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

January 21, 2021

Mr. Kevin Cimorelli
Site Vice President
Susquehanna Nuclear, LLC
769 Salem Boulevard
NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 278 AND 260 TO ALLOW APPLICATION OF ADVANCED FRAMATOME ATRIUM 11 FUEL METHODOLOGIES (EPID L-2019-LLA-0153)

Dear Mr. Cimorelli:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 278 to Renewed Facility Operating License No. NPF-14 and Amendment No. 261 to Renewed Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications in response to your application dated July 15, 2019, as supplemented by letters dated February 6, 2020, and April 1, 2020.

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

Enclosure 4 to this letter contains proprietary information. When separated from Enclosure 4, this document is decontrolled.

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

Sincerely,

/RA/

Sujata Goetz, Project Manager
Plant Licensing Branch I
Division of Operating Reactor
Licensing Office of Nuclear Reactor
Regulation

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 278 to NPF-14
2. Amendment No. 260 to NPF-22
3. Safety Evaluation (Non-Proprietary)
4. Safety Evaluation (Proprietary)

cc without Enclosure 4: Listserv

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

SUSQUEHANNA NUCLEAR, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 278
Renewed License No. NPF-14

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

   A. The application for amendment filed by Susquehanna Nuclear, LLC, dated July 15, 2019, as supplemented by letters dated February 6, 2020, and April 1, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to loading ATRIUM 11 fuel into the core during the spring 2022 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James G. Danna, Chief
Plant Licensing Branch I
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 21, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 278

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Page 3	Page 3
Page 18	Page 18

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

REMOVE	INSERT
2.0-1	2.0-1
5.0-22	5.0-22
5.0-23	5.0-23

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

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

(5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth 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) Maximum Power Level

Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(36), 2.C.(37), 2.C.(38), and 2.C.(39) to this license shall be completed as specified.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 178 to Facility Operating License No. NPF-14, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 178. For SRs that existed prior to Amendment 178, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 178.

result of the test, the test failure shall be addressed in accordance with corrective action program requirements and the provisions of the power ascension test program prior to continued operation of the SSES Unit above 3489 MWt.

(b) Unless the NRC issues a letter notifying the licensee that the tests specified by License Condition 2.C.(37)(a) adequately demonstrate that a single condensate pump trip will not result in a loss of all feedwater while operating at the full CPPU power level of 3952 MWt, the operating licensee shall perform the transient test on either SSES unit (whichever unit is first to achieve the following specified operating conditions) specified by License Condition 2.C.(37)(a) during the power ascension test program while operating at 3872 MWt to 3952 (98% to 100% of the full CPPU power level) with feedwater and condensate flow rates stabilized. The test shall be performed within 90 days of operating at greater than 3733 MWt and within 336 hours of achieving a nominal power level of 3872 MWt with feedwater and condensate flow rates stabilized. The operating licensee will demonstrate through performance of transient testing on either Susquehanna Unit 1 or Unit 2 (whichever unit is first to achieve the specified conditions) that the loss of one condensate pump will not result in a complete loss of reactor feedwater. The operating licensee shall confirm that the plant response to the transient is as expected in accordance with the acceptance criteria that are established. If a loss of all feedwater occurs as a result of the test, the test failure shall be addressed in accordance with corrective action program requirements and the provisions of the power ascension test program prior to continued operation of either SSES Unit above 3733 MWt.

(38) <u>Neutronic Methods</u>

(a) Not Used

(b) Not Used

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 575 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.5 COLR (continued)

The approved analytical methods are described in the following documents, the approved version(s) of which are specified in the COLR.

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company.

2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet pump BWR Reload Fuel," Exxon Nuclear Company.

3. EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation.

4. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation.

5. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company.

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

7. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP.

8. EMF-2292(P)(A), "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation

9. Not used

10. Not used

11. Not used

12. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Advanced Nuclear Fuels Corporation.

13. EMF-2209(P)(A), "SPCB Critical Power Correlation," Siemens Power Corporation.

14. EMF-CC-074(P)(A), "BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation.

5.6 Reporting Requirements

5.6.5 COLR (continued)

15. Not used

16. NEDO-32465-A, "BWROG Reactor Core Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

17. BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors."

18. ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods."

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

20. ANP-10300P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios."

21. ANP-10332P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios."

22. ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)."

23. ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors."

24. BAW-10247P-A Supplement 2P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Supplement 2: Mechanical Methods."

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.

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

SUSQUEHANNA NUCLEAR, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 260
Renewed License No. NPF-22

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

   A. The application for amendment filed by Susquehanna Nuclear, LLC, dated July 15, 2019, as supplemented by letters dated February 6, 2020, and April 1, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 260, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to loading ATRIUM 11 fuel into the core during the spring 2021 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

James G. Danna, Chief
Plant Licensing Branch I
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 21, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 260

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Page 3	Page 3
Page 14	Page 14

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

REMOVE	INSERT
2.0-1	2.0-1
5.0-22	5.0-22
5.0-23	5.0-23

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

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

(5) Susquehanna Nuclear, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth 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>

Susquehanna Nuclear, LLC is authorized to operate the facility at reactor core power levels not in excess of 3952 megawatts thermal in accordance with the conditions specified herein. The preoperational tests, startup tests and other items identified in License Conditions 2.C.(20), 2.C.(21), 2.C.(22), and 2.C.(23) to this license shall be completed as specified.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 260, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. Susquehanna Nuclear, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 151 to Facility Operating License No. NPF-22, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 151. For SRs that existed prior to Amendment 151, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 151.

(22) Neutronic Methods

(a) Not Used

(b) Not Used

(23) Containment Operability for EPU

The operating licensee shall ensure that the CPPU containment analysis is consistent with the SSES 1 and 2 operating and emergency procedures. Prior to operation above CLTP, for each respective unit, the operating licensee shall notify the NRC project manager that all appropriate actions have been completed.

(24) Primary Containment Leakage Rate Testing Program

Those primary containment local leak rate program tests (Type B – leakage boundary and Type C - containment isolation valves) as modified by approved exemptions, required by 10 CFR Part 50, Appendix J, Option B and Technical Specification 5.5.12, are not required to be performed at the CPPU peak calculated containment internal pressure of 48.6 psig (Amendment No. 224 to this Operating License) until their next required performance.

(25) Critical Power Correlation Additive Constants

AREVA NP has submitted EMF-2209(P), Revision 2, Addendum 1 (ML081260442) for NRC review to correct the critical power correlation additive constants due to a prior Part 21 notification (ML072830334). The report is currently under NRC review.

The license shall apply additional margin to the cycle specific OLMCPR, consistent in magnitude with the non-conservatism reported in the Part 21 report, thus imposing the appropriate MCPR penalty on the OLMCPR. This compensatory measure is to be applied until the approved version of

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 575 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6.5 COLR (continued)

The approved analytical methods are described in the following documents, the approved version(s) of which are specified in the COLR.

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5. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company.

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

7. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP.

8. EMF-2292(P)(A), "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation.

9. Not used

10. Not used

11. Not used

12. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Advanced Nuclear Fuels Corporation.

13. EMF-2209(P)(A), "SPCB Critical Power Correlation," Siemens Power Corporation.

14. EMF-CC-074(P)(A), "BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation.

5.6 Reporting Requirements

5.6.5 COLR (continued)

15. Not used

16. NEDO-32465-A, "BWROG Reactor Core Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications."

17. BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors."

18. ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods."

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

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21. ANP-10332P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios."

22. ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)."

23. ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors."

24. BAW-10247P-A Supplement 2P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Supplement 2: Mechanical Methods."

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. 278 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 260 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-22

SUSQUEHANNA NUCLEAR, LLC

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

Proprietary information pursuant to Section 2.390 of Title 10 of
the *Code of Federal Regulations* has been redacted from this document.

Redacted information is identified by blank space enclosed within [[double brackets]]

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 278 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 260 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-22

SUSQUEHANNA NUCLEAR, LLC

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By letter dated July 15, 2019 [1], as supplemented by letters dated February 6, 2020 [2], and April 1, 2020 [3], Susquehanna Nuclear, LLC (the licensee) submitted a license amendment request (LAR) for Susquehanna Steam Electric Station (Susquehanna), Units 1 and 2, to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) operating domain.

The supplemental letters dated February 6, 2020, and April 1, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 22, 2019 (84 FR 56482).

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

2.0 REGULATORY EVALUATION

The NRC staff reviewed the LAR to evaluate the applicability of the Framatome methodologies to Susquehanna to confirm that the use of the methodologies is within the NRC-approved ranges necessary to support a planned transition to ATRIUM 11 fuel and to verify that the results of the analyses and methodologies are in compliance with the requirements of the following general design criteria (GDC) specified in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. In addition, the NRC staff assessed the aging degradation due to irradiation embrittlement in reactor pressure vessel (RPV) base metal and welds to verify compliance with the requirements of the following regulations. Each subsection of this safety evaluation (SE) includes a Regulatory Evaluation section specific to that portion of the review.

- GDC 4, "Environmental and dynamic effects design bases," requiring that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

- 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 RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other RPV 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.

- 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” provides fracture toughness requirements for ferritic materials in the RCPB, including requirements for the Charpy upper-shelf energy (USE) for protecting RPV beltline materials against non-brittle failure and requirements for calculating RCS pressure-temperature (P-T) limits for protection against brittle fracture. Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron radiation.

- 10 CFR 50.55a imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (BPV Code), Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water cooled nuclear power plants. The ASME BPV Code Section XI code of record for the fourth ISI interval at Susquehanna is the ASME BPV Code, Section XI, 2007 Edition through 2008 Addenda.

- 10 CFR 50.36(c) specifies the categories that are to be included in the TSs, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. In 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.

3.0 TECHNICAL EVALUATION

In the LAR, the licensee requested a revision to Susquehanna, Units 1 and 2, TS 5.6.5.b to allow application of Advanced Framatome Methodologies for determining core operating limits in support of loading Framatome fuel type ATRIUM 11. The revision would support the transition to ATRIUM 11 fuel in the approved operating domain at Susquehanna, which includes MELLLA conditions. The LAR also requested revisions to TSs 2.1.1.1 and 2.1.1.2 to revise the low-pressure safety limit and remove neutronic methods penalties on oscillation power range monitor (OPRM) amplitude setpoint, the pin power distribution uncertainty, and bundle power correlation coefficient.

A request to implement Technical Specifications Task Force (TSTF) Traveler TSTF-535, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs," was also included in this LAR. This change was reviewed and approved in Amendment Nos. 274 and 256 [4].

This SE includes a detailed review of the following areas of the LAR:

- applicability of Framatome boiling-water reactor (BWR) methods to Susquehanna with ATRIUM 11 fuel
- mechanical design of ATRIUM 11 fuel assemblies
- thermal-hydraulic design of ATRIUM 11 fuel assemblies
- ATRIUM 11 fuel rod thermal-hydraulic evaluation
- ATRIUM 11 transient demonstration
- loss-of-coolant accident (LOCA) analysis for ATRIUM 11 fuel
- Susquehanna ATRIUM 11 control rod drop accident (CRDA) analyses
- revision of low-pressure safety limit in TSs 2.1.1.1 and 2.1.1.2
- removal of neutronic methods penalties for OPRM amplitude setpoint and pin power distribution uncertainty and bundle power correlation coefficient
- aging degradation

The NRC staff reviewed the LAR in conjunction with the supplemental information and the responses to the NRC staff's requests for additional information (RAIs) [2], [3] to (1) evaluate the acceptability of the Susquehanna transition to Framatome ATRIUM 11 fuel, (2) evaluate the use of the associated Framatome methodologies for licensing applications, and (3) confirm the adequate technical basis for the proposed TS changes.

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

Applicability of Framatome BWR methods is addressed in the BWR compendium [5], which is referenced as part of ANP-3753P (Enclosure 8 to [1]). While the NRC staff did not review and approve this reference, the staff reviewed it for applicability to the use of ATRIUM 11 fuel at Susquehanna. Many of the methodologies discussed in the compendium have previously been confirmed to be applicable to ATRIUM 10 fuel at Susquehanna and apply to the use of ATRIUM 11 fuel because it is fundamentally an evolutionary fuel design with similar geometry and composition characteristics. When appropriate, the applicability of methodologies to specific safety analyses is addressed in the discussion later in this SE associated with that analysis. Three areas of interest are as follows:

1. ANP-3753P Section 5.4 is dedicated to safety limit minimum critical power ratio (MCPR), specifically related to the methodology to determine the TS limit to ensure that 99.9 percent of fuel rods avoid boiling transition during normal reactor operation and AOOs. The NRC staff's evaluation of this section is provided with the remaining safety limit MCPR evaluation in Section 3.5.2.1 (MCPR Fuel Cladding Integrity Safety Limit) of this SE.

2 ANP-3753P Section 6.4 is dedicated to CRDA, specifically related to the critical heat flux (CHF) correlation used for the CRDA calculations. The NRC staff's evaluation of this

section is provided with the remaining CRDA evaluation in Section 3.8 (Control Rod Drop Accident (CRDA)) of this SE.

3. ANP-3753P Section 7.0 is dedicated to stability, specifically related to how Susquehanna updated its Option III stability methods to the capture chromia-doped fuel properties in the ATRIUM 11 fuel design. The NRC staff's evaluation of this section is provided in Section 3.4 (Stability) of this SE.

3.2 ATRIUM 11 Fuel Assembly/Rod Design

3.2.1 Regulatory Basis

The ATRIUM 11 fuel (assembly/rod) design was developed using the thermal mechanical design bases and limits outlined in ANF-89-98(P)(A) [6], compliance with which ensures that the fuel design meets the fuel system damage, fuel failure, and fuel coolability criteria identified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) [7]. 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 Susquehanna 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 GDC 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.

3.2.2 Technical Evaluation

ANP-3762P (Enclosure 9a to the LAR [1]) provides the mechanical design details, fuel structural analysis results of the ATRIUM 11 fuel assemblies, and fuel channel designs, while ANP-3745P (Enclosure 11a to the LAR [1]) provides the design parameters and design evaluation results of the ATRIUM 11 fuel rods to be used at Susquehanna.

3.2.2.1 Summary of Mechanical Design of ATRIUM 11 Fuel Assemblies for Susquehanna

ANP-3762P (Enclosure 9a to the LAR) provides key fuel assembly design details for the Framatome ATRIUM 11 fuel assembly design planned for use at Susquehanna. **[[**

]] Table 2-1 of ANP-3762P 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-3762P.

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

To perform specific evaluations for the ATRIUM 11 fuel assembly mechanical design, the licensee utilized specific NRC-approved methodologies. NRC approval of these methodologies is conditional on adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the topical reports. A discussion of how these limitations and conditions are met for Susquehanna 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 Susquehanna for analysis of the ATRIUM 10 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-3762P, as discussed in Section 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) 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** [8]

  The NRC staff's SE for EMF-93-177-NP-A specified several limitations and conditions that have already been shown to be met at Susquehanna for the channels associated with the ATRIUM 10 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 [9] 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** [10]

  The ATRIUM 11 fuel mechanical design evaluation, as discussed in Section 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) 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.

3.2.2.3 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 [9], (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 Enclosure 11a to the LAR [1] is discussed in Section 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) of this SE.

<u>Stress, Strain, Loading, and Deformation Limits on Assembly Components</u>

The licensee used the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code* (BPV)) [11] 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 BPV 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 licensee's 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 **[[**

]]. 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 licensee's evaluation acceptable because the evaluation is complete and adequate to meet the required design criteria and satisfy the SRP objectives.

<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 **[[**

]] The licensee developed a **[[** **]]** described in BAW-10247P-A, Supplement 2P-A [10]. 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 [10], 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 Susquehanna Z4B is within the scope of the NRC approval of BAW-10247P-A, Supplement 2P-A. Furthermore, the NRC considered and accepted data for the ATRIUM 11 fuel assembly design as part of the basis 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. The licensee 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.

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-93-177P-A, Revision 1, Supplement 1P-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. **[[**

]]

<u>Summary of Sections 3.2.2.1 through 3.2.2.3</u>

Tables 3-1 and 3-2 of Enclosure 9a 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 Susquehanna reactor cores.

3.2.2.4 Summary of ATRIUM 11 Fuel Rod Thermal-Mechanical Design for Susquehanna

ANP-3745P (Enclosure 11a to the LAR) provides key fuel rod design details for Framatome ATRIUM 11 fuel planned for use at Susquehanna. 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-3745P lists the key fuel rod design parameters for the ATRIUM 11 fuel. Further descriptions of the fuel assembly components are provided in ANP-3745P.

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

To perform specific evaluations for the ATRIUM 11 fuel rod design, the licensee utilized specific NRC-approved methodologies. NRC approval of these methodologies is conditional on adequately meeting the limitations and conditions listed in the NRC staff's SE for each of the topical reports. A discussion of how these limitations and conditions are met for Susquehanna 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 Susquehanna for analysis of the ATRIUM 10 fuel rod design that may not automatically apply to the ATRIUM 11 fuel rod design.

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

ANF-89-98(P)(A) provides some generic fuel rod design criteria that were approved by the NRC for use with evaluation of Framatome fuel designs. The ATRIUM 11 fuel rod design as reported in ANP-3745P, as discussed in Section 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) of this SE) , describes how the design criteria presented in ANF-89-98(P)(A) apply to the ATRIUM 11 fuel rod design.

BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, dated February 2008 [13]

Section 3.2.2.6 (ATRIUM 11 Fuel Rod Design Evaluation) 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 [14] 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 [14]

The chromia-doped fuel properties and models described in this topical report 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. The NRC staff notes that the methodologies that will be used to evaluate the ATRIUM 11 fuel at Susquehanna are approved for maximum fuel rod burnups of up to 62 gigawatt days per metric ton of uranium (GWd/MTU).

3.2.2.6 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-89-98(P)(A), Revision 1, and Supplement 1 and the RODEX4 analysis methodology described in BAW-10247PA [10] and [13]. 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 10 design. The ATRIUM 11 fuel also utilizes two relatively new materials in its overall composition—the chromia additive in the fuel pellets and the Z4B alloy used for some of the structural elements. Additional details regarding the fuel rod design are provided in Section 3.1 of ANP-3745P (Enclosure 11a to the LAR). 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 [14]. Therefore, the RODEX4 code was used to evaluate the fuel rod thermal-mechanical performance of the ATRIUM 11 fuel rod design, as appropriate.

Table 2-1 of ANP-3745P provides a summary of the findings from the fuel rod design evaluations that demonstrates that the acceptance criteria are met. The key fuel rod design parameters used in the fuel rod design evaluations are provided in Table 3-1. Table 3-2 provides the specific results based on the equilibrium cycle for MELLLA conditions. The fuel rod analyses, such as those for fuel centerline temperature and cladding strain, cover normal operating conditions and AOOs. More detail on the NRC staff considerations in reviewing each acceptance criterion is provided below.

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-3745P, 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 licensee adjusted the melting point 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 linear heat generation rate (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 BPV Code [11]. This is consistent with NRC-approved mechanical design criteria and, therefore, is acceptable.

Fuel Densification and Swelling

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

Fatigue

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

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

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

Since the acceptance criterion is consistent with the NRC-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.

Oxidation, Hydriding, and Crud Buildup

The RODEX4 code and methodology are used to determine cladding external oxidation and its effect on the heat transfer coefficient from the cladding to the coolant. The acceptance criterion for oxidation is discussed within the NRC-approved RODEX4 fuel rod evaluation methodology, 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 Susquehanna is provided below.

- **[[**

  ]]

- BAW-10247PA [13] 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 Susquehanna for ATRIUM 10. The cladding properties for the ATRIUM 11 fuel assembly design are not different from the ATRIUM 10 fuel assembly design, so no change is expected as a result of transitioning to ATRIUM 11 fuel.

- **[[**

  ]]

The effects of oxidation, crud buildup, and hydriding are addressed through the use of the NRC-approved RODEX4 fuel rod evaluation methodology and its acceptance criteria, as appropriately applied to Susquehanna 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>Water Chemistry</u>

GDC 10 requires that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of AOOs. Oxidation and hydriding are two specified acceptable fuel design limits that ensure components maintain strength and ductility. Section 3.5.1 of ANP-3762P mentions that water chemistry is controlled to reduce oxidation in the fuel channel.

The licensee stated in its February 6, 2020, letter that the plant water chemistry will be operated in accordance with the Electric Power Research Institute (EPRI) BWR Water Chemistry Guidelines (BWRVIP-190). The key figures of merit for water chemistry are those defined as "needed" or "control" parameters in Chapter 2 of BWRVIP-190, Volume 1. The measurement frequencies and operating limits for these parameters are defined in the guidelines, as is the response timeline for any excursions. Any deviations from the guidelines requirements for "needed" or "control" parameters must be justified by the licensee and documented in the plant's strategic water chemistry plan. The NRC staff reviewed this response and found it acceptable because the industrial guideline is followed to ensure the satisfactory performance of ATRIUM 11 fuel and Z4B water channel, which complies with the GDC 10 requirement to maintain fuel integrity.

Summary of Sections 3.2.2.4 to 3.2.2.6

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 Susquehanna. 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 Susquehanna.

3.2.3 Conclusion of ATRIUM 11 Fuel Assembly/Rod Design

For evaluation of the ATRIUM 11 fuel assembly/rod design (Section 3.2 of this SE, (ATRIUM 11 Fuel Assembly/Rod Design), the NRC staff concludes that the application of ATRIUM 11 fuel (fuel assembly and fuel rod) to Susquehanna is acceptable because it complies with the requirements of GDC 10, 27, and 35. This conclusion is based on the following:

1. The application meets the requirements of GDC 10 with respect to the 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 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) and 3.2.2.6 (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 topical reports as evaluated in Sections 3.2.2.2 (Applicability of Methodologies for Analysis of ATRIUM 11 Fuel Assembly Mechanical) Design and 3.2.2.5 (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 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) for Susquehanna) 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 3.2.2.6 (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.

3.3 Thermal-Hydraulic Design of ATRIUM 11 Fuel Assemblies

3.3.1 Regulatory Basis

The ATRIUM 11 fuel design was developed using the thermal-mechanical design bases and limits as outlined in ANF-89-98(P)(A), compliance 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 Susquehanna 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 GDC 10, 12, 15, 20, 25, 26, 27, 28, and 35 (see the following sections below for further discussion).

3.3.2 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 10 fuel at Susquehanna. The licensee is proposing to transition from the current ATRIUM 10 fuel design to ATRIUM 11 fuel. Enclosure 10a to the LAR [1] provides the results of the thermal-hydraulic analyses to support a finding that ATRIUM 11 fuel is hydraulically compatible with the co-resident ATRIUM 10 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 [15]. Susquehanna, Units 1 and 2, have the same core power, flow, geometries, and bundle geometries. Both units operate on a 24-month fuel cycle resulting in minimal differences in fuel and core neutronic design. Based on the minimal differences between Units 1 and 2, the information that is included in the submittal is provided for Unit 2 Cycle 21 – limited information needs to be provided for Unit 1. Therefore, the licensee will include the Unit 1 Cycle 23 reload safety analysis report with transmittal of the combined operating limits report prior to startup from the Unit 1 Cycle 23 refueling outage (i.e., spring 2022), which will load the first reload batch of ATRIUM 11 fuel into the Unit 1 reactor core.

The licensee performed thermal-hydraulic analyses 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-3761 lists the applicable thermal-hydraulic design criteria, analyses, and results for hydraulic compatibility, thermal margin performance, fuel centerline temperature, rod bow, bypass flow, stability, LOCA analysis, CRDA analysis, ASME over-pressurization analysis, and seismic/LOCA liftoff. The subsections below summarize the results from selected design criteria and analyses results.

<u>Hydraulic Characterization</u>

Basic dimension parameters for the ATRIUM 10 and ATRIUM 11 fuel assembly designs are summarized in Table 3.2 of ANP-3761. 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-3761.

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 10 fuel assembly design, as reviewed by the NRC for applicability to other plants operating in the MELLLA 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 10 fuel assembly design. Therefore, the NRC staff finds the hydraulic characterization of the ATRIUM 11 fuel assembly design to be acceptable.

<u>Thermal-Hydraulic Compatibility</u>

The thermal-hydraulic compatibility analyses were performed in accordance with the Framatome thermal-hydraulic methodology for BWRs [15]. 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 10 and ATRIUM 11 fuel assembly designs that are planned to be inserted in the Susquehanna core. The analyses were performed for full-core and mixed-core configurations.

In essence, the hydraulic compatibility analysis **[[**

]] This analysis is performed utilizing different typical axial power shapes and radial power factors for rated and off-rated conditions. The input conditions used for the analysis are listed in Table 3.4 of ANP-3761, 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 10 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.

Based on the changes in **[[** **]]** caused by the transition from ATRIUM 10 fuel to ATRIUM 11 fuel, the NRC staff finds that the hydraulic compatibility analyses for the transition cores at Susquehanna, Units 1 and 2, provide reasonable assurance that the resident and co-resident fuel designs will satisfy the thermal-hydraulic design criteria for mixed cores.

<u>Thermal Margin Performance</u>

The thermal margin analyses were performed using the NRC-approved thermal-hydraulic methodology for steady-state critical power ratio (CPR) evaluations with XCOBRA listed in the Susquehanna TSs. Empirical correlations for the ATRIUM 10 [16] and ATRIUM 11 [17] 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 10 fuel.

<u>Rod Bow</u>

Rod bow is addressed as part of the mechanical design analyses see Section 3.2.2.3 (Fuel Assembly Mechanical Design Evaluation) of this SE for further discussion). **[[**

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The NRC staff finds this disposition to be acceptable based on the fact that this 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-89-98(P)(A) include a requirement to confirm that the stability characteristics for a new fuel design are equivalent to or better than that of prior approved fuel designs. This evaluation is performed using the STAIF code as prescribed in ANF-89-98(P)(A), and the results are documented in ANP-3761 for Susquehanna. 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 confirmation density algorithm-based hardware trip is expected to detect and suppress any power oscillations resulting from stability issues, as confirmed through the use of the Option 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 would ensure that the regulatory requirements associated with stability performance are met.

Void Fraction

Section 5.1 of ANP-3753P discusses the use of the **[[** **]]** correlation for ATRIUM 11 fuel. The NRC staff questioned **[[** **]]** Based on discussions during an audit during the review of the Brunswick Steam Electric Plant (Brunswick) fuel transition to ATRIUM 11, it was clarified that the **[[** **]]** This evidence, **[[** **]]**, and the approach is acceptable.

3.3.3 Thermal-Hydraulic 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 10 fuel currently used at Susquehanna. 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 10 and ATRIUM 11 fuel assemblies contain a number of differences in their geometric and hydraulic characteristics, they remain hydraulically compatible.

3.4 Stability

Stability methodology at Susquehanna is described in Section 7 of ANP-3753P. Stability analyses at Susquehanna are performed using the approved Option III stability methodology in the RAMONA5-FA [18], which was approved before the implementation of chromia-doped fuel. Methods in this stability solution were updated to account for the use of ATRIUM 11 fuel rod property models. Both Susquehanna units continue to implement stability Option III.

3.4.1 Regulatory Basis

The plant-specific Option III long-term stability solution and related licensing basis were developed to comply with the requirements of GDC 10 and 12.

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

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

Consistent with GDC 10 and 12, the NRC staff determines whether the licensee performs the plant-specific trip setpoint calculations for long-term stability using acceptable methodologies as prescribed in the SRP (NUREG-0800), Sections 4.4 and 15.9.

3.4.2 Technical Evaluation

The RAMONA5-FA [18] and STAIF [19] methods used in the Option III methodology have been updated to address advanced fuel design features of ATRIUM 11 using **[[ ]].** The fuel property models implemented are the same models used in the Framatome generic anticipated transient without scram (ATWS)-I methodology described in ANP-10346NP-A [20]. Susquehanna is only implementing the fuel rod property models from the Framatome generic ATWS-I methodology. While the licensee references the topical report for the Framatome generic ATWS-I methodology, it does not intend to adopt the methodology in its entirety, but only adopt the fuel rod property models for chromia-doped fuel. While the fuel rod property models are included within the description of the ATWS-I methodology, they are not a specific feature of the ATWS-I methodology itself. Rather, this topical report was the first opportunity for Framatome to document the implementation of chromia-doped fuel properties and models within the RAMONA5-FA code for NRC review and approval. The NRC staff's evaluation of the implementation of these models, as applicable to the intended application for Susquehanna, is provided in the following sections.

3.4.2.1 **[[ ]]** Fuel Rod Models

ANP-3753P describes that in the Option III methodology at Susquehanna, **[[**

]] The licensee accounted for the effects of chromia doping in fuel pellets by modifying the standard UO_2 thermal conductivity and **[[ ]]** models. The material properties, pellet-clad gap heat transfer coefficient, and radial power distribution in fuel pellets used in the Option III methodology at Susquehanna are identical to that used in the generic ATWS-I methodology. Although **[[**
]], a plant-specific evaluation of these areas is provided below.

3.4.2.1.1 Material Properties

The ANP-10346P-A methodology uses fuel pellet and cladding thermophysical properties based on **[[ ]].** The NRC staff finds this approach acceptable for use in the RAMONA5-FA ATWS-I calculations at Susquehanna because these models account for all important fuel characteristics relevant to ATWS-I, including the **[[**

]].
Appendix A to Duke Energy, ANP-3782P, Revision 1, "Brunswick ATRIUM 11 Advanced Methods Response to Request for Additional Information," dated May 29, 2019 (ADAMS Accession No. ML19149A320 (Non-Public)). Reference [22] includes an update to ANP-10346P-A that, among other changes, appends Appendix D, which presents modified fuel rod models that account for chromia doping of the UO_2 fuel pellets. The fuel thermal conductivity model was adapted from the approved RODEX4 model in Reference [14]. The **[[ ]]** model was developed by benchmarking to the approved RODEX4 model. The NRC staff finds these models acceptable for use in characterizing chromia-doped fuel properties for ATWS-I analyses at Susquehanna because these models are based on previously reviewed and approved models for chromia-doped fuel using the methodology described in ANP-10346P-A.

3.4.2.1.2 Pellet Clad Gap Heat Transfer Coefficient

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

3.4.2.1.3 Radial Power Deposition Distributions in Fuel Pellets

The NRC staff has reviewed the methodology and determined that it provides the needed accuracy for calculating the radial power distribution in fuel pellets, including

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]] Therefore, the NRC staff finds the radial power distribution methodology to be acceptable.

3.4.2.2 STAIF Reactor Benchmarks Using New Fuel Rod Property Models

The licensee reanalyzed all reactor benchmarks in the STAIF benchmarking suite (Section 4.0 of Reference [19]) using the new fuel rod property models evaluated in Section 3.4.2.1 (**[[** **]]** Fuel Rod Models) of this SE. The NRC staff compared decay ratios calculated with the new fuel rod property models to the measured decay ratios from various stability tests. **[[**

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3.4.2.3 RAMONA5-FA Reactor Benchmarks Using New Fuel Rod Property Models

The licensee reanalyzed all reactor benchmarks in the RAMONA5-FA benchmarking suite (Section 5.0 of Reference [19]) using the new fuel rod property models evaluated in Section 3.4.2.1 (**[[** **]]** Fuel Rod Models) of this SE. The predicted growth ratios and frequencies using the RAMONA5-FA with RODEX4 based fuel property models were compared to the results using the original fuel rod property models for each benchmark. **[[**

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3.4.3 Stability Conclusion

Based upon its review, the NRC staff determined that the Option III calculation procedure provides an acceptable means of determining licensing basis safety limit for minimum critical power ratio (SLMCPR) protection during anticipated stability events at Susquehanna.

3.5 ATRIUM 11 Transient Demonstration

3.5.1 Regulatory Basis

In addition to the GDC described in Section 2.0 of this SE, the following regulatory requirement applies to the AOO/ATWS evaluation.

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which requires licensees to provide the means to address an ATWS, which 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 20.3.5.2 Technical Evaluation

3.5.2.1 MCPR Fuel Cladding Integrity Safety Limit

Section 5.4 of ANP-3753P describes the SLMCPR methodology at Susquehanna. The ANP-10307PA, Revision 0 methodology used at Susquehanna is [23] is used to determine that

99.9 percent of fuel rods are expected to avoid boiling transition during normal reactor operation and AOOs. Of note is a plant-specific extension to the approved methodology. After reviewing the licensee's RAI response, the NRC staff concluded that the plant-specific extension is acceptable because the licensee has an appropriate process in place if the error bounds are exceeded.

3.5.2.2 AOOs

The licensee submitted information to demonstrate the applicability of the AURORA-B AOO methodology for Susquehanna, compliance with the NRC limitations and conditions imposed for application of the AURORA-B AOO topical report, and a demonstration analysis of select licensing basis events using the AURORA-B AOO methodology to demonstrate that the results of the analyses meet the applicable acceptance criteria. This information is found in the ANP-3753P and ANP-3783P attachments to the LAR [1] in conjunction with the licensee's responses to the NRC staff's RAIs [2], [3].

3.5.2.2.1 AURORA-B AOO Methodology Overview

The AURORA-B AOO methodology and the NRC staff's SE of the methodology is found in ANP-10300NP-A, Revision 1 [24]. The methodology is used to evaluation transients, postulated accidents, and beyond design-basis scenarios for BWRs. The methodology is built upon three computers codes:

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

The methodology uses non-parametric order statistics to evaluate the impact of uncertainties in the methodology. This means that for each scenario analyzed, several runs are executed (e.g., 59 runs), varying certain parameters to achieve a result at a certain confidence level. In the case of the AURORA-B AOO methodology, the uncertainty analysis is used to bound the 95 percent worst case result at 95 percent confidence. Table 3.6 of the SE for the AURORA-B AOO methodology contains the uncertainty parameters used for the uncertainty analysis.

The licensee provided a demonstration analysis in the ANP-3783P attachment to the LAR. The demonstration analysis provides analyses for the following transients, accidents, and beyond design-basis events:

- load rejection without bypass/turbine trip without bypass;
- feedwater controller failure;
- inadvertent startup of the high-pressure coolant injection (HPCI) pump;
- ASME over-pressurization analysis; and
- ATWS over-pressurization analysis.

3.5.2.2.2 Applicability of the AURORA-B AOO Methodology to Susquehanna

The NRC staff reviewed the LAR to ensure that the AURORA-B AOO methodology was applicable to Susquehanna. As described in Section 3.1 (Applicability of Framatome BWR

Methods to Susquehanna with ATRIUM 11 Fuel) of the SE for the AURORA-B AOO methodology [24], the methodology is applicable, in part, to BWR/3 through BWR/6 plants. Since Susquehanna is a BWR/4 plant, the methodology is applicable to Susquehanna in this respect. The NRC staff considered three additional major considerations to determine the applicability of the methodology to Susquehanna: (1) the applicability for use with ATRIUM 10 fuel; (2) the applicability for use with ATRIUM 11 fuel; and (3) the applicability for use in the MELLLA operating domain.

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

As described in Limitations 4 and 5 in Section 5.0 of the SE for the AURORA-B AOO methodology, a licensee is required to justify new fuel designs relative to those approved for use in the AURORA-B AOO methodology. ATRIUM 11 is a new fuel design for use with the AURORA-B AOO methodology. The licensee provided justification in the ANP-3753P and ANP-3783P attachments to the LAR. Specifically, the licensee provided justification for ATRIUM 11 with respect to transients and accidents in Section 4.0 of ANP-3783P and ATWS in Section 8.3 of ANP-3753P. The major concern for the transients and accidents is how the void-quality correlation uncertainties are incorporated into the analyses for transients and accidents. These uncertainties are important because they could impact the results of the analyses (e.g., MCPR). The NRC staff notes that it is also important for the licensee to use a void-quality correlation that is applicable to the fuel it is using. For Susquehanna, the licensee stated that it will be using the **[[ ]]** void correlation for the ATRIUM 11 fuel.

As described in the LAR, the licensee stated that these uncertainties were not explicitly included in the transient and accident analyses. Rather, they are implicitly included in the power prediction, and the uncertainties in the power prediction are included in the analysis to determine the SLMCPR. Susquehanna uses the SAFLIM3D methodology [25]. The NRC staff confirmed that the power prediction was incorporated into the SAFLIM3D methodology. Additionally, the NRC staff confirmed that the Susquehanna methodology used to calculate the power prediction, MICROBURN-B2 [26], incorporated the void-quality correlation. Since the licensee incorporates the void-quality uncertainty in the power prediction, and the power prediction uncertainty is included in the calculation of the SLMCPR, the NRC staff determined that the licensee appropriately addressed the ATRIUM 11 fuel for SLMCPR.

The LAR describes that the void-quality correlation uncertainty is incorporated into the delta critical power ratio (ΔCPR) as a result of a transient that is used to determine the operating limit minimum critical power ratio (OLMCPR).[1] The licensee also discussed how the void-quality correlation uncertainty is implicitly accounted for by conservatism in the computer code models and input parameters used for the analysis. The conservatism in the computer codes exist because they are tuned to bound the power increases relative to the benchmark tests. The uncertainty in the void-quality correlation uncertainty will impact the uncertainty in the power prediction (which has a direct influence on ΔCPR). Since the computer codes are tuned to bound the power predictions in the benchmark tests, they will inherently incorporate the void-quality correlation uncertainty. The licensee also stated that the input parameters for the transient analysis are biased to, in part, account for void-quality correlation uncertainty. Since the void-quality correlation is inherently accounted for in the transient analysis to determine ΔCPR, and the initial conditions are conservatively biased, the NRC staff determined that the licensee has adequately addressed the ATRIUM 11 fuel for ΔCPR.

The licensee intends to use the AURORA-B AOO methodology to analyze ATWS events except for ATWS-I. ATWS analysis is an approved analysis in the AURORA-B AOO methodology. In Section 8.1 of ANP-3753P, the licensee justified that the ATWS vessel over-pressurization event in the AURORA-B AOO code suite is not impacted by the ACE/ATRIUM 11 critical power correlation that was approved for ATRIUM 11 fuel. The justification provided is that the AURORA-B AOO methodology ignores dryout (and, therefore, does not need to use a critical power correlation) in the ATWS vessel over-pressurization event because it is more conservative to assume maximum heat transfer to the coolant for an overpressure event. The NRC staff determined that this justification is reasonable because maximizing heat transfer to the coolant will increase the pressure in the vessel, which is appropriate for analyzing an overpressure event. The NRC staff also determined that ignoring the dryout in the fuel is conservative because once the fuel is in dryout, heat transfer from the rod to the coolant is diminished, and heat transfer to the coolant would, therefore, be reduced.

The licensee also discussed the void-quality correlation's impact on the ATWS vessel overpressure analysis. Like the transient and accident discussion above, the licensee provided justification that the void-quality correlation uncertainties are inherently incorporated into the code, and that the input parameters are conservatively biased to account for uncertainties. Therefore, the NRC staff determined that the void-quality correlation uncertainties are appropriately accounted for in the ATWS methodology. The NRC staff notes that for ATWS analyses, the void-quality correlation is more important for predicting peak vessel pressure. For Susquehanna, the licensee stated that it will be using the void-quality correlation found in the ATRIUM 11 fuel.

Section 8.3 of ANP-3753P contains an evaluation of the ATWS containment heatup calculation. The licensee provided justification that **[[**
]]. The ATWS containment analysis is addressed in Section 3.5.2.2.5 (ATWS Containment Heatup) of this SE.

3.5.2.2.3 AURORA-B Methodology Limitations and Conditions

The AURORA-B AOO methodology contains 26 limitations and conditions in Section 5.0 of the NRC staff's SE (ANP-10300P-A, Revision 1). The licensee stated in the LAR that the limitations and conditions for the Framatome topical reports are included in ANP-2637P, "Boiling Water

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

Reactor Licensing Methodology Compendium," and compliance with the limitations and conditions is assured by implementing them within the engineering guidelines or by incorporating them into the computer codes. Discussion of the limitations and conditions for the AURORA-B AOO methodology is found starting on page 5-32 of ANP-2637P.

The NRC staff notes that Limitations and Conditions 20 through 26 in Section 5.2 of the SE for the AURORA-B AOO methodology are related to the change process of the methodology itself. The licensee requested AURORA-B AOO methodology as approved; therefore, these limitations and conditions are not applicable to the LAR.

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

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

Limitation and Condition 3 relates to using the approved uncertainty distributions in the analysis. The NRC staff confirmed that the generic uncertainty distributions presented in Table 3.2 of ANP-3783P are consistent with those in Table 3.6 of the SE for the AURORA-B methodology. For the **[[ ]]**, the licensee stated that the range was developed based on the approved process in Section 3.6.4.10 of the methodology. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 4 relates to the justification of void fraction prediction for new fuel designs. The licensee discussed the void fraction prediction in Section 6.1 of ANP-3753P. The NRC staff reviewed the void fraction prediction in Section 3.3.2 (Technical Evaluation) of this SE and found that it was acceptable. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 5 relates to the justification of the **[[ ]]** void-quality correlation for new fuel designs. The licensee discussed the void-quality correlation in Section 5.1 of ANP-3753P. The NRC staff reviewed this in Section 3.3.2 (Technical Evaluation) of this SE and found that it was acceptable. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

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

Limitation and Condition 7 relates to the licensee providing justification for the key plant parameters and initial conditions selected for performing sensitivity analyses on an event-specific basis. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the reload safety analysis report (RSAR) when it is submitted in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 8 relates to the truncation of sampling ranges for uncertainty distributions used in the non-parametric order statistics analyses. The licensee discussed in Section 3.2.2 of ANP-3783P how the sampling performed complies with the limitations and conditions of the SE for the AURORA-B methodology. The NRC staff confirmed that the licensee adequately addressed this limitation and condition.

Limitation and Condition 9 relates to uncertainties of medium or highly ranked phenomena identification and ranking table (PIRT) phenomena that are not addressed in given non-parametric order statistics analysis via sampling. To meet this limitation, the licensee modeled the phenomena as described in Tables 3.2 and 3.4 of the SE for the AURORA-B methodology. The NRC staff confirmed that the licensee complied with the requirements of the tables and, therefore, has adequately addressed this limitation and condition.

Limitation and Condition 10 relates to the assumptions of **[[** **]].** The licensee stated that it complied with the requirements of Tables 3.2 and 3.4 of the SE for ANP-10300P-A, Revision 1 [27], as they relate to this limitation. The NRC staff confirmed that the licensee complied with the requirements of the tables and, therefore, has adequately addressed this limitation and condition.

Limitation and Condition 11 relates to justification for uncertainties used for highly ranked plant-specific PIRT parameters. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the RSAR when it is submitted in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 12 relates to plant-specific changes to AURORA-B to enhance **[[** **]]** when applying the AURORA-B EM to the **[[** **]]**. For Susquehanna, the Inadvertent HPCI event is identified as potentially limiting (see response to RAI 2.1.a). A method to evaluate the mixing was proposed in Section 6.3 of the Methods Applicability Document (ANP-3753P) to be evaluated using **[[** **]]** Once the amount of mixing has been determined, the AURORA-B licensing model will be constructed. In order to ensure a conservative estimation of mixing is used, **[[** **]]** The licensee described how compliance with the requirement will be completed in the Reload Safety Analysis Report, which will be submitted following approval. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

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

Limitation and Condition 14 relates to the scope of the NRC's approval for AURORA-B. Specifically, the approval does not include the advanced BWR design. Since Susquehanna is not an advanced BWR, its use is within the scope. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Limitation and Condition 15 relates to the application of AURORA-B to BWR/2s at extended power uprate or extended flow window conditions. Susquehanna is not a BWR/2; therefore, this limitation and condition is not applicable.

Limitation and Condition 16 relates to the justification of a plant-specific conservative flow rate. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the RSAR when it is submitted in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

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

Limitation and Condition 18 relates to using conservative measures with the justification for the method of determining and applying conservative measures in future deterministic analyses for each figure of merit and re-performance of full statistical analysis if a scenario exceeds a 1σ magnitude difference. In RAI response 2.3 [3], the licensee described how compliance with this requirement will be completed in the RSAR when it is submitted in November 2020. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

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

The NRC staff reviewed each limitation and condition and finds that each was adequately addressed by the licensee for the demonstration case and will be supported by the RSAR when it is submitted in November 2020.

3.5.2.2.4 AURORA-B Methodology Analysis Results

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

Since the licensee's analysis in the LAR is a demonstration analysis, the NRC staff's review is to ensure that the licensee can adequately evaluate AOOs with the AURORA-B AOO methodology and ATRIUM 11 fuel. The NRC staff reviewed this analysis to ensure that the potentially limiting events are identified and considered for explicit analysis, the AOO results are realistic, and the results meet specified acceptable fuel design limits.

In the LAR, the licensee provided demonstration analyses for the load rejection without bypass event/turbine trip without bypass event, feedwater controller failure event, and inadvertent startup of the HPCI pump event.

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

]]

To ensure that there is appropriate coverage of the parameters used in the uncertainty analysis and to ensure that there are no significant trends with respect to the uncertainty parameters in the results, the NRC staff requested additional information in RAI 2.2. Specifically, the NRC staff requested to review the following data sets for the load rejection no bypass/turbine trip without bypass event at 100 percent power/108 percent flow, main steam isolation valve closure ATWS event at 100 percent power and 99 percent flow, and high-pressure coolant injection event at 100 percent power/108 percent flow:

- the sampled values of the uncertainty parameters for all cases executed and
- the figure of merit results for all cases executed.

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

3.5.2.2.5 ATWS Containment Heatup

Section 8.3 in ANP-3753P provides the licensee's evaluation of ATWS containment heatup. The NRC staff's evaluation of this section follows.

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

[[

]]

Additionally, the NRC staff requested information in RAI 1.b regarding the analysis performed to confirm that the fuel transition is bounded by the current analysis of record and the quantitative results for containment pressure and suppression pool temperature response. In its response, the licensee states, "the current licensing basis for Susquehanna ATWS containment shows the peak suppression pool temperature for MELLLA was 206 °F [degrees Fahrenheit] and the peak containment pressure was 16.1 psig [pounds per square inch gauge]…." The analysis is based on **[[** **]]** After this was completed, the licensee determined that the **[[**

]]

Finally, because containment heatup is directly impacted by the stored energy in the fuel and decay heat, a quantitative comparison of the decay heat between Framatome fuel types was reviewed [28]. The study compared **[[**

]]

Based on the above, the NRC staff determined that the analysis of record remains bounding for ATWS containment heatup with the transition to ATRIUM 11 at Susquehanna. Therefore, the NRC staff concludes that the applicable regulatory requirements continue to be met.

3.5.3 Application of Framatome Methodologies for Mixed Cores

Appendix A of ANP-3753P discusses the application of Framatome methodologies to mixed cores.

3.5.4 Transient Demonstration Conclusion

Regarding AOO and ATWS, the NRC staff reviewed the information in the licensee's submittals pertaining to the analysis of AOO and ATWS events for Susquehanna, including the original submittal as well as relevant responses to RAIs [2], [3]. Based upon its review, as summarized above, the NRC staff concludes that:

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

- Compliance with the applicable regulatory requirements has been demonstrated.

3.6 ATWS-1

3.6.1 Regulatory Basis

In addition to the GDC described in Section 2.0 (Regulatory Evaluation) of this SE, the following regulatory requirements apply to the ATWS-I evaluation.

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which requires 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 the 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.

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

The SRP (NUREG-0800) is the primary regulatory guidance document used by the NRC staff to support its review of this LAR. In particular, SRP Section 15.8, "Anticipated Transients Without Scram", establishes acceptance criteria for ATWS events. Although SRP Section 15.8 includes additional GDC beyond those listed above, they define vessel, ECCS, and containment performance requirements. These are not a significant concern for ATWS-I events; therefore, these GDC were not considered as part of this review.

3.6.2 Technical Evaluation

The NRC staff noted that a plant-specific ATWS-I analysis was not included in the submittal. As referenced in the licensee's UFSAR [29], the analysis used by the licensee is found in NEDO-32047-A [30]. NEDO-32047 has been reviewed and approved by the NRC staff for generic use when the assumptions within the ATWS-I analyses are bounding. Since Susquehanna has not yet elected to operate in an extended flow window, the assumptions of the generic analyses remain bounding.

The thermal-hydraulic fuel properties of ATRIUM 11 fuel do not affect the ATWS-I results since they are demonstrated to be more stable than the historical fuel product lines used in the generic analyses (see Section 3.6 of ANP-3761).

3.6.3 Conclusion

Based upon its review, the NRC staff determined that the generic ATWS-I analyses found in Susquehanna NEDO-32047-A are an acceptable means of determining protection during instability events at Susquehanna.

3.7 LOCA Analysis for ATRIUM 11 Fuel

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.

To support the planned transition to ATRIUM 11 fuel at Susquehanna, the licensee analyzed the spectrum of LOCA events for this fuel design using the AURORA-B LOCA evaluation model [31]. The AURORA-B LOCA evaluation model is an Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 conformant analysis methodology that was approved by the NRC in March 2019.

As described in the evaluation below, the NRC staff reviewed the licensee's implementation of the AURORA-B LOCA evaluation model for Susquehanna to ensure compliance with applicable regulatory requirements. The NRC staff's review activities associated with the LOCA analysis for Susquehanna focused upon the review of pertinent sections of the licensee's submittals (particularly ANP-3784P). The NRC staff further conducted a regulatory audit on November 15, 2019 [32], which supported its review of the information.

3.7.1 Applicable Regulatory Requirement

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

- 10 CFR 50.46;
- Appendix K to 10 CFR Part 50; and
- Appendix A to 10 CFR Part 50, GDC 35.

3.7.1.1 10 CFR 50.46

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

- Each boiling or pressurized light-water reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding[2] must perform analysis of core cooling performance under postulated LOCA conditions using an acceptable evaluation model.

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

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

Furthermore, 10 CFR 50.46(b) provides acceptance criteria for analyses of the spectrum of LOCA events, which are summarized below.

Table 1: 10 CFR 50.46 Acceptance Criteria

Subparagraph	Figure of Merit	Acceptance Criterion
(b)(1)	Peak Cladding Temperature	≤ 2,200 °F
(b)(2)	Maximum (Local) Cladding Oxidation	≤ 17% of Unoxidized Thickness
(b)(3)	Maximum (Core-Wide) Hydrogen Generation	≤ 1% of Hypothetical Amount
(b)(4)	Core Geometry	Amenable to Cooling
(b)(5)	Long-Term Cooling	Maintained

[2] Note that the applicability conditions stated in this requirement would be satisfied for both proposed ATRIUM 11 and co-resident ATRIUM 10 fuel designs loaded at Susquehanna.

In accordance with Limitation and Condition 4 from the NRC staff's final SE on ANP-10332P [31], 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 topical report NEDO-20566A [33]. 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.

3.7.1.2 Appendix K to 10 CFR Part 50

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 Susquehanna 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 [31].

3.7.1.3 Appendix A to 10 CFR Part 50, GDC 35

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

3.7.2 Acceptability of LOCA Evaluation Model

The licensee analyzed the spectrum of postulated LOCA events to verify the satisfaction of applicable regulatory requirements following the transition to ATRIUM 11 fuel. The licensee used the AURORA-B LOCA evaluation model [31] to demonstrate compliance with the four acceptance criteria from 10 CFR 50.46 that apply to the short-term LOCA analysis (i.e., subparagraphs (b)(1) through (b)(4) in Table 1).

The AURORA-B LOCA evaluation model is an S-RELAP5 based methodology that incorporates a kernel of transient fuel rod thermal-mechanical subroutines from the RODEX4 code. As documented in an SE dated March 26, 2019 [31], the NRC staff found the AURORA-B LOCA evaluation model acceptable for application to LOCA analysis for BWR/3-BWR/6 plants. Susquehanna, Units 1 and 2, are General Electric BWR/4 plants.

While the generic evaluation model proposed by the licensee to support its proposed fuel transition has been previously found to be acceptable], the NRC staff reviews licensee implementation of analytical evaluation models to ensure:

- Confirmation of acceptable plant-specific inputs to the evaluation model (Section 3.7.3.1 of this SE);
- Confirmation of adherence to the approved evaluation model (Sections 3.7.3.2 and 3.7.3.3);
- Confirmation that results calculated using the evaluation model satisfy regulatory acceptance criteria and otherwise conform to expectations (Section 3.7.4); and
- Verification of acceptable responses to limitations and conditions specified in the NRC staff's SE (Section 3.7.5).

The following sections of this SE describe the NRC staff's review of these areas.

3.7.3 Evaluation Model Implementation

3.7.3.1 Plant-Specific Inputs

Some design differences may exist between Susquehanna, Units 1 and 2, that will affect the LOCA analysis. During an audit conducted on November 15, 2019, the NRC staff confirmed that the principal plant parameter input to the LOCA analysis is not changed between both units.

The NRC staff also confirmed during the audit that 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 10 design. The licensee stated that the thermal-hydraulic compatibility analysis demonstrates that the thermal-hydraulic characteristics of the ATRIUM 11 and the coexistent ATRIUM 10 fuel are similar so that the core responses during LOCA will be insignificant for transition cores. The licensee further stated that the LOCA analysis **[[**

]].

The NRC staff found the licensee's response acceptable because the licensee provided adequate evidence that the impacts of transition cycles containing co-resident ATRIUM 10 fuel on the LOCA evaluation were small and within the conservative bounds established by the existing analysis so that the evaluation results meet the required design criteria.

3.7.3.2 Break Spectrum Implementation

The NRC staff's review found that the break spectrum analysis described in ANP-3784P generally conforms to the approved evaluation model documented in ANP-10332P-A [34].

The analysis for Susquehanna considered a spectrum of postulated double-ended guillotine and split breaks in the recirculation system (i.e., upper and lower suction piping, discharge piping).

Table 5.1 of ANP-3784P identifies the single failures considered in the Susquehanna LOCA analysis. The break spectrum analysis for Susquehanna focused upon four potentially limiting single failures among six potential limiting single failures identified in the UFSAR: (1) the failure of one train of direct current power (i.e., single failure (SF) of battery (direct current) power (SF-backup battery power (BATT)); (2) the failure of an automatic depressurization system valve (i.e., SF-automatic depressurization system valve (ADS)); (3) the failure of an opposite unit false LOCA signal (i.e., SF-LOCA); and (4) the failure of a low-pressure coolant injection system injection valve (i.e., SF-low-pressure coolant injection (LPCI)). The licensee determined that the ECCS resources for SF-diesel generator (DGEN) will be equal to or greater than that for SF-LOCA and that the ECCS resources for SF-HPCI (single failure of HPCI) will be equal to or greater than that for SF-BATT, such that the analyses for SF-DGEN and SF-HPCI are not considered because they can be bounded by SF-LOCA and SF-BATT, respectively. The NRC staff's review found that this determination was appropriate and that the licensee had considered the full set of postulated single failures defined in the Susquehanna UFSAR.

Consistent with ANP-10332P, break spectra were calculated for both mid- and top-peaked axial power profiles at the time of maximum fuel stored energy (i.e., near the beginning of the operating cycle). Furthermore, considering that Susquehanna is licensed to the MELLLA domain, sufficient initial statepoints were considered in the break spectrum analysis to provide confidence that the most limiting conditions have been analyzed. In particular, break spectra were performed for the statepoints shown below.

Table 2: LOCA Analysis Statepoints

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

The first two analyzed statepoints were selected to envelope the full range of permissible core flows at rated thermal power. The third statepoint represents limiting power**[[ ]]** conditions for single-loop operation. The NRC staff found the selected analysis statepoints acceptable because the licensee has taken appropriate regulatory guidance into account with respect to analyzing the MELLLA operating domain.

In RAI 4.1, the NRC staff requested additional information concerning how the LOCA analysis addresses the full suite of operating domains and equipment out-of-service conditions to which

Susquehanna has been licensed.[3] Table 3 below summarizes the licensee's response to RAI 4.1.

Table 3: Susquehanna Licensed Operating Domains

Licensed Domain	Disposition
Two-Loop (Normal) Operation	Explicitly analyzed two statepoints that correspond to the maximum licensed power level.
Single-Loop Operation	Explicitly analyzed statepoint corresponding to limiting power and flow conditions during single-loop operation.
MELLLA	Explicitly analyzed the **[[** **]]** at rated thermal power.
[[**]]**	Licensee qualitatively dispositioned this operating condition, stating that **[[** **]]**.

The NRC staff found the licensee's response to the RAI acceptable because it identified the existing set of licensed operating domains and provided an appropriate basis in each case for concluding that the limiting figures of merit calculated in its LOCA analysis bound all licensed operating conditions.

3.7.3.3 Exposure-Dependent LOCA Analysis Implementation

The NRC staff's review found the exposure study analysis described in ANP-3784P generally conforms to the approved evaluation model documented in ANP-10332P [34]. As shown in Table 9.1 of ANP-3784P, the exposure study considered **[[** **]]**. In particular, the exposure study analyzed **[[** **]]** accounting for exposure-dependent limiting values of the linear heat generation rate and maximum average planar linear heat generation rate.

The exposure study for Susquehanna described in ANP-3784P deviated from the methodology approved in the NRC staff's SE on ANP-10332P in that at **[[** **]]** approved by the NRC staff's SE. However, because these **[[** **]]** do not appear to produce limiting results in the analysis under review, the NRC staff found this deviation from the approved evaluation model acceptable for the Susquehanna LOCA analysis described in ANP-3784P.

3.7.4 <u>Calculated Results</u>

3.7.4.1 Break Spectrum Results

The break spectrum analysis results were reported in Tables 6.2 and 7.1 of ANP-3784P. The following tables summarize the peak cladding temperature (PCT). The limiting case (i.e., the case with the highest PCT **[[** **]]**) is the 0.07 ft^2 break in

[3] The provision of this information is necessary to satisfy Limitation and Condition 16 from the NRC staff's final SE on ANP-10332P.

the recirculation system discharge piping with a single failure of SF-BATT and a top-peaked axial power shape when operating at 102 percent rated power and **[[** **]]**. In contrast, the limiting case for single-loop operation is the $0.09\ ft^2$ break in the pump discharge piping with a single failure of SF-BATT and a top-peaked axial power shape when operating at **[[** **]]** Note that similar to the two-loop operation analysis, **[[** **]]**. The results presented for single-loop operation in Table 4 is the limiting case (i.e., highest PCT case for single-loop operation).

Table 4: ANP-3784 Tables 6.2 and 7.1
Summary of Break Spectrum Analysis Results for Two-Loop Operation and Single-Loop Operation Recirculation Line Breaks

[[

During the regulatory audit, the NRC staff reviewed the break spectrum analysis calculation book and confirmed that the above limiting cases for both two-loop operation and single-loop operation were identified correctly among the **[[** **]]**. The NRC staff also found that the **[[** **]]**. Based on the audit findings, the NRC staff found that the licensee's break spectrum evaluation is acceptable because the break spectrum results are consistent with both (1) the expected physical behavior for the LOCA event at a BWR and (2) the procedure for break spectrum analysis in the approved AURORA-B LOCA evaluation model described in ANP-10332P.

3.7.4.2 Exposure-Dependent LOCA Analysis

To ensure that the ATRIUM 11 exposure-dependent maximum average planar linear heat generation rate (MAPLHGR) limit and [[ ]] presented in ANP-3784 are applicable to Susquehanna, as described in ANP-3784, the licensee performed an exposure-dependent LOCA analysis. The licensee's exposure study for this limiting scenario predicted the figures of merit as shown below.

Table 5: Predicted Figures of Merit for Susquehanna Exposure-Dependent LOCA Analysis

Figure of Merit	Limiting Exposure	Predicted Value	Acceptance Criterion
Peak Cladding Temperature	[[	1,784 °F	≤ 2,200 °F
Maximum (Local) Cladding Oxidation		4.64%	≤ 17%
Maximum (Core-Wide) Hydrogen Generation	]]	< 0.30%	≤ 1%

The NRC staff identified that there is a difference for the limiting PCT results between [[ ]]. The NRC staff issued RAI 4.2 to resolve this discrepancy.

In the response to RAI 4.2, the licensee stated that the break spectrum calculations were performed [[ ]] The NRC staff found the response acceptable because the licensee provided the requested information and explained the difference in the limiting PCT due to [[ ]].

In RAI 4.3, the NRC staff requested that the licensee explain the "abrupt" change of local cladding oxidation from assembly average planar exposure [[ ] in Table 9.1 of ANP-3784P. The licensee responded that the abrupt change in local oxidation is due to [[ ]] The NRC staff found the response acceptable because the requested information had been provided and confirmed.

The NRC staff identified the following additional information to be requested in RAI 4.4:

- The process for determining the LHGR used for both UO_2 and Gd_2O_3-UO_2 pellets during exposure-dependent analysis in the AURORA-B LOCA analysis – specifically, the LHGR limit curves presented in Figures 2.2 and 2.3 as shown in ANP-3784P [[

]]

- Demonstration of the analysis margin for the MAPLHGR limit in Figure 2.1 of ANP-3784P **[[** **]]**.

The licensee responded for RAI 4.4.a as:

> The LHGR limit curves presented in Figures 2.2 and 2.3 from the exposure analysis **[[**
>
> **]**

The licensee responded for RAI 4.4.b as:

> Figure 4-3 shows the **[[**
>
> **]]**

The NRC staff found the responses acceptable because the requested information had been provided and confirmed as reasonable from the figures provided in the RAI response.

3.7.5 Conformance with Limitations and Conditions

The licensee provided information in Appendix A of ANP-3784 on how it satisfies all limitations and conditions from the NRC staff's SE on ANP-10332P. The licensee's proposed disposition of limitations and conditions is in conformance to the regulatory position imposed therein. However, in certain instances, as discussed below, the NRC staff found more detailed review necessary to confirm that the licensee had appropriately addressed the applicable limitations and conditions.

Regarding Limitation and Condition 14, the NRC staff confirmed from the licensee during an audit [33] that the figures of merit for Susquehanna in the **[[** **]]** had been determined and provided in Section 2.0 and its footnote of ANP-3784. Therefore, the NRC staff finds that the licensee adequately addressed this limitation and condition.

Regarding Limitation and Condition 19, the licensee dispositioned that the **[[** **]** for mixed cores. For the first cycle of applying ATRIUM 11 fuel, the core for Susquehanna will be a mixed core of ATRIUM 10 and ATRIUM 11 fuels. The NRC staff notes that the LOCA analysis results presented in the current UFSAR are based on an equilibrium core of ATRIUM 10 fuel. A comparison of current UFSAR LOCA analysis results (UFSAR Table 6.3-3C) with ANP-3784, Table 6.2 in **[[**

]] for the same 102 percent power**[[**
]] and axial power peaked. Although the licensee analyzed and demonstrated that the legacy ATRIUM 10 fuel would be **[[** **]]** with ATRIUM 11 fuel, the cause of the **[[** **]]** must be identified. The NRC staff confirmed the LOCA results presented by the licensee and concluded that the AURORA-B LOCA evaluation model described in ANP-10332P applied to the LOCA analysis for the **[[**
]] Both evaluation models (i.e., the current UFSAR LOCA analysis model and the AURORA-B LOCA model) are NRC-approved models and methodologies; the AURORA-B LOCA evaluation model described in ANP-10332P will be the analysis of record for ATRIUM 11 fuel and the EXEM BWR-2000 [35] analysis of record will remain in place for ATRIUM 10 fuel after this LAR is approved. Based on the above, the NRC staff finds that the licensee adequately addressed this limitation and condition.

3.7.6 <u>Conclusion for LOCA Analysis with ATRIUM 11 Fuel</u>

The NRC staff reviewed the information in the licensee's submittals pertaining to the analysis of the spectrum of postulated LOCA events for Susquehanna, including the submittal as well as relevant responses to RAIs. The NRC staff's review was further supported by a regulatory audit [32], which was used to confirm information referred to in docketed submittals. The NRC staff concludes that the LOCA analysis with ATRIUM 11 fuel to be used in Susquehanna, Units 1 and 2, is acceptable because it complies with the relevant requirements of 10 CFR 50.46, Appendix K to 10 CFR Part 50, and GDC 35. This conclusion is based on the following:

1. The licensee performed analyses of the performance of the ECCS with ATRIUM 11 fuel in accordance with 10 CFR 50.46. The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model that follows Appendix K to 10 CFR Part 50 and meets the requirements of 10 CFR 50.46. The results of the analyses (Sections 3.7.1.1 and 3.7.4 of this SE) show that the ECCS with ATRIUM 11 fuel satisfies the 10 CFR 50.46 criteria.

2. The evaluation meets the requirements of GDC 35 with respect to abundant emergency core cooling being provided that will transfer heat from the reactor core filled with ATRIUM 11 fuel in the event of a LOCA, and the suitable redundancy of components and features being provided so that the safety function can be accomplished assuming a single failure by:

    a. Demonstrating with the LOCA analysis performed for ATRIUM 11 to be used in Susquehanna that abundant emergency core cooling is provided to transfer heat from the reactor core filled with ATRIUM 11 fuel in the event of a LOCA and showing that suitable redundancy of components and features is provided so that the safety function can be accomplished assuming a single failure, irrespective of whether its electrical power is supplied from offsite or onsite sources (Sections 3.7.3 and 3.7.4 of this SE).

b. Applying the NRC-approved LOCA evaluation model and methodology for the LOCA analysis with ATRIUM 11 fuel and adequately meeting the limitations and conditions listed in the NRC staff's SE for the applied topical reports (Sections 3.7.2 and 3.7.5 of this SE).

3.8 Control Rod Drop Accident (CRDA)

3.8.1 Regulatory Basis

GDC 13 and 28 and 10 CFR 50.67 are pertinent to the analysis of CRDA events. GDC 13 primarily applies to the CRDA event by ensuring that the limiting system operating parameters and other controls in place (i.e., rod withdrawal limitations) are sufficient to ensure that the CRDA acceptance criteria are not exceeded. This is satisfied by ensuring that the initial conditions represented in the CRDA analyses are sufficiently representative of the most conservative condition allowed by the aforementioned controls. In addition, Susquehanna is licensed under 10 CFR 50.67 to establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.

The acceptance criteria for CRDA events to satisfy GDC 28 and 10 CFR 50.67 are currently defined in Chapter 15 of the SRP. Along with Chapter 15, SRP Section 4.2 provides an extensive discussion of acceptance criteria related to high temperature cladding failure, pellet clad mechanical interaction induced cladding failure, core coolability, and fission product inventory determination for dose assessment purposes. Regulatory Guides 1.183 and 1.195 are also referenced for further guidance related to fission product inventories.

However, the NRC staff is currently developing new guidance for rod injection accident acceptance criteria that will supersede SRP Section 4.2. The draft guidance document – draft guide (DG)-1327, has not become a final regulatory guide. The licensee indicated that it intends to adopt the DG-1327 criteria for use in analysis of the CRDA event. The NRC staff does not expect the specified acceptance criteria to change significantly, and the technical basis for use of the DG-1327 criteria is more robustly supported by recent research than the CRDA acceptance criteria that Susquehanna is currently licensed under. Therefore, the NRC staff considered the DG-1327 criteria at Susquehanna in lieu of SRP Section 4.2.

3.8.2 Technical Evaluation

In ANP-3771P (Enclosure 16a to the LAR the licensee provided information and some sample calculations demonstrating how the CRDA analysis methodology described in ANP-10333P-A [36] will be applied at Susquehanna to evaluate each cycle. The sample calculations were based on the equilibrium core design, but cycle-specific calculations will be performed to support each reload. A comparison of the information provided by the licensee against ANP-10333P-A shows that the licensee demonstrated an acceptable application of the methodology to evaluate the CRDA event for the Susquehanna equilibrium core design, with a few plant-specific nuances as discussed below. The licensee also provided information that allowed the NRC staff to confirm that all the limitations and conditions for ANP-10333P-A were met for the Susquehanna application.

In addition to finding that the information provided by the licensee shows that it will correctly apply the CRDA analysis methodology at Susquehanna, the NRC staff makes the following additional findings and observations specific to Susquehanna:

- In Section 6.4 of ANP-3753P, the CHF correlation used for the CRDA calculations is discussed. The range of applicability for the fuel-specific CHF correlations for ATRIUM 11 does not extend to the cold startup conditions that the CRDA analyses are performed at. Instead, the licensee used the **[[**

  ]] correlation to be acceptable for use for this purpose.

- The CRDA demonstration calculations utilize the fuel rod failure criteria from DG-1327, which has not yet completed the process of becoming a final regulatory guide. However, the NRC review and approval of ANP-10333P-A indicates that the methodology is acceptable for use with either the current CRDA acceptance criteria in Appendix B to SRP Section 4.2 or the new proposed criteria in DG-1327. Furthermore, the NRC has published the technical and regulatory basis for the new acceptance criteria in the "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," dated March 16, 2015 (ADAMS Accession No. ML14188C423) [37]. A review of this information indicates that sufficient evidence exists to support the use of the fuel failure threshold curves from DG-1327; therefore, the NRC staff finds the proposal to use the DG-1327 guidance in the manner described in ANP-3771P to be acceptable. The NRC staff also notes that the ATRIUM 11 fuel to be loaded at Susquehanna utilizes stress relief annealed unlined cladding, similar to the current ATRIUM 10 fuel at Susquehanna.

- Appendix A to ANP-3771P describes the process used to establish an evaluation boundary curve to simplify the calculations. This process was approved as part of the ANP-10333P-A methodology, with a limitation and condition requiring the licensee to confirm the applicability of the curve to several local characteristics that may be present in the core being analyzed. This information was presented for the equilibrium core, but the licensee will need to confirm that the evaluation boundary curve is also applicable to ATRIUM 10 fuel prior to use in analysis of the transition cores.

The licensee applied NRC-approved analytical methods to perform a demonstration CRDA analysis. The acceptance criteria are derived from the topical report for the approved CRDA analysis method. The licensee showed how it would determine whether fuel failures would occur and considered an artificial scenario where fuel failures occur so it could show how the radiological consequences would be evaluated. All calculations and evaluations were performed in a manner consistent with the basis for the NRC approval of the methods and demonstrated how they would determine whether acceptance criteria are met.

Based on the above, the NRC staff concludes that the proposed adoption of the CRDA analysis methods as part of the planned transition to ATRIUM 11 fuel is acceptable.

3.8.3 Conclusion

Pertaining to CRDA, the NRC staff reviewed the information in the Susquehanna submittals pertaining to the analysis of Susquehanna events, including the original submittal as well as relevant responses to RAIs [2], [3]. Based upon its review as summarized above, the NRC staff has concluded that:

1. The licensee has proposed to implement the AURORA-B CRDA evaluation model in an acceptable manner and

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

3.9 Revision of Low-Pressure Safety Limit in TSs 2.1.1.1 and 2.1.1.2

TSs 2.1.1.1 and 2.1.1.2 ensure that the critical power correlation is only evaluated within the approved range of applicability. The ACE/ATRIUM 11 correlation that will be used for the ATRIUM 11 fuel at Susquehanna [17] is valid at pressures of at least 575 psig to ensure that it results in valid calculated CPR values. Therefore, the licensee proposes to increase the low-pressure safety limit from 557 psig to 575 psig. The proposed new limit conservatively bounds existing application of the Siemens Power Corporation B (SPCB) correlation used for the ATRIUM 10 fuel. Accordingly, the proposed change to TSs 2.1.1.1 and 2.1.1.2 continues to ensure that a valid CPR calculation is performed for AOOs at Susquehanna and, therefore, the NRC staff finds it acceptable.

3.10 Removal of Neutronic Methods Penalties

3.10.1 OPRM Amplitude Setpoint

The current Susquehanna operating licenses include a license condition on the OPRM setpoint determination. The OPRM amplitude setpoint penalty is applied to account for a reduction in thermal neutrons around the low-power range monitor detectors caused by transients that increase voiding, ultimately reducing the OPRM scram setpoint. This license condition was created before an in-depth review of this issue was fully evaluated by the NRC staff in the RAMONA5-FA licensing topical report [38]. The NRC staff's review of the approved RAMONA5-FA methodology [39] concluded **[[**

]] Therefore, the NRC staff finds it acceptable to remove the OPRM amplitude setpoint penalty applied through this license condition.

3.10.2 Pin Power Distribution Uncertainty and Bundle Power Correlation Coefficient

The current Susquehanna operating licenses include a license condition for a penalty on SLMCPR pin power distribution uncertainty and bundle power correlation coefficient. No significant change in the uncertainty of the predicted detector response relative to the measurements is anticipated for the transition to ATRIUM 11 fuel. The NRC staff's review of the AURORA-B methodology concluded that since the analysis and core monitoring at Susquehanna is based upon the CASMO-4/MICROBURN-B2 methodology, there is no need for any uncertainty penalties when using AURORA-B methods, and the use of the **[[ ]]** correlation for ATRIUM 11 fuel is justified. In addition, since Susquehanna is currently operating within approved extended power uprate conditions (and not in extended flow windows), operating conditions are within previously approved power/flow ratios. Therefore, the NRC staff finds it acceptable to remove the pin power distribution uncertainty and bundle power correlation coefficient penalty applied through this license condition.

3.11 Technical Evaluation Conclusions

The NRC staff reviewed the licensee's analyses related to the effect of the proposed amendments for Susquehanna to allow application of the Framatome analysis methodologies necessary to support a planned transition to ATRIUM 11 fuel under the currently licensed MELLLA operating domain under extended power uprate conditions. The NRC staff further reviewed the licensee's proposed changes to TS 5.6.5.b that support adoption of the intended Framatome analysis methodologies, to TSs 2.1.1.1 and 2.1.1.2 to revise the low-pressure safety limit, and to license conditions to remove neutronic methods penalties on OPRM amplitude setpoint and the pin power distribution uncertainty and bundle power correlation coefficient. Based on its review, as summarized in this SE, the NRC staff concludes that the proposed amendments for Susquehanna are acceptable.

3.12 Vessels and Internals Branch Evaluation of Aging Degradation of Vessel Internals

The NRC staff determined that the assessment of aging degradation due to irradiation embrittlement in RPV base metal and welds is determined by the evaluation of pressure--temperature (P-T) limits, evaluation of upper shelf energy (USE) of the RPV beltline base metals and welds, and the evaluation of adjusted reference temperature (ART) for the beltline base metals and welds. Higher ART values and lower USE values indicate that RPV base metals and welds are embrittled. An increase in transition temperature of the RPV materials due to exposure to neutron fluence results in an increase in embrittlement, and this is reflected in higher operating temperature of the vessel for given operating pressure.

For the reactor vessel internals (RVI) components, boiling-water reactor units now examine the RPV interior surfaces, attachments, and core support structures in accordance with BWR vessel internals inspection program (BWRVIP) and evaluation guidelines. Operating experience to date indicates that aging degradation would be active when the accumulated neutron fluence exceeds threshold limits applicable to each of the following aging degradation mechanisms in the RVI components: (1) inspection and evaluation; (2) irradiation assisted stress corrosion cracking; (3) irradiation stress relaxation; and (4) irradiation embrittlement.

In its February 6, 2020, letter, the licensee submitted information on the aging degradation of the RPV and RVI components due to the implementation of ATRIUM 11 fuel at Susquehanna, Units 1 and 2. The licensee stated that one of the benefits of the ATRIUM 11 fuel design is that smaller reload batch sizes will be required. To successfully design a core with smaller reload batch size, a greater number of older bundles are moved to the periphery core locations. Due to their higher exposures, the older bundles will have lower fission power and, therefore, generate fewer fast neutrons at the core periphery when compared to an ATRIUM 10 core.

Furthermore, the licensee stated that based on this general understanding of the core physics, the expectation is that the neutron fluence at the RPV wall, and also for beltline components located within the RPV (e.g., core shroud, jet pump components), will decrease. The licensee stated that an analysis of the fast neutron fluence in the RPV plates, welds, and nozzles throughout the beltline region, determined at 60 years, has been completed using the NRC -approved RAMA fluence methodology. The fast neutron fluence was determined in accordance with the guidelines and requirements presented in RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.”

In addition, the licensee stated that a review of the analysis results determined that the fast neutron fluence levels at the reactor vessel wall throughout the beltline region with ATRIUM 11 fuel is lower than with ATRIUM 10 fuel when analyzed out to 60 years. Similarly, an analysis was performed for RPV beltline internal components (e.g., core shroud, jet pump components, top guide, core plate) using the same methods as described above. These results are consistent with the fundamental understanding of the core physics for ATRIUM 11 fuel. Based on the above discussion, the change from ATRIUM 10 to ATRIUM 11 fuel results in a lower fast neutron fluence for both the RPV and RVI components located within the RPV. Based on this result, there is no effect on the aging degradation due to the transition to ATRIUM 11 fuel at Susquehanna, Units 1 and 2, in the current licensing period.

The implementation of ATRIUM 11 fuel design requires smaller reload batch sizes. Designing a core with smaller reload batch sizes results in a greater number of older bundles being moved to the periphery core locations. Due to their higher exposures, the older bundles will have lower fission power and, therefore, generate fewer fast neutrons at the core periphery when compared to an ATRIUM 10 core. This fuel arrangement falls under the “low leakage” category, which indicates that outer periphery in the beltline region will be exposed to lower fast neutron fluence (Energy level > 1 MeV) in comparison to the locations near the core region. Therefore, the NRC staff concludes that RPV and RVI components would be exposed to lower neutron fluence values with high energy levels (i.e., > 1 MeV) than the previous period of operations with ATRIUM 10 fuel.

Some of the RPV and RVI components in the beltline region may not have been exposed to neutron fluence values exceeding threshold limits for the onset of aging degradation mechanisms. In this case, there would be a delay in the onset of any aging degradation due to exposure to lower neutron fluence radiation associated with “low leakage” fuel arrangement with older bundles. The affected components are (1) RPV beltline base metals and welds (irradiation embrittlement is only active degradation mechanism) and (2) RVI beltline base metals and welds in core shroud, top guide, core plate, and jet pump components. For components that were already exposed to neutron fluence threshold limits, the damage associated with the aging degradation would be reduced. This is due to exposure to lower fluence values related to a “low leakage” fuel arrangement with older fuel bundles to the core periphery locations.

Based on the above, the NRC staff determined that the evaluation of inspection and evaluation of the RPV base metals and welds, which includes development of P-T limits, the evaluation of USE of the RPV beltline welds, and the evaluation of ART for the beltline base metals and welds will remain valid for the current licensing period at Susquehanna. Based on the plant operations, the neutron fluence values would increase over time, which requires reevaluation of the P-T limits, USE, and ART values. Accordingly, the licensee is expected to update these values at that time. Current aging management programs for the RVI components include implementation of NRC-staff approved BWRVIP inspection and evaluation guidelines at Susquehanna, Units 1 and 2. This program evaluates aging effects in components (i.e., core shroud, top guide, core plate, and jet pumps) and it remains valid for the current licensing period at Susquehanna. Therefore, the staff finds that the licensee's implementation of ATRIUM 11 fuel at Susquehanna is acceptable.

3.12.1 Conclusion Regarding Aging Degradation

The NRC staff determined that implementation of ATRIUM 11 fuel at Susquehanna, Units 1 and 2, results in exposure to lower neutron fluence values on the RPV and RVI components due to "low leakage" fuel arrangement. Accordingly, the staff determined that the current evaluation for the RPV base metals and welds and current aging monitoring program for the RVI components remain valid for Susquehanna during the current licensing period and, therefore, concludes that the licensee's implementation of ATRIUM 11 fuel at Susquehanna is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the NRC's proposed issuance of the amendments on June 16, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (84 FR 56482; October 22, 2019). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 278 AND 260 TO ALLOW APPLICATION OF ADVANCED FRAMATOME ATRIUM 11 FUEL METHODOLOGIES (EPID L-2019-LLA-0153) DATED JANUARY 21, 2021

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