Slow Closure Event	No further analysis required	The NRC staff finds the disposition acceptable because the LAR stated that the application of the generic analysis to MNGP demonstrates this analysis is far from limiting. Previous licensee's analyses also demonstrated this analysis was non-limiting.
14A Section 5.0	Pneumatic System Degradation (Turbine Trip with Bypass and degraded scram speed)	Address for initial reload.	The licensee letter dated June 6, 2022, stated that this event will be addressed in the calculation plan mentioned in ANP-3925P, Section 3.2. The NRC staff finds the disposition acceptable because the calculation plan will identify the licensing campaign in which the analysis was performed and state that for the upcoming cycle, the analysis (a) is not needed if this event is determined to be non-limiting in its previous analysis, and (b) is needed if this event is determined to be limiting in its previous analysis.
14A Section 5.0	Loss of Stator Cooling	Address for initial reload.	The licensee letter dated June 6, 2022, stated that this event will be addressed in the calculation plan mentioned in ANP-3925P, Section 3.2. The NRC staff finds the disposition acceptable because the calculation plan will identify the licensing campaign in which the analysis was performed and state that for the upcoming cycle, the analysis (a) is not needed if this event is determined to be non-limiting in its previous analysis, and (b) is needed if this event is determined to be limiting in its previous analysis.

USAR Section	Event/Analysis	LAR Disposition	NRC Staff Evaluation
14A Table 5.1	Main Steam Isolation Valve (MSIV) Closure (One / All Valves)	No further analysis required.	The NRC staff finds the disposition acceptable because of the following reasons provided in the LAR: • Analysis results of this event are bounded by the turbine trip without bypass event (USAR Section 14.4.5). • Closure of all MSIVs with failure of the valve position scram function is addressed each reload to show compliance with the ASME vessel overpressure protection (USAR Section 14.5.1). • The MSIV closure event is addressed each reload as a potentially limiting ATWS overpressure event (USAR Section 14.8).
14A Table 5.1	Loss of Condenser Vacuum	No further analysis required.	The NRC staff finds the disposition acceptable because this event is bounded by the turbine trip without bypass (USAR Section 14.4.5).
14A Table 5.1	Pressure Regulator Failure – Full Close (Downscale)	Address each reload.	The NRC staff finds the disposition acceptable because with one pressure regulator out-of-service, it is a potentially limiting event and therefore it is appropriate to address at each reload.
14A Table 5.1	Loss of Auxiliary Power – All Grids	No further analysis required.	The NRC staff finds the disposition acceptable because this event is bounded by the turbine generator load rejection without bypass (USAR Section 14.4.1).
14A Table 5.1	Inadvertent High Pressure Coolant Injection	Address each reload.	The NRC staff finds the disposition acceptable because it is appropriate to address at each reload.

USAR Section	Event/Analysis	LAR Disposition	NRC Staff Evaluation
14A Table 5.1	Pressure Regulator Failure – Full Open	No further analysis required.	The NRC staff finds the disposition acceptable based on the following statements in the LAR: • Consequences of this event relative to thermal operating limits are non-limiting. • The TS 2.1.1.1 low dome pressure safety limit protects the lower boundary of the ACE/ATRIUM 11 critical power correlation. • Previous analyses conservatively showed that this safety limit is protected with the current MSIV steam line low pressure trip setpoint. • This event is also analyzed as an initiator for ATWS overpressure (USAR Section 14.8).
14A Table 5.1	Inadvertent Opening of Safety/Relief Valve	No further analysis required.	The NRC staff finds the disposition acceptable because the reactor depressurization during this event is less limiting (severe) than the Pressure Regulator Failure-Full Open event.

USAR Section	Event/Analysis	LAR Disposition	NRC Staff Evaluation
14A Table 5.1	Loss of Feedwater Flow	No further analysis required.	The NRC staff finds the disposition acceptable because the LAR stated the following: • Previous evaluations for a different fuel design showed that the lowest level following a loss of feedwater event remained well above the top of active fuel. • The long term water level transient is dependent upon the decay heat which is **[[ ]]** • This event does not pose any direct threat to the fuel in terms of a thermal power increase from the initial conditions. • The fuel will be protected provided the water level inside the shroud does not drop below the top of active fuel.
14A Table 5.1	Loss of Auxiliary Power Transformers	No further analysis required.	The NRC staff finds the disposition acceptable because this event is bounded by the Turbine Generator Load Rejection Without Bypass event (USAR Section 14.4.1).
14A Table 5.1	Recirculation Flow Control Failure – Decrease	No further analysis required.	The NRC staff finds the disposition acceptable because this event is bounded by a recirculation pump trip event.
14A Table 5.1	Trip of Two Recirculation Pumps	No further analysis required.	The NRC staff finds the disposition acceptable because this event is bounded by the Turbine Generator Load Rejection Without Bypass (USAR Section 14.4.1).

USAR Section	Event/Analysis	LAR Disposition	NRC Staff Evaluation
14A Table 5.1	Slow Recirculation Control Failure – Increase ($MCPR_f$ [minimum critical power ratio])	Address each reload.	The LAR stated that this event is used to determine the flow dependent MCPR operating limit and therefore NRC staff finds the disposition to address this event at each reload cycle acceptable.
14A Table 5.1	Slow Recirculation Control Failure – Increase ($LHGRFAC_F$, $MAPLHGR_F$)	Address each reload.	The LAR stated that this event is used to determine the flow dependent linear heat generation rate (LHGR) setdown factors and therefore the NRC staff finds the disposition to address this event at each reload cycle acceptable.
14A Table 5.1	Fast Recirculation Control Failure – Increase	Address each reload.	The NRC staff finds the disposition acceptable because it is a potentially limiting event and is conservative to analyze at each reload cycle.
14A Table 5.1	Startup of an Idle Recirculation Loop	No further analysis required.	The NRC staff finds the disposition acceptable because it is bounded by the Inadvertent high pressure coolant injection (HPCI) Startup event.

<u>Demonstration of Applicability of the AURORA-B Methodology for Selected AOOs.</u>

ANP-3925P provided a demonstration analysis using the AURORA-B AOO methodology. Since the analysis is a demonstration analysis, the NRC staff's review is to ensure that AOOs with the AURORA-B AOO methodology applied for the ATRIUM 11 fuel can be adequately evaluated.

The following transient events were analyzed:

- Load Rejection No Bypass (LRNB)
- Turbine Trip No Bypass (TTNB)
- Feedwater Controller Failure (FWCF)
- Inadvertent HPCI Startup 14A Table 5.1
- Loss of Feedwater Heating (LFWH)
- Fast Flow Runup
- ASME Overpressurization Analysis
- ATWS Overpressurization Analysis.

Load Rejection No Bypass (LRNB)

The LRNB event is described in Section 14.4.1 of the USAR. For this event the following conditions within the EFW power/flow map at the EOFP (end of full power) cycle exposure were analyzed using nominal scram speed (NSS) insertion times:

- 100 percent (%) core power, with 105 percent and 80 percent core flow
- 85 percent core power, with 105 percent core flow
- 60 percent core power, with 108.3 percent core flow
- 40 percent core power above Pbypass, with 111.1 percent core flow
- 40 percent core power below Pbypass, with 111.1 percent core flow

ANP-3925P, Table 4.2, shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3925P, Table 4.1, shows the ΔMCPR (maximum critical power ratio) (change in MCPR) at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (% of rated) = (40/111.1 below Pbypass **[[ ]]**).

ANP-3925P, Figures 4.1, 4.2, and 4.3, show the responses of reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

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

Turbine Trip No Bypass (TTNB)

The TTNB event is described in Section 14.4.5 of the USAR. The event was analyzed for the same power/flow conditions as for the LRNB event within the EFW power/flow map at the EOFP cycle exposure, using NSS insertion times.

ANP-3925P, Table 4.3, shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3925P, Table 4.1, shows the ΔMCPR at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (% of rated) = (40/111.1 below Pbypass).

ANP-3925P, Figures 4.4, 4.5, and 4.6, show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

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

Feedwater Controller Failure (FWCF)

The FWCF event is described in Section 14.4.4 of the USAR. The event was analyzed for the same power/flow conditions as for the LRNB event within the EFW power/flow map at the EOFP cycle exposure, using NSS insertion times.

ANP-3925P, Table 4.4, shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3925P, Table 4.1, shows the ΔMCPR at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (% [percent] of rated) = 40/111.1 and same at (40/111.1 below Pbypass).

ANP-3925P, Figures 4.7, 4.8, and 4.9, show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

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

Inadvertent High-pressure Coolant Injection (HPCI) Startup

The inadvertent HPCI startup event is described in Section 14A of the USAR. This event was analyzed for the following conditions within the EFW power/flow map at the EOFP cycle exposure, using NSS insertion times:

- 100 percent (%) core power, with 105 percent and 80 percent core flow
- 85 percent core power, with 105 percent core flow
- 60 percent core power, with 108.3 percent core flow
- 40 percent core power, with 111.1 percent core flow

ANP-3925P, Table 4.5, shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3925P, Table 4.1, shows the ΔMCPR for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (% of rated) = 100/105). The reactor water level in the inadvertent HPCI startup analysis case at rated power reached the high-level trip setpoint resulting in larger ΔMCPR values than the off-rated cases which did not reach this setpoint. The licensee will investigate the impact of plant parameters on this trend as part of the initial fuel transition at each reload analysis as stated in Table 1 above.

ANP-3925P, Figures 4.10, 4.11, and 4.12, show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

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

Loss of Feedwater Heating (LFWH)

The LFWH event is described in Section 14.4.2 of the USAR. This event was analyzed for the same power/flow conditions as for the Inadvertent HPCI startup event within the EFW power/flow map at the EOFP cycle exposure, using NSS insertion times.

ANP-3925P, Table 4.6, shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3925P, Table 4.1, shows the ΔMCPR at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (% of rated) = 60/108.3 and 40/111.1.

ANP-3925P, Figures 4.13, 4.14, and 4.15, show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and 105 percent of rated core flow.

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

Fast Flow Runup

This event was analyzed for the following conditions within the EFW power/flow map at the EOFP cycle exposure, using NSS insertion times:

- 100 percent (%) core power, with 80 percent core flow
- 85 percent core power, with 60.6 percent core flow
- 60 percent core power, with 60 percent core flow
- 40 percent core power, with 70 percent core flow

ANP-3925P, Table 4.7, shows the sequence of event timing at 100 percent power with 105 percent core flow.

ANP-3925P, Table 4.1, shows the ΔMCPR at the power/core-flow state points analyzed for this event. The maximum ΔMCPR is **[[ ]]** at the power/core-flow (% of rated) = 40/70.

ANP-3925P, Figures 4.16, 4.17, and 4.18, show the responses of various reactor and plant parameters during this event initiated at 100 percent of rated power and percent of rated core flow.

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

ASME Overpressurization Analysis

The ASME overpressurization event is described in Section 14.5.1 of the USAR. The event can be initiated by various steam line or reactor system malfunctions and by various operator actions. Closure of all main steam isolation valves (MSIVs) while at power can result in a significant overpressure transient in the reactor vessel. Normally, as the MSIVs close, a reactor scram is initiated by position switches which sense closure. This event was analyzed at 102 percent power with 105 percent flow and 102 percent power with 80 percent flow at the latest full-power exposure in the cycle design.

The event causes a rapid pressurization of the core resulting in a decrease in void fraction which in turn causes a rapid increase in power.

To demonstrate the applicability of the ANP-10300P-A, AURORA-B AOO methodology for ASME overpressurization event, the MSIV, TSV, and TCV closure were analyzed at 102 percent power with 105 percent flow and 102 percent power with 80 percent flow at the latest full-power exposure in the cycle design. ANP-3925P stated the following was assumed for the analysis:

- The most critical active component (direct scram on valve closure) was assumed to fail. However, scram on high neutron flux and high dome pressure is available.
- Opening of the turbine bypass valves was not credited.
- 3 safety/relief valves (SRVs) out-of-service (SRV/OOS).
- Opening the SRV at the relief setpoints was not credited (open at safety setpoint)
- SRV open at 1145 psig (pounds per square inch gauge) (approximately 3 percent drift over the TS opening setpoint).
- NSS insertion times were used.
- The initial dome pressure was set at the maximum allowed 1040.0 psia (pounds per square inch differential).
- A fast MSIV closure time of 3.0 seconds was used.
- High-pressure recirculation pump trip (ATWS-RPT) was considered.

The acceptance criteria for this transient are based on compliance with GDC 15. This is to demonstrate compliance with the ASME Code by showing that the pressure in the reactor coolant and main steam systems remain below 110 percent of the design values.

ANP-3925P, Table 4.8, presents the results of the MSIV and TSV closure event overpressurization analysis results and Table 4.9 presents the sequence of event timing for the TSV closure event at 102 percent power with 105 percent core flow. The results show the TSV closure event has lesser margin than the MSIV closure event. For both events the acceptance criteria of maximum vessel lower plenum pressure limit of 1375 psig and dome pressure limit of 1332 psig are not exceeded.

ANP-3925P, Figures 4.19, 4.20, 4.21, and 4.22, show the response of various reactor plant parameters during the limiting TSV closure event.

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

ATWS Overpressurization Analysis

The ATWS event is described in Section 14.8 of the USAR. This event was analyzed using the AURORA-B AOO methodology for MSIV closure and pressure regulator failure open (PRFO) transients at 100 percent power with 80 percent flow and 100 percent power with 105 percent flow at the beginning of cycle (BOC) exposure. The PRFO transient causes the TCV and turbine bypass valve to open such that steam flow increases until the maximum combined steam flow limit is attained.

The system pressure decreases until the low main steam line pressure setpoint is reached, resulting in the closure of the MSIVs. The resulting pressurization wave causes a decrease in core voids and an increase in core pressure thereby increasing the core power. The following was assumed for the analysis:

- The analytical limit ATWS-RPT setpoint and function.
- 1 SRV/OOS and the remaining 7 SRV open at safety setpoint.
- SRV open at 1145 psig (approximately 3 percent drift over the TS opening setpoint).
- All scram functions were disabled.
- Nominal values were used for initial dome pressure and feedwater temperature.
- The MSIV closure is based on a nominal closure time of 4.0 seconds for both events.

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

ANP-3925P, Table 4.8, presents the results of the limiting (least margin) MSIV closure overpressurization analysis and Table 4.10 presents the sequence of event timing at 100 percent power with 80 percent core flow.

ANP-3925P, Figures 4.23, 4.24, 4.25, and 4.26, show the response of various reactor plant parameters during the ATWS MSIV closure event which results in the maximum reactor vessel pressure.

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

3.4.3 Evaluation of ANP-10300P Limitations and Conditions for Transient Demonstration

The AURORA-B AOO methodology has 26 limitations and conditions listed in section 5.0 of the NRC staff's SE for ANP-10300P-A, Revision 1. The disposition of these limitations and condition is provided in ANP-3925P, appendix A. The NRC staff evaluation of the disposition is given below.

Limitation and Condition 1

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

Evaluation

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

Limitation and Condition 2

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

Evaluation

[[

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

Limitation and Condition 3

> Parameter uncertainty distributions and their characterizing upper and lower 2σ levels are presented in Table 3.6 and discussed in section 3.6 of this SE. The distribution types will not be changed and the characterizing upper and lower 2σ uncertainties will not be reduced without prior NRC approval. In the cases of the parameters **[[** **]]**, the respective methodologies discussed in section 3.6.4.10 and section 3.6.4.17 shall be used when determining the associated upper and lower 2σ levels. The **[[** **]]** is subject to Limitation and Condition No. 0, below.

Evaluation

The NRC staff confirmed that the generic uncertainty distributions presented in Table 2.2 of ANP-3925P are consistent with those in Table 3.6 of the SE for the AURORA-B methodology. For the **[[** **]]**, ANP-3925P stated that the range was developed based on the approved process in section 3.6.4.10 of the methodology. Therefore, the NRC staff finds that this limitation and condition is adequately addressed.

Limitation and Condition 4

> As discussed in section 3.3.1.2, before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are modeled in licensing analyses using AURORA-B, AREVA must justify that the AURORA-B EM can acceptably predict void fraction results for the new fuel designs within the **[[ ]]** prediction uncertainty bands. Otherwise, the prediction uncertainty bands should be appropriately expanded, and the **[[ ]]** should be appropriately updated utilizing the methodology discussed in Section 0 of this SE.

Note: Section 0 does not exist in the AURORA-B AOO methodology SE. This is a typographical error and should have been noted as section 3.6.4.15.

Evaluation

The NRC staff reviewed the discussion on void-fraction prediction for ATRIUM 11 fuel in Section 5.1 of ANP-3924P and finds the 2-sigma error of **[[ ]]** acceptable because it bounds the 2-sigma void-fraction error **[[ ]]**

Limitation and Condition 5

> As discussed in Section 3.3.2.4.4 [of ANP-10300P-A, Revision 1], before new fuel designs (i.e., designs other than ATRIUM-10 and ATRIUM-10XM) are included in licensing analyses performed using the AURORA-B EF, AREVA must justify that the **[[ ]]** void-quality correlation within MICROBURN-B2 is valid for the new fuel designs at EPU and EFW conditions.

Evaluation:

The NRC staff reviewed the justification on the use of **[[ ]]** void-quality correlation in Section 5.1 of ANP-3924P and finds it acceptable for ATRIUM 11 fuel because it is acceptable for including ATRIUM 10 and ATRIUM 10XM which are closer to ATRIUM 11 fuel at EPU and EFW conditions. Therefore, the NRC staff finds that this limitation and condition is adequately addressed.

Limitation and Condition 6

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

Evaluation

ANP-3925P stated that **[[ ]]** Based on this statement from ANP-3925P, the NRC staff finds this limitation and condition is satisfied.

<u>Limitation and Condition 7</u>

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

Evaluation

Section 5.3 of ANP-3925P stated that:

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

Based on the above statement, the NRC staff finds this limitation and condition is satisfied for the first ATRIUM 11 fuel application cycle because the plant parameters that have a range of values during normal operation which includes initial conditions such as power, flow, pressure, and inlet subcooling will be reviewed. ANP-3925 states that sensitivity studies for all of these key parameters/conditions for all FoMs to **[[ ]]** will be performed.

<u>Limitation and Condition 8</u>

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

Evaluation

ANP-3925P, Table 2.2, shows the sampling ranges for the uncertainty distributions used in the analysis. The licensee stated that **[[**

]] Based on the ANP-3925P statement, the NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 9

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

Evaluation

ANP-3925P stated that the analyses comply with the information of Tables 3.2 and 3.4 of the SE for AURORA-B AOO methodology as they relate to this limitation. The NRC staff confirmed that the phenomena was modeled as described in Tables 3.2 and 3.4 of the SE of the AURORA-B methodology and, therefore, this limitation and condition has been adequately addressed.

Limitation and Condition 10

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

Evaluation

ANP-3925P stated that the **[[** **]]** as they relate to this limitation. The NRC staff confirmed that the phenomena was modeled as described in Tables 3.2 and 3.4 of SE of the AURORA-B methodology and, therefore, this limitation and condition has been adequately addressed.

Limitation and Condition 11

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

Evaluation

ANP-3925P, Section 5.3, provided the following related to the uncertainties of the highly ranked plant-specific PIRT parameters. ANP-3925P stated that **[[**

]]

For the parameter C12, the ANP-3925P stated:

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

For the parameter R01, the ANP-3925P stated:

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

For the parameter R02, ANP-3925P stated:

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

For the parameter SL02, ANP-3925P stated:

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

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

Limitation and Condition 12

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

Evaluation

The conservative method described in ANP-3924P, Revision 0, Section 6.3.1, for transient mixing determination in the **[[** **]]** for the analysis of transients **[[** **]]** will be followed. ANP-3924P states that the results and conclusions of this analysis will be provided as a part of the initial cycle reload safety analysis report (RSAR) of ATRIUM 11 fuel. Based on this statement, the NRC finds this limitation and condition is satisfied.

Limitation and Condition 13

> The AURORA-B uncertainty methodology discussed in section 3.6 of this SE may be used in licensing applications for the events listed in section 3.1 of this SE, with the exception of three specific events identified in section 3.6.2 of this SE: **[[** **]]** These events are generally expected to be benign and hence non-limiting. While the NRC staff's review concluded that the AURORA-B EM contains code models and correlations sufficient for simulating these events, the uncertainty methodology developed in the TR did not address certain important phenomena or conditions associated therewith. Therefore, while licensing applications may rely on nominal calculations with the AURORA-B EM for these events in the course of demonstrating that all regulatory limits are satisfied with significant margin, the existing uncertainty methodology may not be applied directly to these specific events.

Evaluation

As stated in ANP-3925P, Table 3.1, for the event in USAR Section 14.A, Table 5.1, "Inadvertent High Pressure Coolant Injection" the disposition status is to be address at each reload. Based on the regulatory audit (Reference 43) findings, the NRC staff notes that the existing uncertainty methodology does apply to HPCI and not to high-pressure core spray (HPCS) because the statistical methodology did not address uncertainty in the void prediction in the upper plenum. The HPCS injects via core spray above the core in the voided region where the uncertainty can be important. MNGP has a HPCI system; therefore, the NRC staff finds the limitation and condition is satisfied.

Limitation and Condition 14

> The scope of the NRC staff's approval for AURORA-B does not include the ABWR [advanced boiling water reactor] design.

Evaluation

This limitation and condition is not applicable because MNGP is not an ABWR design.

Limitation and Condition 15

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

Evaluation

This limitation and condition is not applicable since MNGP is not a BWR/2 design.

Limitation and Condition 16

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

Evaluation

ANP-3925P stated that the plant parameter document contains the **[[** **]].** However, for the transient analysis, the AURORA-B model **[[** **]]** The NRC staff therefore finds that this limitation and condition is satisfied.

Limitation and Condition 17

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

Evaluation

ANP-3925P stated that **[[** **]]** for MNGP. Therefore, the NRC staff finds this limitation and condition is satisfied.

<u>Limitation and Condition 18</u>

As discussed in section 3.6.5 of this SE regarding conservative measures:

a. Plant-specific licensing applications shall describe and provide justification for the method for determining and applying conservative measures in future deterministic analyses for each FoM (e.g., biasing calculational inputs, post-processing adjustments to calculated nominal results), and

b. If the 95/95 FoMs for a given parameter calculated according to the defined conservative measures during a deterministic analysis shows a difference in magnitude exceeding 1σ from the corresponding value calculated in the most recent baseline full statistical analysis, AREVA must re-perform the full statistical analysis for the affected scenario and determine new conservative measures.

Evaluation

a. The FoMs considered are **[[**

]].

For the **[[** **]],** ANP-3925P stated that **[[** **]]** The NRC staff therefore finds it acceptable that **[[** **]]**

For the LHGRFACp evaluations, ANP-3925P stated that **[[** **]** The NRC staff reviewed the description **[[** **]]** and finds it acceptable because the **[[** **]]**

b. The LAR stated that **[[** **]]** Based on this statement, the NRC staff finds that this limitation and condition is satisfied.

<u>Limitation and Condition 19</u>

As discussed in section 3.6.5 of this SE, the following stipulations are necessary to ensure that the FoMs calculated by AREVA in accordance with ANP-10300P would satisfy the 95/95 criterion:

a. AREVA will use multivariate order statistics when multiple FoMs are drawn from a single set of statistical calculations,

b. AREVA will choose the sample size prior to initiating statistical calculations,

c. AREVA will not arbitrarily discard undesirable statistical results, and

> d. AREVA will maintain an auditable record to demonstrate that its process for performing statistical licensing calculations has been executed in an unbiased manner.

Evaluation

To satisfy this limitation and condition ANP-3925P made the following statements:

a. Framatome calculations will utilize the multivariate order statistics when a single transient is used to determine multiple FoMs.

b. Framatome will choose the sample size prior to initiating statistical calculations.

c. Framatome will not arbitrarily discard undesirable statistical results.

d. Framatome will maintain an auditable record to demonstrate the process for performing statistical licensing calculations is being executed in an unbiased manner.

Based on these statements, the NRC staff finds this limitation and condition is satisfied.

<u>Limitation and Condition 20</u>

> The implementation of any new methodology within the AURORA-B EM (i.e., replacement of an existing CCD) is not acceptable unless the AURORA-B EM with the new methodology incorporated into it has received NRC review and approval. An existing NRC-approved methodology cannot be implemented within the AURORA-B EM without NRC review of the updated EM.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the methodology used the AURORA-B EM as described in ANP-10300P-A, Revision 1, and CCDs as described in ANP-10300P-A, Revision 1, are not replaced.

<u>Limitation and Condition 21</u>

> NRC-approved changes that revise or extend the capabilities of the individual CCDs comprising the AURORA-B EM may not be incorporated into the EM without prior NRC approval.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the Framatome regulatory procedures require use of NRC approved methodologies within the applicability defined for that methodology.

Limitation and Condition 22

> As discussed in section 3.3.1.5 and section 4.0 of this SE, the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EF. Other CPR correlations (existing and new) that would be used with the AURORA-B EM must be reviewed and approved by the NRC or must be developed with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel" Reference 50). Furthermore, if transient thermal-hydraulic simulations are performed in the process of applying AREVA CPR correlations to co-resident fuel, these calculations should use the AURORA-B methodology.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the Framatome regulatory procedures require use of NRC approved methodologies. The applicability of the ACE/ATRIUM 11 correlation for use in the AURORA-B AOO methodology is described in NRC-approved TR ANP-10335P-A, Revision 0.

Limitation and Condition 23

> Except when prohibited elsewhere, the AURORA-B EM may be used with new or revised fuel designs without prior NRC approval provided that the new or revised fuel designs are substantially similar to those fuel designs already approved for use in the AURORA-B EM (i.e., thermal energy is conducted through a cylindrical ceramic fuel pellet surrounded by metal cladding, flow in the fuel channels develops into a predominantly vertical annular flow regime, etc.). New fuel designs exhibiting a large deviation from these behaviors will require NRC review and approval prior to their implementation in AURORA-B.

Evaluation

The NRC staff finds this limitation and condition is satisfied because as stated in ANP-3925P Framatome has no fuel designs that exhibit a large deviation from the behaviors described in this limitation and condition. ANP-3925P stated that if a fuel design is developed that is significantly different, this fuel design will be submitted to the NRC for approval.

Limitation and Condition 24

> Changes may be made to the AURORA-B EM in the **[[**
>
> **]]** areas discussed in section 4.0 of this SE without prior NRC approval.

Evaluation

ANP-3925P stated that **[[**

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

Limitation and Condition 25

The parallelization of individual CCDs may be performed without prior NRC approval as discussed in section 4.0 of this SE.

Evaluation

ANP-3925P stated that **[[** **]]** The NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 26

AREVA must continue to use existing regulatory processes for any code modifications made in the **[[** **]]** areas discussed in section 4.0 of this SE.

Evaluation

ANP-3925P stated that **[[** **]]** The LAR stated that **[[** **]]** The NRC staff finds this limitation and condition is satisfied.

The NRC staff reviewed all limitation and conditions and finds that each was adequately addressed by the licensee for the demonstration case and will be supported by the RSAR when it will be submitted.

3.5.4 Conclusions for ATRIUM 11 Transient Demonstration

The following is a summary of NRC staff technical conclusions for the ATRIUM 11 transient demonstration:

- ANP-3925P appropriately justified the use of AURORA-B AOO methodology for analyzing transient events for MNGP.
- For the USAR chapter 14 events, the licensee identified (a) events that should be analyzed at each reload, (b) events that should be analyzed at the first reload only, and (c) events for which no further analysis is necessary. The NRC staff finds the ANP-3925P dispositions acceptable.
- The NRC staff reviewed ATRIUM 11 transient demonstration analysis and finds that the licensee analyzed the potentially limiting events and their results are realistic and meet the specified acceptance criteria.
- All limitations and conditions for using the AURORA-B AOO methodology documented in TR ANP-10300P-A, Revision 1 are satisfied.

- Compliance with the applicable regulatory requirements 10 CFR 50.62, GDCs 10, 15, and 20 has been demonstrated.

3.5 Loss-of-Coolant Analysis for ATRIUM 11 Fuel (ANP-3934P)

The evaluation of the LOCA analysis is discussed in ANP-3934P (Reference 7).

3.5.1 Regulatory Evaluation

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

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

- Regulation 10 CFR 50.46
- Regulation 10 CFR Part 50; Appendix A, GDC 35
- Regulation 10 CFR Part 50, Appendix K

Regulation 10 CFR 50.46

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

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

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

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

Regulation 10 CFR 50.46(b) Acceptance Criteria:

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

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

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

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

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

ANP-3934P termed the acceptance criteria (1) through (5) as the PCT criterion, the maximum local oxidation (MLO) criterion, the hydrogen generation or core average metal water reaction (CMWR) criterion, the coolable geometry criterion, and the long-term cooling criterion, respectively. A maximum average planar linear heat-generation ratio (MAPLHGR) limit is established for each fuel type to ensure these criteria are met.

In accordance with Limitation and Condition 4 from the NRC staff's final SE included in ANP-10332P-A, the AURORA-B LOCA evaluation model may not be referenced as a basis for demonstrating adequate long-term core cooling in satisfaction of 10 CFR 50.46(b)(5). To demonstrate continued adherence to this requirement, the licensee cited existing licensing basis analysis performed on a generic basis by the nuclear reactor vendor (i.e., General Electric), which is documented in approved TR NEDO-20566A (Reference 26). 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.

Regulation 10 CFR Part 50, Appendix A, GDC 35

Criterion 35 – "Emergency core cooling" states:

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

Regulation 10 CFR Part 50, Appendix K

Appendix K to 10 CFR, part 50, consists of two parts:

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

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

The second part specifies requirements for the documentation of LOCA evaluation models, including a complete description, a code listing, sensitivity studies, and comparisons against experimental data.
The NRC staff's basis for concluding that the AURORA-B LOCA evaluation model used to perform the LOCA analysis for MNGP conforms to the requirements of Appendix K to 10 CFR, Part 50, is discussed in section 6.2.1 of the NRC staff's SE on ANP-10332P-A (Reference 20).

3.5.2 Technical Evaluation

LOCA Analysis

The LOCA event response is divided into 3 phases: the blowdown phase, the refill phase, and the reflood phase. ANP-3934P, Section 3.1, described these phases. To support the planned transition to ATRIUM 11 fuel the spectrum of postulated LOCA events were analyzed to verify the satisfaction of applicable regulatory requirements following the transition to ATRIUM 11 fuel. The analyses used the AURORA-B LOCA EM to demonstrate compliance with the acceptance criteria in 10 CFR 50.46(b)(1) through (b)(4) that apply to the short-term LOCA analysis.

Methodology

The AURORA-B LOCA methodology described in NRC-approved LTR ANP-10332P-A, is an Appendix K, "ECCS Evaluation Models," to 10 CFR, Part 50, analysis methodology. The methodology is based on S-RELAP5 code that incorporates transient fuel rod thermal-mechanical subroutines from the RODEX4 code documented in BAW-10247P-A (References 21, 22, and 23). The fuel parameters are specified using RODEX4 code, which is used to determine the **[[**

]].

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

]].

As documented in the SER on ANP-10332P-A, the AURORA-B LOCA EM is acceptable for application to LOCA analysis for BWR/3-BWR/6 plants, and therefore is applicable because MNGP is a BWR/4 plant.

Break Spectrum Analysis

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

As described in ANP-3934P, break spectrum analysis was performed over a range of break locations, break sizes, break types (double-ended guillotine (DEG) and split), initial state points, axial power shapes (top-peaked, mid-peaked), and assumed single-failures to determine the

break that yields the highest PCT [[
]].

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

- The analysis is based on a full core of ATRIUM 11 fuel.
- Reactor thermal power is assumed to be 102 percent of the rated thermal power to address the maximum measurement uncertainty.
- [[ ]].
- Reactor core is modeled with heat generation rates as required by Appendix K of 10 CFR 50.
- [[ ]] were assumed to be at the MAPLHGR limit shown in Figure 2.1 of ANP-3934P.
- [[
]] of ANP-3934P.
- With operation in the EFW domain shown in ANP-3934P, Figure 1.1, [[
]]
- With operation in the EFW domain shown in ANP-3934P, Figure 1.1, [[
]].
- The analysis used S-RELAP5 code which incorporates the clad swelling and rupture models from NUREG-0630 (Reference 25) to calculate the thermal-hydraulic response during all phases of LOCA.
- The following tables in ANP-3934P provide inputs used in the break spectrum analysis:
    - Table 4.1 shows reactor initial conditions

- Table 4.2 shows reactor system parameters
- Table 4.3 shows ATRIUM 11 fuel assembly parameters
- Table 4.4 shows HPCI system parameters
- Table 4.5 shows low pressure coolant injection (LPCI) system parameters
- Table 4.6 shows low pressure core spray (LPCS) parameters
- Table 4.7 shows automatic depressurization system (ADS) parameters
- Table 4.8 shows RDIV parameters

- ANP-3934P, Figure 4.1, shows the reactor vessel nodalization, Figure 4.2 shows the core nodalization used in the analysis which are consistent with those in NRC approved LTR ANP-10332-P-A, and Figure 4.3 shows the ECCS schematic.

- **[[ ]].**

- ECCS initiation is assumed to occur when the water level drops to the applicable water level setpoint (ANP-3934P, Tables 4.4 and 4.5) and conservatively the analysis does not credit ECCS flow until the ECCS injection valves open and the ECCS pumps reach rated speed.

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

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

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

For evaluating the ECCS performance during LOCAs, consistent with 10 CFR 50.46 and 10 CFR, part 50, Appendix K, a spectrum of possible pipe breaks up to and including the instantaneous DEG break and longitudinal splits in the recirculation system pipes with the split area equal to the cross-sectional area of the pipe were analyzed. The break types and sizes along with consideration of single failures are analyzed for both suction and discharge recirculation pipe breaks. The largest diameter recirculation pipes are the suction line between the reactor vessel and the recirculation pump and the discharge line between the recirculation pump and the riser manifold ring. The LOCA analyses are performed for breaks in both locations for DEG and split breaks. The break areas analyzed range between full pipe area and **[[ ]]** ft^2 with discharge coefficients from 1.0 to 0.4. ANP-3934P stated that the range of discharge coefficients is used to cover uncertainty in the actual geometry at the break. **[[ ]].** The most limiting DEG break was determined by varying the discharge coefficient.

ANP-3934P, Table 5.1, identifies the following single failures (SFs) considered for LOCA analysis:

(a) Failure of one train of direct current (DC) power or battery (BATT) (i.e., SF-BATT)).
(b) Failure of a LPCI system injection valve (i.e., SF-LPCI).
(c) Failure of a diesel generator (DGEN) (i.e., SF-DGEN).

(d) Failure of a HPCI system (i.e., SF-HPCI).
(e) Failure of ADS valve (i.e., SF-ADS).

The analysis determined that the ECCS resources for both SF-DGEN and SF-HPCI cases will be equal to or greater than the ECCS resources for SF-BATT such that the analyses for SF-DGEN and SF-HPCI cases is not needed because they would be bounded by the analysis assuming SF-BATT. The NRC staff's review found that this determination was appropriate, and that the analysis considered all postulated SFs described in the MNGP USAR.

To determine an exposure-dependent MAPLHGR limit, and **[[**

]] was analyzed from beginning-of-life to end-of-life **[[** **]]** increments. To confirm the acceptability of the LOCA analysis with respect to 10 CFR 50.46 criteria, the licensee calculated PCT, MLO, and CMWR over the range of exposures. **[[** **]]** were also analyzed. **[[**

]] The cases in ANP-3934P, **[[** **]]** using the S-RELAP5 model. The analysis is performed at **[[** **]].** The MAPLHGR input is consistent with the data in ANP-3934P, Figure 2.1. **[[** **]]**. Exposure-dependent fuel rod data is provided from RODEX4 results. The impact of TCD is addressed using RODEX4.

The NRC staff finds that the exposure-dependent analysis generally conforms to the approved evaluation model documented in ANP-10332P-A. In particular, the exposure study analyzed **[[** **]]** accounting for exposure-dependent limiting values of the LHGR and MAPLHGR. The exposure study deviated from the methodology approved in the NRC staff's SER on ANP-10332P-A in that at **[[**

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

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

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

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

- SFs are appropriately assumed in each analysis.

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

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

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

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

[[

]] ANP-3934P, Section 7.2, stated that the SLO analyses are performed using a 0.8 multiplier to the two-loop MAPLHGR limit resulting in a maximum SLO MAPLHGR limit of **[[ ]]** kW/ft. SLO breaks in the EFW region were not analyzed because plant operation is not permitted in this domain.

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

- The analysis used an NRC-approved S-RELAP5 methodology.

- In a letter dated June 6, 2022, the licensee justified the SLO MAPLHGR multiplier by stating that the reduction factor is defined so that the SLO PCT is bounded by the TLO PCT.

<u>Main Steam Line Breaks</u>

ANP-3934P, Section 5.3.1 described the large main steam line break inside the containment in and stated that **[[**

]]

The NRC staff finds the conclusion that **[[**
]] acceptable because during the blowdown period from the break, a higher secondary flow will lead to a high heat transfer from

the primary to the secondary side. In this condition, a rapid initiation of the low-pressure ECCS will lead to a PCT significantly less than the DEG break of the recirculation line.

Feedwater Line Breaks

ANP-3934P, Section 5.3.2, described the feedwater line break inside the containment and stated that **[[**

]]

The NRC staff finds the conclusion **[[**

]]

HPCI Line Breaks

ANP-3934P, Section 5.3.3, described the HPCI line break inside the containment and stated that this line is connected to the feedwater line outside containment and **[[**

]]. The HPCI steam supply line is connected to the main steam line inside containment and **[[**

]]

The NRC staff finds the conclusion that **[[**

]]

LPCS Line Breaks

ANP-3934P, Section 5.3.4, described the LPCS line break inside the containment. The break is assumed to be just outside the reactor vessel. The **[[**

]]

Based on the **[[**

]]

LPCI Line Breaks

ANP-3934P, Section 5.3.5, described the LPCI line break inside the containment and stated that the LPCI injection lines are connected to the larger recirculation discharge lines. **[[**

]]

The NRC staff finds the conclusion that the LPCI line break would be nonlimiting relative to the acceptance criteria to be acceptable because it is **[[**
]] for which the PCT is bounded by the recirculation pump suction line break PCT.

Reactor Core Isolation Cooling (RCIC) Line Breaks

ANP-3934P, Section 5.3.6, described the RCIC line break inside the containment and stated that since the RCIC discharges to the feedwater line **[[**

]]

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

]

Reactor Water Cleanup (RWCU) Line Breaks

ANP-3934P, Section 5.3.7, described the RWCU line break inside the containment and stated that the extraction line is connected to a recirculation suction line with an additional connection to the vessel bottom head, **[[**

]

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

]

Shutdown Cooling Line Breaks

ANP-3934P, Section 5.3.8, described the shutdown cooling line break inside the containment and stated that because the line is connected to a recirculation discharge line, **[[**
]]

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

Instrument Line Breaks

ANP-3934P, Section 5.3.9, described the instrument line break inside the containment and stated that **[[**

]]

The NRC staff finds the conclusion that the steam-filled or liquid-filled instrument line breaks PCT would be nonlimiting relative to the acceptance criteria acceptable because **[[**

]]

Results

ANP-3934P, Table 7.1, summarizes the limiting case PCT results for SLO and TLO. The PCTs for SLO and TLO are **[[ ]]**ºF and **[[ ]]**ºF respectively both based on pump discharge break area of 1.0 DEG, SF-LPCI, and top-peaked axial power shape.

ANP-3934P, Table 9.1, shows the **[[**

]

ANP-3934P, Table 6.1, describes the following parameters for the TLO LOCA analysis case that produced the **[[ ]]** result:

- Break Size and Type 1.0 DEG
- Break Location Pump Suction
- Single Failure SF-LPCI
- Power 102 percent (%) Rated Core Power
- Flow **[[ ]]**
- Power Shape Top-Peaked Axial

ANP-3934P, Table 6.1, describes the following parameters for the TLO LOCA analysis case that produced the **[[ ]]** result:

- **[[**

]]

ANP-3934P, Table 9.1, provides the exposure-dependent LOCA analysis results. The following limiting highest results are obtained at **[[ ]]** for the same case parameters as above:

- **[[**

]]

The results show that the 10 CFR 50.46(b)(1) through (b)(3) acceptance criteria for PCT ≤ 2200 ºF, MLO ≤ 17 percent, CMWR ≤ 1 percent respectively are satisfied. The results also demonstrate the acceptability of the ATRIUM 11 MAPLHGR limit shown in ANP-3934P, Figure 2.1.

The regulation at 10 CFR 50.46(b)(4) requires that calculated changes in core geometry shall be such that the core remains amenable to cooling. ANP-3934P stated that compliance with 10 CFR 50.46(b)(1) through (b)(3) ensures that the core coolable geometry is maintained. In addition, ANP-3882P (Reference 9), section 3.4 addressed the ATRIUM 11 fuel coolability and component structural deformation in section 3.4.4 under accident conditions. The NRC staff evaluation of ANP-3882P is given in section 3.9 of this SE.

The NRC finds compliance with 10 CFR 50.46(b)(4) is satisfied because the 10 CFR 50.46(b)(1) through (b)(3) acceptance criteria are met and the structural integrity of the core components is maintained under accident conditions.

For compliance with 10 CFR 50.46(b)(5) regarding long-term coolability, ANP-3934P states:

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

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

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

3.5.3 Evaluation of ANP-10332P-A Limitations and Conditions

The AURORA-B LOCA methodology has 27 limitations and conditions listed in section 5.0 of the NRC staff's SE for ANP-10332P-A. ANP-3934P, appendix A, provides disposition on how these limitations and conditions are met. The following is NRC staff evaluation to confirm these limitations and conditions are satisfied:

Limitation and Condition 1

> The AURORA-B LOCA evaluation model shall be supported by an approved nodal core simulator and lattice physics methodology. Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall identify the nodal core simulator and lattice physics methods supporting the AURORA-B

LOCA analysis and reference an NRC-approved TR confirming their acceptability for the intended application.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the nodal core simulator and lattice physics methodology from the NRC-approved TR EMF-2158(P)(A) Revision 0 (Reference 27) are used.

Limitation and Condition 2

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

Evaluation

The NRC staff finds this limitation and condition is satisfied because the stand-alone version of RODEX4 to provide the steady-state fuel thermal-mechanical input in accordance with the NRC-approved methodology BAW-10247P-A is used.

Limitation and Condition 3

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

Evaluation

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

Limitation and Condition 4

TR ANP-10332P **[[**

]

Evaluation

The NRC staff evaluation of this limitation and condition is provided in section 3.5.2 of this SE under the heading "Results".

Limitation and Condition 5

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

Evaluation

The NRC staff finds this limitation and condition is satisfied because the applicable regulatory requirements for the licensee are the same as the regulatory requirements in effect at the time the NRC staff's review for ANP-10332P was completed.

Limitation and Condition 6

This SE does not **[[**

]] of the evaluation model.

Evaluation

The NRC staff finds this limitation and condition is satisfied because the AURORA-B EM **[[** **]]**

Limitation and Condition 7

[[

]]

Evaluation

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

Limitation and Condition 8

[[

]]

Evaluation

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

Limitation and Condition 9

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

Evaluation

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

]

Limitation and Condition 10

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

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the ANP-3934P statement that a **[[ ]].**

Limitation and Condition 11

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[[**
>
> **]]**

Evaluation

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

]].

Limitation and Condition 12

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

Evaluation

The analyses **[[**

]].

Limitation and Condition 13

> **[[**
>
> **]** (section 3.3.4.1.4)

Evaluation

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

Limitation and Condition 14

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[[**
>
> **]**

Evaluation

The FoM for MNGP in the **[[** **]]** Therefore, the NRC staff finds that this limitation and condition is adequately addressed.

Limitation and Condition 15

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

Evaluation

The NRC staff finds this limitation and condition is satisfied because ANP-3934P stated that in the analysis the **[[**
]]

Limitation and Condition 16

Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[[**
]]

Evaluation

Considering MNGP is licensed to the EFW domain, sufficient initial state points were considered in the break spectrum analysis to provide confidence that the most limiting conditions have been analyzed. **[[**
] Based on this the NRC staff finds this limitation and condition is satisfied.

Limitation and Condition 17

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

Evaluation

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

Limitation and Condition 18

Safety analyses performed with the AURORA-B LOCA evaluation model **[[**
]].

Evaluation

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

Limitation and Condition 19

> Safety analyses for **[[**
>
> **]].**

Evaluation

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

- **[[**

]

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

Limitation and Condition 20

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

]]

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the ANP-3934P statement that simulations **[[**
]] The NRC staff confirmed this during the regulatory audit that **[[**

]]

<u>Limitation and Condition 21</u>

> As discussed in section 3.3.5.7, Framatome used a **[[**
>
> **]]**

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the ANP-3934P statement that this **[[** **]].**

<u>Limitation and Condition 22</u>

> The NRC staff has not specifically reviewed any plant parameters in ANP-10332P or deemed them acceptable for use in plant safety analyses. Therefore, **[[**
>
> **]]** (section 3.6)

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the ANP-3934P statement that the licensee **[[**

]].

Limitation and Condition 23

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

]] (sections 3.4.3.5 and 3.6.2.1)

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the ANP-3934P statement that the [[

]

Limitation and Condition 24

[[

]

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the ANP-3934P statement that [[

]]

Limitation and Condition 25

Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model [[

]]

Evaluation

The NRC staff finds this limitation and condition is satisfied based on the following justification provided in ANP-3934P for the changes in the AURORA-B LOCA EM:

- The **[[**

  ]].

Limitation and Condition 26

> Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[[**
>
> **]**

Evaluation

In its review, the NRC staff finds **[[** **]]** and therefore this limitation and condition is satisfied.

Limitation and Condition 27

> As discussed in section 4.3 of this SE, new or modified Framatome **[[**
>
> **]**

Evaluation

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

3.6.4 Conclusions - LOCA Analysis

The NRC staff reviewed the information in the licensee's submittal, ANP-3934P, and the responses to RAIs in letter dated June 6, 2022, and concludes that the LOCA analysis for MNGP with ATRIUM 11 fuel is acceptable because it complies with the relevant requirements of 10 CFR 50.46, Appendix K to 10 CFR, Part 50, and GDC 35. The NRC staff conclusion is based on the following:

- The analyses of the performance of the ECCS with ATRIUM 11 fuel were performed in accordance with 10 CFR 50.46.

- The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model that follows Appendix K to 10 CFR, Part 50, and meets the requirements of 10 CFR 50.46. The results of the analyses show that the 10 CFR 50.46(b)(1) through (b)(3) acceptance criteria for PCT ≤ 2200 ºF, MLO ≤ 17 percent, and CMWR ≤ 1 percent, respectively, are satisfied.

- Compliance with 10 CFR 50.46(b)(1), (b)(2), and (b)(3) criteria ensures that 10 CFR 50.46(b)(4) on maintaining a coolable geometry is satisfied.

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

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

- The analyses apply the NRC-approved LOCA EM (evaluation model) and methodology for the LOCA analysis with ATRIUM 11 fuel and adequately meet the limitations and conditions listed in the NRC staff's SE for the applied licensing technical reviews.

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

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

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

- Considering MNGP is licensed to the EFW, sufficient initial state points were considered in the break spectrum analysis to provide confidence that the most limiting conditions have been analyzed. The NRC staff finds the selected analysis state points acceptable because the licensee took the appropriate regulatory guidance into account with respect to analyzing the EFW domain.

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

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

- The thermal-hydraulic compatibility analysis demonstrates that the thermal-hydraulic characteristics of the ATRIUM 11 and the coexistent ATRIUM 10XM fuel are similar so that the core responses during LOCA will be insignificant for transition cores.

- The LOCA analysis **[[**

  ]]

- These results demonstrate the acceptability of the ATRIUM 11 MAPLHGR limit shown in ANP-3934P, Figure 2.1.

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

3.6 Stability Best-Estimate Enhanced Option (BEO)-III (ANP-3932P)

The evaluation of the application of BEO-III methodology with period-based detection algorithm (PBDA) is discussed in ANP-3932P (Reference 8).

The BEO-III methodology was approved in ANP-10344P-A. MNGP uses the PBDA as the primary stability protection feature, the approved algorithm for use with the BEO-III methodology. The PBDA algorithm tracks the number of confirmation counts, or successive oscillations which have a characteristic period which is within a small tolerance compared to the average of the previous periods. The oscillating power range monitors (OPRM) are tripped when the PBDA algorithm counts, or magnitude exceeds setpoints. When a reactor trip occurs, the oscillations are suppressed by control rod insertion. ANP-3932P provided a sample analysis which established the cycle-specific operating limit maximum critical power ratio (OLMCPR) based on statistical analyses of pump trip scenarios and evaluation of the time dependent local power range monitors (LPRM) and core minimum critical power ratio to determine the most limiting event based on the MNGP PBDA detect and suppress hardware response.

3.6.1 Regulatory Evaluation

The plant-specific BEO-III long term stability solution (LTSS) and related licensing basis were developed to comply with the requirements of GDCs 10 and 12 in Appendix A to 10 CFR, Part 50.

GDC 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."

GDC 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

Backup Stability Protection

The backup stability protection is provided in accordance with the ANP-3932P to be active when the OPRM system is declared inoperable.

RAMONA5-FA Qualification for MNGP

The licensee collected OPRM data during the Long-Term Stability Solution Option III Implementation testing. RAMONA5-FA statistical calculations of the OPRM amplitudes were performed and the calculated results were compared to the measured amplitudes. The testing was performed during Cycle 25 before the EPU power uprate. The testing was completed at three different power/flow combinations and then these points were converted to the current rated power and flow. The statistical calculations of the OPRM amplitudes were performed and the calculated results were compared to the measured amplitudes. **[[**

]

ATRIUM 11 Equilibrium Cycle Sample Application

The equilibrium cycle sample analyses demonstrate the PBDA can detect and suppress oscillations for ATRIUM 11 fuel with a high degree of confidence. All of the full power cycle exposures in the equilibrium cycle design depletion were assessed **[[**

]] consistent with the approved methodology.

The analysis used sample parameters for the statistical analysis that are consistent with the approved methodology. There are **[[**

]

The analysis used representative values for the PBDA setpoints in the sample application. The analysis used a setpoint of **[[ ]]** The values used for licensing of the actual core design will be specified once the design is established.

Limiting EFW 2RPT Scenario

The limiting EFW 2RPT scenario is analyzed at 100 percent rated core power and 80 percent rated core flow. The sample size used for the evaluation was **[[ ]]** trials and the licensee provided the minimum critical power ratio (MCPR) FoM. Based on a **[[ ]]** OLMCPR, the 95/95 MCPR of **[[ ]]** is shown to be well above the **[[**

]]. The analysis demonstrated that the assumption that the reactor protection system can detect and suppress the oscillations prior to violation of the specified acceptable fuel design limits.

Evaluation of Pump Coastdown

The pump coastdown was evaluated to determine if the FoM is impacted by causing a reset in the PBDA confirmation counts that result in a delayed reactor trip. The evaluation found that **[[**

]]

Potentially Limiting Scenarios

ANP-10344P was reviewed for applicable pump trip scenarios. MNGP is not licensed for any reduced feedwater temperature scenarios and therefore, the only potentially limiting scenario is the pump trip from single loop operation (SLO) conditions. This limiting pump trip is analyzed at 66 percent rated power and 52.5 percent rated flow. The evaluation found that the core MCPR FoM is **[[**

]]

Tmin Confirmation

ANP-3932P documents that the Tmin oscillation period of **[[ ]]** seconds specified as the minimum oscillation period for the MNGP plant-specific detect and suppress hardware is acceptable by evaluating the limiting pump scenarios. The EFW 2RPT scenario yielded a minimum period, Tmin, of **[[ ]]** seconds. The 1PT from the EFW corner produced **[[**

]]

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

]] Each of the EFW 2RPT trials and the limiting trial for the EFW 1RPT scenario [[ ]] was evaluated. Based on these analyses the Tmin value of [[ ]] seconds is confirmed to bound the minimum oscillation period for any credible oscillations.

The [[

]]

3.6.3 Evaluation of ANP-10344P-A Limitations and Conditions

The following is the NRC staff evaluation of the disposition of the limitations and conditions associated with ANP-10344P-A:

<u>Limitation and Condition 1</u>

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

Evaluation

MICROBURN-B2 is the specified core simulator for the MNGP Application Methodology.

<u>Limitation and Condition 2</u>

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

Evaluation

All settings and modeling options, including core and vessel nodalization and time step control parameters are consistent with the validation basis.

<u>Limitation and Condition 3</u>

> [[
>
>]

Evaluation

[[

]]

Limitation and Condition 4

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

Evaluation

The **[[**
]] for protecting the SAFDL.

Limitation and Condition 5

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

Evaluation

The RAMONA5-FA code utilized for the MNGP Application Methodology was evaluated as per existing regulatory processes. There are no changes that would alter the basis of the methodology.

Limitation and Condition 6

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

Evaluation

The concern associated with this limitation and condition is that variations in the pump coastdown rate could lead to system resets that would cause the reactor trip to be delayed and allow the amplitude to grow to larger magnitudes. The assessment of **[[**

]] on the final MCPR for the ATRIUM 11 equilibrium cycle.

Limitation and Condition 7

If the 1RPT EFW event remains stable, additional analyses are required using **[[ ]]** to ensure that the lowest oscillation period remains above Tmin under any anticipated conditions.

Evaluation

As presented in Section 4.3, the 1RPT from the minimum core flow at rated power in the EFW domain **[[ ]]**

Limitation and Condition 8

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

Evaluation

The **[[ ]].** Review of the FoMs trend as a function of exposure indicates that there is a reasonable assurance that the limiting exposure points were analyzed.

The analysis provided in ANP-3929P has adequately followed the methodology and addressed the limitations and conditions associated with the LTR ANP-10344P-A.

3.6.4 Conclusions- Application BEO-III Methodology

The NRC staff evaluated the information in ANP-3932P and concludes that the BEO-III calculation for MNGP in accordance with ANP-10344P-A provides an acceptable means of determining licensing basis SLMCPR protection during the anticipated stability events. All limitations and conditions specified in the SE for ANP-10344P-A have been satisfied and GDCs 10 and 12 requirements are met.

3.7 ATRIUM 11 Fuel Assembly/Rod Design (ANP-3882P and ANP-3903P)

3.7.1 Regulatory Evaluation

The ATRIUM 11 fuel (assembly/rod) design was developed using the thermal mechanical design bases and limits outlined in ANF-89-98(P)(A) (Reference 34), compliance with which ensures that the fuel design meets the fuel system damage, fuel failure, and fuel coolability criteria identified in the SRP. The SRP is intended to provide comprehensive guidance for

NRC staff review of whether LARs satisfy regulatory requirements, including the evaluation of the safety of light-water nuclear power plants and the review of safety analysis reports.

SRP, Section 4.2, "Fuel System Design"; Section 4.3, "Nuclear Design"; and Section 4.4, "Thermal and Hydraulic Design," provide regulatory guidance for the review of fuel rod cladding materials, the fuel system, the design of the fuel assemblies and control systems, and the thermal and hydraulic design of the core. In addition, the SRP provides guidance for compliance with the applicable GDC in Appendix A to 10 CFR, part 50. In accordance with SRP, Section 4.2, the 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 reviewed the LAR to evaluate the applicability of Framatome BWR methodology to the use of ATRIUM 11 fuel at MNGP to confirm that the use of the methodology is within the NRC-approved ranges of its applicability and to verify that the results of the analyses comply with the requirements of the following GDCs in Appendix A to 10 CFR, part 50:

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

- GDC 27, "Combined reactivity control systems capability," requiring that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions.

- GDC 35, "Emergency core cooling," requiring that a system to provide abundant emergency core cooling is provided 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.

3.7.2 Technical Evaluation

ANP-3882P provides the mechanical design details, fuel structural analysis results of the ATRIUM 11 fuel assemblies, and fuel channel designs, while ANP-3903P (Reference 11) provides the design parameters and design evaluation results of the ATRIUM 11 fuel rods to be used at MNGP.

3.7.2.1 Summary of Mechanical Design of ATRIUM 11 Fuel Assemblies for MNGP

ANP-3882P provides key fuel assembly design details for the Framatome ATRIUM 11 fuel assembly design planned for use at MNGP. The fuel design is comprised of a 11x11 array of fuel rods with a square internal water channel that displaces a 3x3 array of rods, with **[[**

]] Table 2-1 of ANP-3882P lists the fuel assembly and component description of the ATRIUM 11 fuel assembly design. Further descriptions of the fuel assembly components are provided in ANP-3882P. The NRC staff noted that most of the changes relative to the ATRIUM 10XM fuel assembly design are evolutionary changes. The exceptions include the use of a 11x11 array of fuel rods, **[[**

]], the use of chromia-doped fuel pellets, the use of non-lined SRA cladding, and the use of Z4B material for the water channel and some fuel channels. The change in geometry to an 11x11 fuel array 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 in EMF-93-177P-A, "Mechanical Design for BWR Fuel Channels," Revision 1, dated August 2005, and Supplement 1P-A, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," Revision 0, dated September 2013 (Reference 35) and Z4B material in reactor cores in EMF-93-177, "Framatome Inc. EMF-93-177-NP-A Suppl 2P "Mechanical Design for BWR Fuel Channels: Z4B Material" (Reference 36).

3.7.2.2 Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Assembly Design

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

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

ANF-89-98(P)(A) provides some generic mechanical design criteria that were approved by the NRC for use with evaluation of Framatome fuel designs. The ATRIUM 11 fuel mechanical design as reported in ANP-3882P, as discussed below in the 3.7.2.3 of this SE, describes how the design criteria presented in ANF-89-98(P)(A) apply to the ATRIUM 11 fuel assembly mechanical design.

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

The NRC staff's SE for EMF-93-177P-A specified several limitations and conditions that have already been shown to be met at MNGP for the channels associated with the ATRIUM 10XM fuel. Since the ATRIUM 11 channels are very similar, the disposition of the limitations and conditions remains applicable. The two exceptions are the use of Z4B channels, as approved in Supplement 2P-A and interior milling, which is addressed through the use of the Supplement 1P-A methodology. The Supplement 1P-A methodology was approved with no limitations or conditions.

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

The ATRIUM 11 fuel mechanical design evaluation, as discussed in section 3.7.2.3 of this SE, confirms that the **[[** **]]** and that **[[** **]]**. The remaining limitations and conditions are met for the ATRIUM 11 fuel assembly design, since the channels are constructed of either Zircaloy-4 or Z4B, and the fuel rod materials fall within the range of applicability for the database used to support the fuel rod growth correlations.

3.7.2.3 Fuel Assembly Mechanical Design Evaluation

The objectives of the fuel design are that (i) the fuel assembly (system) is not damaged as a result of normal operation and AOOs, (ii) fuel system damage is never so severe as to prevent control rod insertion when it is required, (iii) the number of fuel rod failures is not underestimated for postulated accidents, (iv) fuel coolability is always maintained as documented in ANP-3882P, (v) the mechanical design of the fuel assemblies shall be compatible with co-resident fuel and the reactor core internals, and (vi) fuel assemblies shall be designed to withstand the loads from handling and shipping. The first four objectives are from SRP, Section 4.2, and the latter two are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel (co-resident fuel). This fuel assembly mechanical design evaluation contains only fuel assembly structural analyses, while the fuel rod evaluation, as documented in ANP-3903P to the LAR is discussed in section 3.7.2.6 of this SE.

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

ANP-3882P states that the ASME BPV Code was used as a guide to establish the acceptable stress, deformation, and load limits for standard assembly components. These limits are applied to the design and evaluation of the upper tie plate (UTP), lower tie plate (LTP), spacer grids, springs, and load chain components, as necessary and applicable. The fuel assembly structural component criteria under faulted conditions are based on appendix F of the ASME Code, Section III, with some criteria derived from component tests.

Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. In summary, analyses were performed to determine the mechanical performance of assembly components during accidents (e.g., seismic events or LOCA events), fuel handling events, or during normal and AOO conditions.

For accident conditions, the dynamic characteristics of the fuel assembly and grids were obtained from testing the assemblies for stiffness, natural frequencies, and damping values, and 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 like other BWR fuel designs that have the same basic channel configuration and weight. The 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 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 **[[**

]]

For fuel structural characteristics for normal and AOO conditions, the licensee performed evaluations on the stress for ATRIUM 11 fuel channels due to pressure differential and found that the pressure load, including AOO, meets the ASME BPV Code criteria of **[[**
]] and **[[**

]]. The stress as a result of vertical acceleration is found to be less than allowable. Hence, the deformation during AOO remains within functional limits for normal control blade operation.

Based on the above, the NRC staff finds the evaluation acceptable because the evaluation is complete and adequate to meet the required design criteria and is consistent with the SRP guidance.

<u>Fatigue and Fretting Wear</u>

Fatigue of structural components is generally 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 per Section 2.3 of ANF-89-98(P)(A). While some of the fuel channels will be constructed with Z4B rather than conventional zirconium alloys, **[[**
]] 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. Although the testing period is short relative to the time that a fuel assembly will typically spend in the reactor core, this result is sufficient to provide reasonable assurance that structural flaws in the fuel rod cladding would not be expected to lead to widespread fuel rod failures.

The NRC staff finds that based on the fatigue loads, the fuel channels will continue to perform their function and will not interfere with control blade insertion. Furthermore, the NRC staff finds that based on the results of the fretting wear testing, widespread rod failures would not be expected because of fretting effects. The NRC staff notes that isolated rod failures due to localized mechanisms leading to excessive fretting are not explicitly required by regulatory acceptance criteria to be addressed; therefore, the generic testing performed in support of this conclusion was sufficient to establish a regulatory finding.

<u>Rod Bow</u>

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

] A **[[** **]]** has been developed as described in BAW-10247P-A, Supplement 2P-A. The NRC has approved the use of the BAW-10247P-A, Supplement 2P-A, correlation for all current and future Framatome BWR fuel designs up to an **[[** **]]**, provided that the change process described in section 5.0, "Change Process," is followed.

<u>Axial Irradiation Growth</u>

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. In accordance with BAW-10247P-A, Supplement 2P-A, **[[**

]] The channel material that will be used in MNGP Z4B is within the scope of the NRC approval of BAW-10247P-A, Supplement 2P-A. Furthermore, the NRC staff considered and accepted data for the ATRIUM 11 fuel assembly design as part of the basis and applicability for 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 as discussed above.

<u>Assembly Liftoff</u>

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

]

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

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

]]. The fuel will not lift under normal or AOO conditions. It will not become disengaged from the fuel support under faulted conditions or block the insertion of the control blade in all operating conditions.

In its June 6, 2022, letter, the licensee provided steps for calculating assembly liftoff during normal operating conditions to ensure the fuel does not separate from the fuel support. The NRC staff finds the steps described by the licensee acceptable because it satisfies the objectives in the SRP.

In its June 6, 2022, letter, the licensee described how the criteria for assembly liftoff is satisfied for faulted or accident conditions. The NRC staff finds the steps described by the licensee acceptable because it satisfies the objectives in the SRP.

Based on the above, the NRC staff finds the liftoff evaluation acceptable because the evaluation is complete and adequate to meet the required design criteria and satisfy the SRP objectives.

<u>Fuel Channel Irradiation Induced Changes</u>

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. The proposed fuel channels are constructed of Z4B, which was approved by the NRC as part of EMF-93-177, Revision 1, Supplement 2P-A, Revision 0. **[[**

]]. 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 use of Z4B material with the EMF-93-177 methodology was reviewed by the NRC in Supplement 2P-A. **[[**

]

Summary of Sections 3.7.2.1 through 3.7.2.3

Tables 3-1 and 3-2 of ANP-3882P to the LAR 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 summarized above. As a result, the NRC staff finds that evaluations have been performed acceptably to ensure that the mechanical design criteria for the ATRIUM 11 fuel assembly design are met for use in the MNGP reactor core.

3.7.2.4 Summary of ATRIUM 11 Fuel Rod Thermal-Mechanical Design for MNGP

ANP-3903P provides key fuel rod design details for Framatome ATRIUM 11 fuel planned for use at MNGP. The ATRIUM 11 fuel rod is conventional in design configuration and is very similar to past designs such as the ATRIUM 10XM and ATRIUM 10 fuel rods. **[[**

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

There are two-part length fuel rod (PLFR) designs incorporated in the fuel assembly. **[[**

]].

Table 3-1 of ANP-3903P lists the key fuel rod design parameters for the ATRIUM 11 fuel. Further descriptions of the fuel assembly components are provided in ANP-3903P.

3.7.2.5 Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Rod Design

The specific evaluations for the ATRIUM 11 fuel rod design, used specific NRC-approved methodologies. NRC staff approval of these methodologies is conditional on adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the topical reports. A discussion of how these limitations and conditions are met for MNGP is provided below for each of the topical reports directly supporting the ATRIUM 11 fuel rod design evaluations, as well as a discussion of the applicability of topical reports already in use at MNGP for analysis of the ATRIUM 10XM fuel rod design that may not automatically apply to the ATRIUM 11 fuel rod design.

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

ANF-89-98(P)(A) provides some generic fuel rod design criteria that were approved by the NRC staff for use with evaluation of Framatome fuel designs. The ATRIUM 11 fuel rod design as reported in ANP-3903P, as discussed in section 3.7.2.6 of this SE, describes how the design criteria presented in ANF-89-98(P)(A) apply to the ATRIUM 11 fuel rod design.

- BAW10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, dated February 2008 (Reference 21)

Section 3.7.2.6 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 38) contains a similar limitation and condition on the **[[ ]]**, which is addressed through an automated software check. The remaining limitations and conditions are addressed by only utilizing the methodology within the bounds defined by the limitations and conditions.

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

The chromia-doped fuel properties and models described in this TR are directly applicable to the ATRIUM 11 fuel pellets. The limitations and conditions are met through a combination of automated software checks and administrative controls, as described in Section 2-18 of the BWR compendium. The automated software checks are managed through the Framatome software quality assurance program, which is subject to normal NRC oversight activities as part of verifying compliance with Appendix B to 10 CFR Part 50.

3.7.2.6 ATRIUM 11 Fuel Rod Design Evaluation

The NRC staff's review of fuel rod thermal-mechanical analyses for the ATRIUM 11 fuel was performed using acceptance criteria from ANP-8998(P)(A), Revision 1, and Supplement 1 and the RODEX4 analysis methodology described in BAW10247PA. The methodology described in ANP-10340P-A 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.

<u>Fuel Rod Design Evaluation</u>

The ATRIUM 11 fuel assembly design contains multiple changes in geometry to accommodate the change from a 10x10 rod array to an 11x11 rod array within the same basic channel dimensions. The part length rod specifications also differ from the ATRIUM 10XM design. The ATRIUM 11 fuel also uses two 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-3903P. The fuel rod geometry and compositions generally fit within the applicability of the NRC-approved RODEX4 thermal-mechanical analysis methodology, with the addition of the chromia-doped fuel properties and models reviewed and approved by the NRC staff.

Therefore, the RODEX4 code was used to evaluate the fuel rod thermal-mechanical performance of the ATRIUM 11 fuel rod design, as appropriate.

In its June 6, 2022, letter, the licensee specifically described the neutronic impact of chromia additive in the fuel, as well as the impact of chromia additive on fission gas release, fuel densification and swelling, corrosion, and creep in its RAI responses. The licensee referenced the applicable sections in ANP-10340P-A which addressed these items. Table 2-1 of

ANP-3903P 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 EFW 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.

Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and the formation of hydride platelets. As stated in Section 3.3.1 of ANP--3903P, a fabrication limit is imposed **[[ ]]** and enforced via moisture controls. The NRC staff finds this to be an acceptable approach to ensure that the potential sources for hydrogen absorption inside the cladding are minimized, since the fabrication limit is based on NRC-approved mechanical design criteria.

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 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 and, therefore, is acceptable.

Overheating of Fuel Pellets

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

Stress and Strain Limits

Under transient conditions, the inner diameter of the cladding may shrink more rapidly than the outer diameter of the fuel pellet due to differences in their rates of change in temperature. If the cladding surface presses on the outside of the fuel pellet, this results in the pellet-clad interaction phenomenon. The pressure of the fuel pellet resisting the shrinkage of the cladding can cause local deformation of the cladding or cladding strain. The RODEX4 methodology is used to calculate the predicted cladding strain **[[ ]]** to confirm that the strain is no more than one percent.

This is consistent with the RODEX4 methodology, and the one percent strain limit is consistent with the NRC-approved fuel rod evaluation methodology 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 were then compared against the design limits prescribed by Section III of the ASME Code. This is consistent with NRC-approved mechanical design criteria and, therefore, is acceptable.

<u>Fuel Densification and Swelling</u>

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. The NRC staff has reviewed and approved the models used in RODEX4 to address these phenomena; therefore, this is an acceptable disposition.

<u>Fatigue</u>

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

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

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

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

<u>Oxidation, Hydriding, and Crud Buildup</u>

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, 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 MNGP is provided below.

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

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

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

ANP-3903P describes the process that would be required to follow in the **[[** **]]** The NRC staff finds the process acceptable because it was reviewed and approved in BAW-10247PA.

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

<u>Rod Internal Pressure</u>

The fuel rod internal pressure is calculated using the RODEX4 code and methodology. 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).

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.

<u>Summary of Sections 3.7.2.4 to 3.7.2.6</u>

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) and BAW-10247PA, in the fuel rod thermal-mechanical analyses for the ATRIUM 11 fuel design that is planned to be loaded and used for operation at MNGP. The NRC staff determined that the fuel design criteria, as supported by the applicable regulations and sections of NUREG-0800, have been satisfied and provide reasonable assurance of safe operation at MNGP.

3.7.3 Conclusion of ATRIUM 11 Fuel Assembly/Rod Design

For evaluation of the ATRIUM 11 fuel assembly/rod design, the NRC staff concludes that the application of ATRIUM 11 fuel (fuel assembly and fuel rod) to MNGP is acceptable because it complies with the requirements of GDCs 10, 27, and 35. This conclusion is based on the following:

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

    a. Developing and complying with fuel system damage criteria for all known damage mechanisms and operating conditions as evaluated in sections 3.7.2.3 (Fuel Assembly Mechanical Design Evaluation) and 3.7.2.6, (ATRIUM 11 Fuel Rod Design Evaluation) and

    b. Applying NRC-approved fuel system design methodologies and adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the applied TRs as evaluated in sections 3.7.2.2 (Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Assembly Mechanical Design) and 3.7.2.5 (Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Rod Design)

2. The application meets the requirements of GDC 27 with respect to the reactivity control system being designed with margin to have capability of reliably controlling reactivity changes by ensuring that fuel system damage is never so severe as to prevent control rod insertion when it is required. For example, as evaluated in section 3.7.2.3 (Fuel Assembly Mechanical Design Evaluation) of this SE, the fatigue and fretting wear of the fuel assembly components was tested to ensure that it does not interfere with control blade insertion. As demonstrated by analysis, the fuel will not lift under normal or AOO conditions. It will not become disengaged from the fuel support under faulted conditions or block insertion of the control blade in all operating conditions. The fuel channel was specifically evaluated for changes due to exposure to the reactor environment that may lead to loss of strength or deformation to affect the control rod insertability.

3. The application meets the requirements of GDC 35 with respect to the fuel system being able to transfer heat from the reactor core following any loss of reactor coolant at an acceptable rate by ensuring that the fuel rod damage does not interfere with effective emergency core cooling and that the cladding temperatures do not reach a temperature high enough to allow a significant metal-water reaction to occur. These assurances are achieved by developing and complying with the fuel coolability-related criteria for all severe fuel rod damage mechanisms as addressed in section 3.7.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) (e.g., internal hydriding, cladding collapse, overheating of fuel pellets, cladding stress and strain limits, fuel densification and swelling, and clad oxidation, hydriding, and crud buildup). The application applied NRC-approved RODEX4 fuel rod evaluation methodology and adequately met the limitations and conditions listed in the NRC staff's SE for each of the applied topical reports.

3.8 Thermal-Hydraulic Design Compatibility of ATRIUM 11 Fuel Assemblies (ANP-3893P)

3.8.1 Regulatory Evaluation

The ATRIUM 11 fuel design was developed using the thermal-mechanical design bases and limits as outlined in ANF-89-98(P)(A), consistency with which ensures that the fuel design meets the criteria for fuel system damage, fuel failure, and fuel coolability identified in Section 4.2 of the SRP. The SRP is intended to provide comprehensive guidance for NRC staff review of whether LARs satisfy regulatory requirements, including the evaluation of the safety of light-water nuclear power plants and review of safety analysis reports.

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

In accordance with SRP, Section 4.2, the 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 reviewed the LAR to evaluate the applicability of Framatome BWR methodology to the use of ATRIUM 11 fuel at MNGP to confirm that the use of the methodology is within NRC-approved ranges of its applicability and to verify that the results of the analyses comply with the requirements of GDCs 10, 12, 15, 20, 25, 26, 27, 28, and 35 (see the following sections below for further discussion).

3.8.2 Technical Evaluation

This section describes the NRC staff's evaluation of the licensee's thermal-hydraulic analyses to demonstrate the hydraulic compatibility of ATRIUM 11 fuel with the co-resident ATRIUM 10XM fuel at MNGP. The LAR is proposing to transition from the current ATRIUM 10XM fuel design to ATRIUM 11 fuel. Attachment 5c to the LAR provides the results of the thermal-hydraulic analyses to support a finding that ATRIUM 11 fuel is hydraulically compatible with the co-resident ATRIUM 10XM fuel. The results from the thermal-hydraulic analyses are compared to acceptance criteria established in NRC-approved topical reports ANF-89-98(P)(A), Revision 1, Supplement 1, and XN-NF-80-19(P)(A), Volume 4, Revision 1 (Reference 40).

Thermal-hydraulic analyses were performed to verify that the design criteria were satisfied and to 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 plant and cycle specific basis and are documented in plant and cycle specific reports. Table 3.1 of ANP-3893 (Reference 10) lists the applicable thermohydraulic design criteria, analyses, and results for hydraulic compatibility, thermal margin performance, fuel centerline temperature, rod bow, bypass flow, stability, LOCA analysis, CRDA analysis, ASME over-pressurization analysis, and seismic/LOCA liftoff. The subsections below summarize the results from selected design criteria and analyses results.

Hydraulic Characterization

Basic dimension parameters for the ATRIUM 10XM and ATRIUM 11 fuel assembly designs are summarized in Table 3.2 of ANP-3893. Table 3.3 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-3893.

The testing and analysis approaches used for the ATRIUM 11 fuel assembly design are similar to the approaches that have previously been used to characterize the ATRIUM 10XM fuel assembly design, as reviewed by the NRC for applicability to other plants operating in the EFW flow regime. 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.

Thermal-hydraulic Compatibility

The thermal-hydraulic compatibility analyses were performed in accordance with the Framatome thermal-hydraulic methodology for BWRs provided in XF-NF-80-19(P)(A). 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 are planned to be inserted in the MNGP core. The analyses were performed for full core and mixed core configurations.

In essence, the hydraulic compatibility analysis **[[** **]]** This analysis is performed using 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 ANP-3893, while representative results are given in Tables 3.5 through 3.8 and Figures 3.2 and 3.3. **[[** **]]** The most important result from the perspective of thermal-hydraulic compatibility is that 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 correlations and specifications unique to each fuel assembly design.

The hydraulic compatibility of ATRIUM 11 fuel assemblies with the co-resident ATRIUM 10XM fuel was verified through an NRC staff RAI. Thermal-hydraulic compatibility was analyzed by the licensee and at seven power-to-flow state points across the entire power-to-flow operating domain, including the extended flow window, and the results are provided by letter dated June 6, 2022.

Based on the changes in **[[ ]]** 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 MNGP, provide reasonable assurance that the resident and co-resident fuel designs will satisfy the thermal-hydraulic design criteria for mixed cores.

Thermal Margin Performance

The thermal margin analyses were performed using the NRC-approved thermal-hydraulic methodology for steady-state critical power ratio (CPR) evaluations with XCOBRA. Empirical correlations for the ATRIUM 10XM (provided in ANP-10298P-A) and for ATRIUM 11 (provided in ANP-10335P-A) fuel assembly designs were used based on results of boiling transition test programs. These CPR correlations account for assembly design features through modification of the K factor term in the CPR correlations.

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

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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 constraints on design and operation of the cores throughout the transition.

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.

Rod Bow

Rod bow is addressed as part of the mechanical design analyses (see section 3.7.2.3 of this SE (Fuel Assembly Mechanical Design Evaluation) for further discussion). **[[**

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The NRC staff finds this disposition to be acceptable based on the fact that it is consistent with Framatome methodologies, and the impact is appropriately evaluated.

Bypass Flow

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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.

Stability

The thermal-hydraulic design criteria approved by the NRC in ANF-8998(P)(A) include a requirement to confirm that the stability characteristics for a new fuel design are equivalent to or better than that of prior approved fuel designs. This evaluation is performed using the STAIF code as prescribed in ANF8998(P)(A), and the results are documented in ANP-3893 for MNGP. This evaluation is adequate to meet the requirements within the NRC-approved generic fuel assembly mechanical design criteria used by Framatome to qualify new fuel designs. However, the NRC staff did not review the STAIF evaluation in detail because the period-based detection algorithm trip is expected to detect and suppress any power oscillations resulting from stability issues, as confirmed using the BEO-III analytical methodology. 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 have not been invalidated. This will ensure that the regulatory requirements associated with stability performance are met.

3.8.3 Thermal-Hydraulic Compatibility Design Conclusion

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 MNGP. The NRC staff determined that the generic thermal-hydraulic design criteria, as approved by the NRC in ANF-89-98(P)(A), have been used in the analyses. Based on the above, the NRC staff concludes 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.

3.9 ATRIUM 11 Equilibrium Fuel, Nuclear Fuel Design (ANP-3877P)

ANP-3877P (Reference 12) provides results from neutronic design analyses for MNGP ATRIUM 11 equilibrium design using the approved topical report ANF-89-98(P)(A) and the design criteria in the topical report EMF-2158(P)(A). The fuel design assumptions include **[[**

]].

Neutronic design analysis meets applicable design criteria, as well as reactivity and control requirements.

Neutronic design parameters including local power distribution in the fuel combined with the core power distribution and shall result in LHGR and MCPR values. Kinetic parameters include moderator void reactivity coefficients, Doppler fuel temperature reactivity coefficients. The control blade reactivity will include TS shutdown requirements that should be met for all reactor operating conditions.

The neutronic design parameters for the fabrication batch are presented in Table 2.1 of ANP-3877P and contain the number of fuel assemblies in the core, uranium enrichments, gadolinia enrichments and its distributions, and fuel rod distribution and axial distribution. Kinetics parameters are calculated for fuel temperature (Doppler), moderator void, and moderator temperature. The Doppler reactivity is presented over a fuel temperature range from hot standby to hot operating. The moderator void reactivity was evaluated between the 0% and 40% voided hot operating cases. Tables 2.2 through 2.8 in ANP-3877P list lattice control blade worth at beginning of life for various control blades and their corresponding kinetics parameters. All pertinent fuel and reactor core design information are given in Appendices A through D. The NRC staff reviewed the results of the neutronic design analyses presented in ANP-3877P and determined that the fuel was designed to meet applicable design criteria, as well as reactivity and control requirements.

3.10 ATRIUM 11 Equilibrium Cycle, Fuel Cycle Design (ANP-3881P)

ANP-3881P (Reference 13) documents the results from an equilibrium cycle design and representative Cycle N for MNGP. This equilibrium cycle design analysis utilizing ATRIUM 11 fuel design is performed with the neutronics methodologies, EMF-2158(P)(A), and ANP-10335P-A. The nuclear data comprising of cross sections and local power peaking factors were generated using the CASMO-4 lattice depletion code. The MICROBURN-B2, Version 2, core simulator code is used to model the MNGP core for pin power distribution to determine thermal margins, **[[**

]].

Tables 2.1 and 2.2 of ANP-3881 list Cycle 15 equilibrium cycle energy and a key summary of results and assemblies for Cycle 15 nuclear fuel type, respectively. Tables 2.3 through 2.5 of ANP-3881P contain the assumed thermal limits for the equilibrium design. Figures 2.1 and 2.2 of ANP-3881 provide a summary of the Cycle 15 design step-through projection.

Figures 3.1 and 3.2, along with Table 3.1 of ANP-3881P, define the reference loading pattern used in the equilibrium Cycle 15. Appendix A of ANP-3881P provides control rod patterns and resultant key operating parameters including thermal margins from Cycle 15.

The Cycle 15 calculations demonstrate adequate hot excess reactivity, standby liquid control (SLC) shutdown margin, and cold shutdown margin throughout the cycle. The shutdown margin is in conformance with the Technical Specification limit of R + 0.38 percent Δk/k at beginning of cycle. Appendix B provides elevation view for the equilibrium cycle for each fuel assembly type. Appendix C provides representative equilibrium Cycle 15 radial exposures and power distributions.

The NRC staff reviewed the process of developing the equilibrium Cycle 15 and the results of projected control rod patterns with acceptable margin to thermal limits, adequateness of hot excess reactivity and cold shutdown margin throughout the cycle. The NRC staff has determined that the Cycle 15 equilibrium cycle has been designed to meet the energy requirements for the licensee and has been developed using NRC-approved procedures.

3.11 Technical Evaluation Conclusion

The NRC staff reviewed the analyses related to the proposed amendments to allow application of the methodologies necessary to support a MNGP planned transition to ATRIUM 11 fuel under the currently licensed EFW operating domain under EPU conditions. The NRC staff further reviewed the proposed changes to TS 5.6.3.b that support adoption of the intended analysis methodologies. Based on its review, as summarized in this SE, the NRC staff concludes that the proposed amendment to allow MNGP transition to ATRIUM 11 fuel is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment on December 1, 2022. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR, part 20, or changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (86 FR 55016, October 5, 2021) and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Xcel Energy, MNGP Nuclear Generating Plant, "License Amendment Request: Application to Adopt Advanced Framatome Methodologies," July 29, 2021, (ML21211A594).

2. Xcel Energy, MNGP Nuclear Generating Plant, "Response to a Request for Additional Information for the Monticello Nuclear Generating Plant Related to the Amendment to Adopt Advanced Framatome Methodologies (EPID: L-2021-LLA-0144)," June 6, 2022 (ML22157A427).

3. ANP-3924P, Revision 0 (Attachment 3c to the LAR), "Applicability of Framatome BWR Methods to Monticello with ATRIUM 11 Fuel," June 2021.

4. ANP-3933P, Revision 0 (Attachment 9c to the LAR) "Monticello ATWS-I Evaluation with ATRIUM 11 Fuel," June 2021.

5. ANP-3929P, Revision 0 (Attachment 10c to the LAR), “Monticello ATRIUM 11 Control Rod Drop Analyses with the AURORA-B CRDA Methodology,” June 2021.

6. ANP-3925P, Revision 0 (Attachment 11c to the LAR), “Monticello ATRIUM 11 Transient Demonstration,” July 2021.

7. ANP-3934P, Revision 0 (Attachment 12c to the LAR), “Monticello LOCA Analysis for ATRIUM 11 Fuel,” July 2021

8. ANP-3932P, Revision 0 (Attachment 13c to the LAR), “Application of BEO-III Methodology with Period-Based Detection Algorithm at Monticello,” June 2021.

9. ANP-3882P, Revision 0 (Attachment 4c to the LAR), “Mechanical Design of Monticello ATRIUM 11 Fuel Assemblies,” March 2021.

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