Enclosures 8a, 9a, 10a, 11a, 12a, 13a, 14a, 15a, and 16a Contain Proprietary Information – Withhold in Accordance with 10 CFR 2.390

Kevin Cimorelli
Site Vice President

Susquehanna Nuclear, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3795 Fax 570.542.1504
Kevin.Cimorelli@TalenEnergy.com

TALEN ENERGY

Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10 CFR 50.90

JUL 1 5 2019

SUSQUEHANNA STEAM ELECTRIC STATION PROPOSED AMENDMENT TO LICENSES NPF-14 AND NPF-22: APPLICATION OF ADVANCED FRAMATOME METHODOLOGIES AND TSTF-535
PLA-7783

Docket No. 50-387 and 50-388

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), is submitting a request for an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License numbers NPF-14 and NPF-22, to improve safety margins and fuel cycle economics. The proposed change revises 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. Further, the proposed change revises the low pressure safety limit in TS 2.1.1.1 and TS 2.1.1.2 and removes the neutronic methods penalties on Oscillation Power Range Monitor amplitude setpoint, and the pin power distribution uncertainty and bundle power correlation coefficient that were added during the Extended Power Uprate approved in Amendment 246/224 (ADAMS Accession No. ML080020201). The penalties are no longer warranted with the introduction of the Advanced Framatome Methodologies.

Additionally, the proposed change would adopt Technical Specification Task Force (TSTF) Traveler TSTF-535, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs." Specifically, the proposed change modifies the TS definition of "Shutdown Margin" (SDM) to require calculation of the SDM at a reactor moderator temperature of 68°F or a higher temperature that represents the most reactive state throughout the operating cycle. This change is needed to address new Boiling Water Reactor fuel designs which may be more reactive at shutdown temperatures above 68°F.

Enclosure 1 provides a description and assessment of the proposed changes along with Susquehanna's determination that the proposed changes do not involve a significant hazard consideration. Enclosure 2 provides the existing Unit 1 and Unit 2 Operating License pages marked to show the proposed changes. Enclosure 3 provides the revised (clean) Operating License pages. Enclosure 4 provides the existing TS pages marked to show the proposed

changes. Enclosure 5 provides revised (clean) TS pages. Enclosure 6 provides existing TS Bases pages marked to show the proposed changes and are provided for information only. Enclosure 7 provides a list of regulatory commitments associated with the proposed change.

Information submitted in enclosures to this letter is considered proprietary to Framatome (i.e., Enclosures 8a, 9a, 10a, 11a, 12a, 13a, 14a, 15a, and 16a). Within these enclosures, proprietary information has been denoted by brackets. As owners of the proprietary information, Framatome has executed affidavits for each proprietary document, which identify the information as proprietary, is customarily held in confidence, and should be withheld from public disclosure in accordance with 10 CFR 2.390. Enclosures 8b, 9b, 10b, 11b, 12b, 13b, 14b, 15b, and 16b provide non-proprietary versions of each proprietary Framatome document. Corresponding affidavits are provided in Enclosures 8c, 9c, 10c, 11c, 12c, 13c, 14c, 15c, and 16c.

Susquehanna requests NRC approval of the proposed changes and issuance of the requested license amendment by January 31, 2021 to support core loading and reactor startup following the Unit 2 refueling outage. Once approved the Unit 2 amendment shall be implemented prior to loading ATRIUM 11 fuel into the core during the spring 2021 refueling outage, and the Unit 1 amendment shall be implemented prior to loading ATRIUM 11 fuel into the core during the spring 2022 refueling outage.

In accordance with 10 CFR 50.91, Susquehanna is providing a copy of this application, with enclosures, to the designated Commonwealth of Pennsylvania state official.

Both the Plant Operations Review Committee and the Nuclear Safety Review Board have reviewed the proposed changes.

Should you have any questions regarding this submittal, please contact Ms. Melisa Krick, Manager – Nuclear Regulatory Affairs, at (570) 542-1818.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 7/15/19

K. Cimorelli

Enclosures:

1. Description and Assessment
2. Marked-Up Operating License Pages

3. Revised (Clean) Operating License Pages
4. Marked-Up Technical Specification Pages
5. Revised (Clean) Technical Specification Pages
6. Marked-Up Technical Specification Bases Pages (For Information Only)
7. List of Regulatory Commitments

8a. ANP-3753P, Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel Report **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

8b. ANP-3753NP, Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel Report

8c. Affidavit for ANP-3753P, Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel Report

9a. ANP-3762P, Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

9b. ANP-3762NP, Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies

9c. Affidavit for ANP-3762P, Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies

10a. ANP-3761P, Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

10b. ANP-3761NP, Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies

10c. Affidavit for ANP-3761P, Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies

11a. ANP-3745P, ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Susquehanna LAR **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

11b. ANP-3745NP, ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Susquehanna LAR

11c. Affidavit for ANP-3745P, ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Susquehanna LAR

12a. ANP-3727P, Susquehanna ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

12b. ANP-3727NP, Susquehanna ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report

12c. Affidavit for ANP-3727P, Susquehanna ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report

13a. ANP-3724P, Susquehanna ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

13b. ANP-3724NP, Susquehanna ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report

13c. Affidavit for ANP-3724P, Susquehanna ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report

14a. ANP-3783P, Susquehanna ATRIUM 11 Transient Demonstration **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

14b. ANP-3783NP, Susquehanna ATRIUM 11 Transient Demonstration

14c. Affidavit for ANP-3783P, Susquehanna ATRIUM 11 Transient Demonstration

15a. ANP-3784P, Susquehanna Units 1 and 2 LOCA Analysis for ATRIUM 11 Fuel **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

15b. ANP-3784NP, Susquehanna Units 1 and 2 LOCA Analysis for ATRIUM 11 Fuel

15c. Affidavit for ANP-3784P, Susquehanna Units 1 and 2 LOCA Analysis for ATRIUM 11 Fuel

16a. ANP-3771P, Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology **(Proprietary Information – Withhold from Public Disclosure in Accordance With 10 CFR 2.390)**

16b. ANP-3771NP, Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology

16c. Affidavit for ANP-3771P, Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology

Copy: NRC Region I
Ms. L. H. Micewski, NRC Sr. Resident Inspector
Ms. T. E. Hood, NRC Project Manager
Ms. J. C. Tobin, NRC Project Manager
Mr. M. Shields, PA DEP/BRP (w/o Proprietary Enclosures)

Enclosure 1 to PLA-7783

Description and Assessment

SUSQUEHANNA ASSESSMENT

1. Summary Description

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), is submitting a request for an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License numbers NPF-14 and NPF-22. The proposed change revises 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. Further, the proposed change revises the low pressure safety limit (SL) in TS 2.1.1.1 and TS 2.1.1.2 and removes the neutronic methods penalties on Oscillation Power Range Monitor (OPRM) amplitude setpoint, pin power distribution uncertainty, and bundle power correlation coefficient that were added during the Extended Power Uprate approved in Amendment 246/224 (Reference 1); the penalties are no longer warranted with the introduction of the Advanced Framatome Methodologies.

Additionally, the proposed change would adopt Technical Specification Task Force (TSTF) Traveler TSTF-535, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs." Specifically, the proposed change modifies the TS definition of "Shutdown Margin" (SDM) to require calculation of the SDM at a reactor moderator temperature of 68°F or a higher temperature that represents the most reactive state throughout the operating cycle. This change is needed to address new Boiling Water Reactor (BWR) fuel designs which may be more reactive at shutdown temperatures above 68°F.

2. Detailed Description

2.1 System Design and Operation

Core operating limits are established each operating cycle. These operating limits ensure that the fuel design limits are not exceeded during any conditions of normal operation and in the event of any Anticipated Operational Occurrence (AOO).

2.2 Current Technical Specification Requirements

TS 1.1 defines SDM, among other requirements, to be calculated at a moderator temperature of 68°F.

TS 2.1.1.1 establishes, for each unit, the requirement that at a reactor steam dome pressure below 557 psig or core flow below 10 million lbm/hr, the reactor power level be no more than 23 percent RATED THERMAL POWER. Further, TS 2.1.1.2 establishes the requirement that at pressure greater than 557 psig and steam flow greater than 10 million lbm/hr, the MINIMUM

CRITICAL POWER RATIO (MCPR) be at least 1.09 or 1.08 (for Unit 1 or 2, respectively) when two recirculation loops are in operation and at least 1.12 or 1.11 (for Unit 1 or 2, respectively) with only one recirculation loop in operation.

The Core Operating Limits Report (COLR) is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits are determined for each reload cycle in accordance with TS 5.6.5.

TS 5.6.5.a lists the core operating limits required to be established for each cycle. The methods used to determine the operating limits are those previously found acceptable by the NRC and are listed in TS 5.6.5.b.

2.3 Reason for the Proposed Change

Susquehanna plans to transition to the Framatome fuel type ATRIUM 11. These proposed license amendments to allow application of Advanced Framatome Methodologies are necessary for this fuel transition. Susquehanna is pursuing the ATRIUM 11 fuel type due to the improved fuel cycle economics and safety margins.

With implementation of the Advanced Framatome Methodologies, current penalties on neutronic methods, added during the EPU approved in Reference 1 are no longer necessary.

The ATRIUM 11 fuel type consists of an 11 by 11 array of fuel rods, whereas the current fuel design (i.e., ATRIUM 10) consists of a 10 by 10 array of fuel rods. This increase in the number of fuel rods significantly reduces LINEAR HEAT GENERATION RATE (LHGR) and fuel duty, thereby improving safety margin.

The ATRIUM 11 fuel type incorporates enhanced debris protection features which make the fuel design less susceptible to debris related fuel failures. In addition, the channel design changes incorporated with ATRIUM 11 make the fuel design less susceptible to channel bow and bulge.

Based on the physical properties of ATRIUM 11 fuel, the most reactive state may occur at a moderator temperature greater than 68°F. Modifying the definition of SDM to require evaluation at a reactor moderator temperature of 68°F or a higher temperature ensures that the SDM is evaluated at the most reactive state throughout the operating cycle for the most reactive moderator temperature.

TS 2.1.1.1 and 2.1.1.2 ensure that the critical power correlation is only evaluated within the NRC-approved range of applicability. The ACE/ATRIUM 11 correlation that will be used for the ATRIUM 11 fuel requires a slightly higher low pressure limit to ensure it results in valid calculated Critical Power Ratio (CPR) values. The new low pressure limit is 575 psig and

conservatively bounds existing application of the SPCB correlation used for the ATRIUM 10 fuel. The proposed change to TS 2.1.1.1 and 2.1.1.2 continues to ensure that a valid CPR calculation is performed for AOOs described in the Final Safety Analysis Report (FSAR).

2.4 Description of the Proposed Change

SHUTDOWN MARGIN Definition

The definition of SDM in TS 1.1 is modified to require evaluation of SDM at a reactor moderator temperature of 68°F or a higher temperature corresponding to the most reactive state throughout the operating cycle.

Low Pressure Safety Limit

The reactor steam dome pressure value in TS 2.1.1.1 and 2.1.1.2 is raised from 557 psig to 575 psig. This change is required to reflect that the ACE/ATRIUM 11 correlation (Reference 2) is valid for critical power calculations at pressures of at least 575 psig. This change also conservatively bounds the SPCB correlation (Reference 3) which will continue to be used for the ATRIUM 10 fuel designs.

Advanced Framatome Methodologies

The following methodologies will be removed from TS 5.6.5.b:

- ANF-524(P)(A), “ANF Critical Power Methodology for Boiling Water Reactors,” Advanced Nuclear Fuels Corporation

- ANF-913(P)(A), “COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,” Advanced Nuclear Fuels Corporation

- XN-NF-84-105(P)(A), “XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis,” Exxon Nuclear Company

- NE-092-001A, “Licensing Topical Report for Power Uprate with Increased Core Flow,” Pennsylvania Power & Light Company

The above methodologies are no longer applicable with addition of the Advanced Methodologies described below.

The Advanced Methodologies that will be added to TS 5.6.5.b are listed below:

- BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA Inc. (References 4, 5, and 6)

- ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," Framatome Inc. (Reference 7)

- ANP-10335P-A, "ACE/ATRIUM 11 Critical Power Correlation," Framatome Inc. (Reference 2)

- ANP-10300P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," Framatome Inc. (Reference 8)

- ANP-10332P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," Framatome Inc. (Reference 9)

- ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," Framatome Inc. (Reference 10)

- ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP Inc. (Reference 11)

SSES Unit 1 and Unit 2 Operating License markups are provided in Enclosure 2. Clean (re-typed) versions of the Operating License pages are provided in Enclosure 3. TS markups are provided in Enclosure 4. Clean (re-typed) versions of the TS pages are provided in Enclosure 5. Additionally, TS Bases markups are provided in Enclosure 6 for information only.

OPRM Amplitude Setpoint Penalty License Condition

The proposed change will also remove Unit 1 License Condition 2.C.(38)(a) and Unit 2 License Condition 2.C.(22)(a). These License Conditions require Susquehanna to reduce the OPRM scram setpoint to account for a reduction in thermal neutrons around the Local Power Range Monitor (LPRM) detectors caused by transients that increase voiding at EPU conditions. This commitment was to be applied until NRC evaluations determined that a penalty to account for this phenomenon is not warranted. During the NRC review and approval of BAW-10255PA, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," (Reference 12) the NRC explicitly reviewed the determination of bypass voiding and its impact on LPRM and OPRM response. The NRC staff review concluded that "the methods and procedures documented in the TR [Topical Report], and as supplemented by the responses to the NRC staff's RAI [request for additional information], represent a technically acceptable methodology to calculate DIVOM [Delta CPR over Initial CPR Versus Oscillation Magnitude] slope values." Further, "the slope values calculated by AREVA DIVOM Methodology are applicable to any D&S [Detect and Suppress] long term stability solution methodology that

requires a setpoint calculation to suppress power oscillation before specified acceptable fuel design limits are compromised." Therefore, no additional penalties beyond that described in the approved TR are required.

Pin Power Uncertainty and Bundle Power Correlation Coefficient License Condition

The proposed change will also remove Unit 1 License Condition 2.C.(38)(b) and Unit 2 License Condition 2.C.(22)(b). These License Conditions require Susquehanna to conservatively adjust the pin power distribution uncertainty and bundle power correlation coefficient when performing analyses in accordance with ANF-524(P)(A), "Critical Power Methodology for Boiling Water Reactors," (Reference 13) using the uncertainty parameters associated with EMF-2158(P)(A) "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2" (Reference 14). The pin power distribution calculated within a fuel assembly is used to determine the CPR of that assembly during normal operation, normal operational transients and AOOs. During the review of Amendment 246/223 (Reference 1), the NRC required a conservative adjustment to pin power uncertainty and bundle power correlation coefficient applied to the safety limit MCPR (SLMCPR) calculation to account for the fact that there was limited test data available under EPU conditions. As a result, the License Conditions were added to require this conservative adjustment to the pin power distribution uncertainty and bundle power correlation coefficient (Reference 15). Since that time, Framatome has provided additional gamma scan, Traversing In-Core Probe (TIP) statistics and LPRM data as part of the AURORA-B submittal and approval (Reference 8). The NRC has concluded in the AURORA-B safety evaluation that the additional penalty for EPU conditions is no longer required.

3. Technical Evaluation

3.1 TSTF-535 Assessment

3.1.1 Applicability of Published Safety Evaluation

Susquehanna has reviewed the model safety evaluation dated February 19, 2013, as part of the Federal Register Notice of Availability. This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-535. Susquehanna has concluded that the justifications presented in the TSTF-535 proposal and the model safety evaluation prepared by the NRC staff are applicable to SSES, Units 1 and 2, and justify this amendment for the incorporation of changes to the SSES TS.

3.1.2 Optional Changes and Variations

Susquehanna is proposing the following variations from the TS changes described in the TSTF-535, Revision 0, or the applicable parts of the NRC staff's model safety evaluation dated February 19, 2013.

The SSES TS definition for SDM is arranged slightly different from that of the standard TS on which TSTF-535 was based. The final sentence of the definition of SDM in the SSES TS is separated from sub-heading c. In the standard TS, as shown in NUREG-1433 (Reference 16), the final sentence of the definition of SDM is part of sub-heading c. Therefore, in addition to the changes described in TSTF-535, Susquehanna proposes modifying the definition of SDM to place the last sentence of the definition of SDM under sub-heading c.

This variation is administrative in nature, and results in the alignment of the SSES TS with the standard TS wording in NUREG-1433. It does not impact the conclusion that TSTF-535 is applicable to the SSES TS, nor does it preclude the NRC's conclusion that the change is acceptable as documented in the Federal Register Notice of Availability. Therefore, Susquehanna concludes this administrative variation is acceptable.

3.2 Low Pressure Safety Limit Criteria

The changes described in Section 2.4 of this enclosure for the low pressure SL were made necessary by use of the ACE/ATRIUM 11 correlation for monitoring the ATRIUM 11 fuel design as supported in this license amendment request. Note that the SPCB correlation (Reference 3) will continue to be used for the ATRIUM 10 fuel assembly design and operation. The current NRC approval for the ACE/ATRIUM 11 correlation (Reference 2) is valid for critical power calculations at pressures of at least 575 psig. The current 557 psig limit is based on the use of the SPCB correlation.

10 CFR 50, Appendix A, General Design Criteria (GDC) 10 requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients or AOOs. TS 2.1.1.1 and 2.1.1.2 ensure compliance with GDC 10 by setting reactor conditions such that no significant fuel damage will occur if conditions are met.

The changes required in the steam dome low pressure limit for SLMCPR applicability in TS 2.1.1.1 and 2.1.1.2 are to ensure that the critical power correlation is only evaluated within the NRC-approved range of applicability. If the steam dome pressure is lower than the applicable limit, then restrictions on the core thermal power and flow are such that no significant fuel damage will occur. With steam dome pressure at least 575 psig, the ACE/ATRIUM 11 or SPCB correlation are within their respective NRC-approved range of applicability and hence can be used to ensure the SLMCPR will not be violated during steady state operation, normal operational transients or AOOs, again ensuring that no significant fuel damage will occur.

Since the low pressure criteria for the ACE/ATRIUM 11 correlation bounds the low pressure criteria for the SPCB correlation, use of the low pressure limit of at least 575 psig will bound both correlations.

3.3 Advanced Framatome Methodologies

Enclosures 8a through 16a provide the detailed technical evaluation for the proposed change outlined in Section 2.4 of this enclosure. The information presented in these enclosures demonstrates acceptable safety margin for the proposed change supporting operation of the new ATRIUM 11 fuel type in the currently approved operating domain. The currently approved operating domain includes EPU conditions, approved for SSES in 2008 (Reference 1) as well as Maximum Extended Load Line Limit Analysis (MELLLA), approved for SSES in 2007 (Reference 17).

Table 1 – Advanced Methodology Applicability to Analyses Provided in this Request

Methodology	Application
BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Framatome Inc. (References 4, 5, and 6)	Enclosures 9a, 11a, 14a, and 16a
ANP-10340P-A, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," Framatome Inc. (Reference 7)	Enclosure 11a
ANP-10335P-A, "ACE/ATRIUM 11 Critical Power Correlation," Framatome Inc. (Reference 2)	Enclosures 10a, 12a, and 14a
ANP-10300P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," Framatome Inc. (Reference 8)	Enclosure 14a
ANP-10332P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," Framatome Inc. (Reference 9)	Enclosure 15a
ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," Framatome Inc. (Reference 10)	Enclosure 16a
ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP Inc. (Reference 11)	Enclosure 14a

The sections below provide a brief summary of what is included in the enclosures. Table 1 is provided to correlate the Advanced Methodologies that will be added to TS 5.6.5.b with the enclosures in which the methodology is applied. Note that the enclosures with the 'a'

designation provide the full report, while the enclosures with the ‘b’ designation provide the non-proprietary version of the full report (i.e., proprietary information is redacted). For ease of reference throughout this request, only enclosures with the ‘a’ designation are referenced in discussions.

Enclosure 8a: ANP-3753P, Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel Report

ANP-2637P, “Boiling Water Reactor Licensing Methodology Compendium,” is a compendium of Framatome methodologies and design criteria, which are described in TRs that the NRC has found acceptable for referencing in BWR licensing applications. Framatome provided this document to the NRC for information by letter dated June 5, 2019 (Reference 18). This compendium provides a concise, organized source for BWR TRs. It presents information about the application of each TR, the associated Safety Evaluation Report (SER) and its conclusions and restrictions/limitations for each TR, the relationships among the TRs, and, for certain methodologies, descriptions of their unique characteristics or applications. Compliance with the SER restrictions/limitations is typically assured by implementing them within the engineering guidelines or by incorporating them into the computer codes.

ANP-3753P demonstrates that the Framatome licensing methodologies presented in ANP-2637P are applicable to the ATRIUM 11 fuel type and operation of SSES in the currently approved EPU operating domain.

Enclosure 9a: ANP-3762P, Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies

ANP-3762P documents the successful completion of all licensing analyses and related testing necessary to verify that the mechanical design criteria are met for the ATRIUM 11 fuel assemblies supplied by Framatome for insertion into the SSES reactors. This report also provides a description of the mechanical design and licensing methods for ATRIUM 11. The scope of this report is limited to an evaluation of the mechanical design of the fuel assembly and fuel channel. The fuel assembly design was evaluated according to the Framatome BWR generic mechanical design criteria (Reference 19). The fuel channel design was evaluated to the criteria given in the fuel channel TRs (References 20 and 21). The generic design criteria have been approved by the NRC and the criteria are applicable to the subject fuel assembly and channel design. Mechanical analyses for ATRIUM 11 have been performed using NRC-approved design analyses methodology (References 5, 19, 20, and 21).

Enclosure 10a: ANP-3761P, Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies

ANP-3761P presents the results of SSES thermal-hydraulic analyses which demonstrate that Framatome ATRIUM 11 fuel is hydraulically compatible with the previously loaded ATRIUM 10 fuel design. These reports also provide the hydraulic characterization of the ATRIUM 11 and the coresident ATRIUM 10 design for both units. The generic thermal-hydraulic design criteria applicable to the design have been reviewed and approved by the NRC in Reference 19. In addition, thermal-hydraulic criteria applicable to the design have also been reviewed and approved by the NRC in Reference 22.

Enclosure 11a: ANP-3745P, ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Susquehanna LAR

ANP-3745P reports the results of thermal-mechanical analyses for the performance of ATRIUM 11 fuel assemblies inserted into an equilibrium cycle for the SSES units and demonstrates that the design criteria relevant to the thermal-mechanical limits are satisfied. These analyses assume the use of chromia additive in the fuel and assume operation in the currently approved operating domain. Both the design criteria and the analysis methodology used in this report have been approved by the NRC. The analysis results are evaluated according to the generic fuel rod thermal and mechanical design criteria contained in Reference 19 along with design criteria provided in Reference 4. In addition, the approved methodology for the inclusion of chromia additive in the fuel pellets (Reference 7) is also used.

Enclosure 12a: ANP-3727P, Susquehanna ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report

In ANP-3727P, Framatome has performed an equilibrium fuel cycle design for SSES. This design uses the ATRIUM 11 fuel assembly and the currently approved operating domain. This analysis has been performed with the approved Framatome neutronic modeling methodology (Reference 14). This analysis has also used the Reference 2 critical power methodology. The CASMO-4 lattice depletion code was used to generate nuclear data including cross sections and local power peaking factors. The MICROBURN-B2 three-dimensional core simulator code, combined with the ACE/ATRIUM 11 critical power correlation, was used to model the core. The MICROBURN-B2 pin power reconstruction model was used to determine the thermal margins presented in the report. Design results including projected control rod patterns and evaluations of thermal and reactivity margins are presented.

Enclosure 13a: ANP-3724P, Susquehanna ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report

ANP-3724P provides results of the neutronic design analyses performed by Framatome for SSES ATRIUM 11 equilibrium cycle fuel assemblies (i.e., used in Enclosure 12a). NRC-approved neutronic design criteria are provided in Reference 19 and the NRC-approved neutronic design analysis methodology (Reference 14) was used to determine conformance to design criteria. Pertinent fuel design information is given in Section 2.0 and in Appendices A through D of this enclosure.

Enclosure 14a: ANP-3783P Susquehanna ATRIUM 11 Transient Demonstration

ANP-3783P summarizes the results of a subset of limiting transient analyses performed to show example SSES results utilizing the References 2 and 8 methodologies based upon an equilibrium cycle of ATRIUM 11 fuel (i.e., Enclosure 12a). The AURORA-B AOO methodology (Reference 8) is used to calculate the change in the minimum critical power ratio (ΔMCPR) during the AOO. The SLMCPR is determined using the Reference 2 and 11 methodology. The ΔMCPR is combined with the SLMCPR to establish or confirm the plant operating limits for MCPR. The AURORA-B AOO methodology is also used to calculate the maximum reactor vessel pressure and the maximum dome pressure during the American Society of Mechanical Engineers overpressure and Anticipated Transient Without Scram events. The ACE/ATRIUM 11 critical power correlation (Reference 2) is used to evaluate the thermal margin of the ATRIUM 11 fuel.

Enclosure 15a: ANP-3784P, Susquehanna LOCA Analysis for ATRIUM 11 Fuel

ANP-3784P presents the results of a loss of coolant accident (LOCA) break spectrum and emergency core cooling system (ECCS) analyses for SSES Units 1 and 2. The analyses documented in this report are performed with Framatome LOCA evaluation models for reactor licensing analyses. The models and computer codes used by Framatome for LOCA analyses are collectively referred to as the AURORA-B LOCA Evaluation Model (References 4, 9, and 23). The purpose of the break spectrum analysis is to identify the break characteristics that result in the highest calculated peak cladding temperature (PCT) during a postulated LOCA. The results provide the MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE limit for ATRIUM 11 fuel as a function of exposure. The calculations described in this report are performed in conformance with 10 CFR 50, Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46.

<u>Enclosure 16a: ANP-3771P, Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology</u>

ANP-10333P-A (Reference 10) is the Framatome methodology to analyze the BWR CRDA. The methodology includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods.

The Framatome methodology for the CRDA evaluation includes both generic evaluations and cycle-specific analysis. Generic studies are used to address at-power conditions and system pressurization. The cycle-specific analysis includes the determination of candidate control rods that could challenge fuel failure criteria and the subsequent evaluation of these candidate rods with a three-dimensional neutron kinetics and thermal-hydraulics code system.

This methodology has been developed to support recent changes in the CRDA acceptance criteria and evaluation process as reflected in the Interim Acceptance Criteria and Guidance of Appendix B of NUREG-0800, Section 4.2 (Reference 24).

ANP-3771P provides the initial application demonstration of the new CRDA methodology (Reference 10). This CRDA analysis is performed using the ATRIUM 11 equilibrium cycle design (i.e., Enclosure 12a). Though not part of the SSES licensing basis, the criteria used for the SSES initial application demonstration are based upon Draft Regulatory Guide DG-1327 (Reference 25) which was also used in the Reference 26 responses to NRC RAI.

<u>ATRIUM 11 Fuel Design and Cycle Specific Reports</u>

The NRC-approved the use of Framatome fuel and core design methodologies to determine SSES core operation limits with the issuance of License Amendments 231 and 194 for SSES Units 1 and 2, respectively (References 27 and 28). Framatome TR ANF-89-98(P)(A), Revision 1 and Supplement 1 (Reference 19), is one of these NRC-approved methodologies. Reference 19, as clarified by a Siemens Power Corporation letter dated October 12, 1999 (Reference 29), and an NRC letter dated May 31, 2000 (Reference 30), requires that a summary of the evaluation of the ATRIUM 11 design against the NRC-approved generic design criteria be provided to the NRC for information. Framatome provided this evaluation to the NRC for information by letter dated September 18, 2018, which transmitted Framatome document ANP-3653P, Revision 0, “Fuel Design Evaluation for ATRIUM 11 BWR Reload Fuel” (Reference 31). In accordance with the process described in Reference 19, new fuel designs or fuel design changes satisfying the ANF-89-98(P)(A) design criteria do not require explicit NRC review and approval (i.e., satisfaction of the design criteria is sufficient for approval by reference to the criteria).

ANP-3653P identifies fuel design criteria, specified in ANF-89-98(P)(A), Revision 1, and Supplement 1, which are evaluated on a cycle-specific basis. Reports summarizing the results of analyses performed to demonstrate SSES compliance with the cycle-specific criteria are provided by Framatome to Susquehanna as part of the normal reload licensing document package. This type of information is not available until later in the reload licensing process. Consistent with the process described in ANF-89-98(P)(A), Revision 1 and Supplement 1 (as clarified by References 29 and 30), Susquehanna will provide the SSES Unit 2 Cycle 21 reload reports outlined in the table below to the NRC for information. The reports will be provided in supplemental letters as documented in Enclosure 7. The anticipated schedule is presented in Table 2.

Table 2 – Anticipated Submittal Schedule of SSES Unit 2 Cycle 21 Reload Reports

Report	Estimated Transmittal Date
Fuel Cycle Design Report	April 2020
Nuclear Fuel Bundle Design Report	April 2020
SLMCPR Report	July 2020
Fuel Rod Design Report	November 2020
Reload Safety Analysis Report	November 2020

ANP-3653P also identifies fuel design criteria, specified in ANF-89-98(P)(A), Revision 1 and Supplement 1, that are evaluated on a plant-specific basis. SSES 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 this submittal, and the information in Table 2 which will be provided for Unit 2 Cycle 21, limited information needs to be provided for Unit 1. Therefore, Susquehanna will include, for information, the Unit 1 Cycle 23 Reload Safety Analysis Report with transmittal of the COLR 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.

Enclosure 7 documents the commitment to provide these reports.

3.4 OPRM Amplitude Setpoint Penalty License Condition

The ability of any D&S solution to prevent fuel failure that could occur during core wide or local power instabilities depends on timely detection of oscillatory behavior by monitoring signals of several OPRMs against predefined setpoints and determination of the MCPR margin that exists prior to the onset of the oscillation. Plant and cycle specific calculations determine the

minimum expected MCPR prior to the potential onset of oscillatory behavior. Statistical calculations of the peak oscillation magnitude capture the effects of the plant specific trip system and are used to determine the required OPRM setpoints to prevent fuel damage hot channel power oscillation immediately before its suppression by scram. The determination of this setpoint is plant and cycle specific.

During implementation of Framatome methods in the EPU domain, concerns about the accuracy of the bypass voiding calculation and its subsequent impact on the LPRM and OPRM response lead to a commitment to add additional penalties on to the OPRM setpoint calculation (Reference 15). The setpoint penalty was implemented since the Framatome Cycle-Specific DIVOM methodology was still under review by NRC. Since that time, the NRC has completed the review and approval of BAW-10255PA, Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code" (Reference 12). During the NRC review and approval of BAW-10255PA, Revision 2, the NRC explicitly reviewed the determination of bypass voiding and its impact on LPRM and OPRM response. The NRC staff review concluded that "the methods and procedures documented in the TR, and as supplemented by the responses to the NRC staff's RAI, represent a technically acceptable methodology to calculate DIVOM slope values." Further, "the slope values calculated by AREVA DIVOM Methodology are applicable to any D&S long term stability solution methodology that requires a setpoint calculation to suppress power oscillation before a specified acceptable fuel design limits are compromised." Application of Framatome's advanced methods to Susquehanna operation in the approved EPU operating domain is summarized in Enclosure 8a. Additional basis for removal of the OPRM amplitude setpoint penalty is provided in Enclosure 8a, Section 7.5. Since Framatome advanced methods are applicable to Susquehanna and the approved EPU operating domain, and based on NRC's prior approval of these methods for EPU conditions, specific penalties on the OPRM amplitude setpoint are not required.

3.5 Pin Power Uncertainty and Bundle Power Correlation Coefficient License Condition

The pin power distribution calculated within a fuel assembly is used to determine the CPR of that assembly during normal operation, normal operational transients and AOOs. Therefore, the uncertainties in pin power distribution calculation will impact the accuracy of the CPR calculation. The Susquehanna submittal for EPU conditions relied on pin-by-pin gamma scans for once-burnt ATRIUM-10 fuel bundles that experienced a softer spectral index (the ratio of the fast to thermal flux) than would be expected during EPU core conditions. This situation was caused by lower gadolinia loadings and enrichment in the supporting test data. As a result, the NRC concluded that the gamma scans, while supporting the original application of MICROBURN-B2, did not adequately justify the use of the previously established pin and bundle power uncertainties in EMF-2158 (Reference 14) for application to EPU cores. As such, an additional penalty on pin power uncertainty and bundle power correlation coefficient was required for licensing calculations in determination of the SLMCPR in accordance with

ANF-524(P)(A), "Critical Power Methodology for Boiling Water Reactors," using the uncertainty parameters associated with EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2."

As part of the development and licensing approval of AURORA-B for transient evaluation (Reference 8), AREVA provided additional gamma scan, TIP statistics, and LPRM data to support the use of the EMF-2158(P)(A) uncertainties at EPU conditions. As part of the review of the data provided to support the licensing of AURORA-B, the NRC found (in Section 3.3.2.4.5, page 42 of the Reference 8 SER), that "for the power-to-flow ratios examined, it is unlikely that a bundle power uncertainty exceeding the acceptance criteria of TR EMF-2158(P)(A) will be encountered at EFW [Extended Flow Window] conditions. Hence the uncertainties quantified for pin and bundle power distributions within TR EMF-2158(P)(A) remain applicable." The NRC staff "further concludes that imposition of a SLMCPR penalty for EPU conditions is not necessary."

The Reference 8 SER suggests uncertainty impacts be reviewed on a plant-specific basis such that conclusions drawn in the SER can be confirmed. In particular, the SER recommends confirmation on use of the MICROBURN-B2 based core monitoring system, plant operation within the existing power-to-flow database, and CASMO4/MICROBURN-B2 qualification to fuel designs for which Framatome has justified that the void quality correlation is valid at EPU and EFW conditions.

Application of Framatome's advanced methods to Susquehanna operation in the approved EPU operating domain is summarized in Enclosure 8a. Additional basis for removal of penalties on the pin power uncertainty and bundle power coefficient, including confirmation of these Reference 8 SER requirements, is provided in Enclosure 8a, Section 9.3. Since Framatome advanced methods are applicable to Susquehanna and the approved EPU operating domain, and based on NRC's prior approval of these methods for EPU conditions without penalties on EMF-2158(P)(A) pin power uncertainty and bundle power correlation coefficient, the existing Unit 1 License Condition 2.C.(38)(b) and Unit 2 License Condition 2.C.(22)(b) are no longer required.

4. <u>Regulatory Evaluation</u>

4.1 Applicable Regulatory Requirements/Criteria

<u>Title 10 Code of Federal Regulations (10 CFR) 50.36(c)(5)</u>

10 CFR 50.36(c)(5) states, "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner."

Conclusion

The COLR is required as a part of the reporting requirements specified in the SSES TS Administrative Controls section. The TS require the core operating limits to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and to be documented in the COLR. In addition, it requires the analytical methods used to determine the core operating limits to be those that have been previously reviewed and approved by the NRC, and specifically to be those described in TS 5.6.5.b. The proposed change ensures that these requirements are met.

10 CFR 50.46

10 CFR 50.46 establishes the acceptance criteria for the design basis LOCA. Paragraph (b)(1) requires the calculated maximum fuel element cladding temperature (i.e., PCT) to not exceed 2200°F. 10 CFR 50, Appendix K, establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA.

Conclusion

The use of the proposed analytical methods to determine core operating limits will continue to ensure that fuel performance during normal, transient, and accident conditions complies with these requirements. Specific AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits will be determined in conformance with 10 CFR 50, Appendix K requirements and documented in the COLR to ensure compliance with 10 CFR 50.46(b)(1).

General Design Criteria

Following approval of the proposed license amendment, SSES will maintain the ability to meet the applicable General Design Criteria (GDC) as outlined in 10 CFR 50, Appendix A. The applicable GDC are:

GDC-10, Reactor Design

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

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.

GDC-28, Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Conclusion

Susquehanna will use the proposed analytical methods to perform plant-specific analyses for APLHGR, MCPR, and LHGR. The limits on the APLHGR are specified to ensure that the PCT during the postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46. The SLMCPR ensures that sufficient conservatism exists in the operating limit MCPR such that, in the event of an AOO, there is a reasonable expectation that at least 99.9 percent of the fuel rods in the core will avoid boiling transition for the power distribution within the core including all uncertainties. Limits on the LHGR are specified to ensure that fuel thermal-mechanical design limits are not exceeded anywhere in the core during normal operation, including AOOs. Therefore, compliance with GDC 10 is maintained.

The proposed change will not replace nor change any of the previously approved stability methods (References 12, 32, and 33). The currently approved STAIF methodology (Reference 32) will continue to be used. No changes are being made in plant systems or procedures that are used to detect and suppress stability-related power oscillations. Therefore, compliance with GDC 12 is maintained.

The use of the proposed analytical methods for the CRDA calculations will continue to demonstrate compliance with GDC 28.

4.2 No Significant Hazards Considerations Analysis

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), is submitting a request for an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License numbers NPF-14 and NPF-22. The proposed change revises 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. Further, the proposed change revises the low pressure safety limit (SL) in TS 2.1.1.1 and TS 2.1.1.2 and removes the neutronic methods penalties on Oscillation Power Range Monitor (OPRM) amplitude setpoint, pin power distribution uncertainty, and bundle power correlation coefficient that were added during the Extended Power Uprate approved in Amendment 246/224; the penalties are no longer warranted with the introduction of the Advanced Framatome Methodologies.

Additionally, the proposed change would adopt Technical Specification Task Force (TSTF) Traveler TSTF-535, "Revise Shutdown Margin Definition to Address Advanced Fuel Designs." Specifically, the proposed change modifies the TS definition of "Shutdown Margin" (SDM) to require calculation of the SDM at a reactor moderator temperature of 68°F or a higher temperature that represents the most reactive state throughout the operating cycle. This change is needed to address new Boiling Water Reactor (BWR) fuel designs which may be more reactive at shutdown temperatures above 68°F.

Susquehanna has evaluated the proposed amendment against the standards in 10 CFR 50.92 and has determined that the operation of the SSES in accordance with the proposed amendment presents no significant hazards. Susquehanna's evaluation against each of the criteria in 10 CFR 50.92 follows.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   <u>Advanced Framatome Methodologies</u>

   The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The proposed change revises the list of NRC-approved analytical methods used to establish core operating limits, adjusts the low pressure SL, and eliminates neutronic methods penalties on OPRM amplitude setpoint, pin power distribution uncertainty, and bundle power correlation coefficient. The change does not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Since no individual precursors of an

accident are affected, the proposed amendments do not increase the probability of a previously analyzed event.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. The proposed change revises the list of NRC-approved analytical methods used to establish core operating limits, adjusts the low pressure SL, and eliminates neutronic methods penalties on OPRM amplitude setpoint, pin power distribution uncertainty, and bundle power correlation coefficient. The changes in methodology do not alter the assumptions of accident analyses. Based on the above, the proposed amendments do not increase the consequences of a previously analyzed accident.

<u>TSTF-535</u>

The proposed change revises the definition of SDM. SDM is not an initiator of any accident previously evaluated. Accordingly, the proposed change to the definition of SDM has no effect on the probability of any accident previously evaluated. SDM is an assumption in the analysis of some previously evaluated accidents and inadequate SDM could lead to an increase in the consequences for those accidents. However, the proposed change revises the SDM definition to ensure that the correct SDM is determined for all fuel types at all times during the fuel cycle. As a result, the proposed change does not adversely affect the consequences of any accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

<u>Advanced Framatome Methodologies</u>

Creation of the possibility of a new or different kind of accident requires creating one or more new accident precursors. New accident precursors may be created by modifications of plant configuration, including changes in allowable modes of operation. The proposed change revises the list of NRC-approved analytical methods used to establish core operating limits, adjusts the low pressure SL, and eliminates neutronic methods penalties on OPRM amplitude setpoint, pin power distribution uncertainty, and bundle power correlation coefficient. The proposed amendments do not involve any plant configuration modifications or changes to allowable modes of operation thereby ensuring no new accident precursors are created.

TSTF-535

The proposed change revises the definition of SDM. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operations. The change does not alter the assumptions made in the safety analysis regarding SDM.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

Advanced Framatome Methodologies

The proposed change revises the list of NRC-approved analytical methods used to establish core operating limits, adjusts the low pressure SL, and eliminates neutronic methods penalties on OPRM amplitude setpoint, pin power distribution uncertainty, and bundle power correlation coefficient. The proposed change will ensure that the current level of fuel protection is maintained by continuing to ensure that the fuel design safety criteria are met. The proposed changes will not impact the capabilities of the existing NRC-approved CPR correlations and ensure valid CPR calculations including applicable uncertainties for AOOs defined in the FSAR. The proposed amendment would have no impact on the structural integrity of the fuel cladding, reactor coolant pressure boundary, or containment structure. Based on the above considerations, the proposed amendment would not degrade the confidence in the ability of the fission product barriers to limit the level of radiation to the public.

TSTF-535

The proposed change revises the definition of SDM. The proposed change does not alter the manner in which SLs, limiting safety system settings or limiting conditions for operation are determined. The proposed change ensures that the SDM assumed in determining SLs, limiting safety system settings, or limiting conditions for operation is correct for all BWR fuel types at all times during the fuel cycle.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluation, Susquehanna concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

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

5. Environmental Consideration

Susquehanna has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. References

1. NRC letter to Susquehanna, "Issuance of Amendment Regarding the 13-Percent Extended Power Uprate (TAC Nos. MD3309 and MD3310)," dated January 30, 2008 (ADAMS Accession No. ML080020201)

2. Framatome Topical Report ANP-10335P-A, "ACE/ATRIUM 11 Critical Power Correlation," Revision 0, dated May 2018

3. AREVA Topical Report EMF-2209(P)(A), "SPCB Critical Power Correlation," Revision 3, dated September 2009

4. Framatome Topical Report BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, dated April 2008

5. Framatome Topical Report BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," Revision 0, Supplement 2P-A, dated August 2018

6. Framatome Topical Report BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," Revision 0, Supplement 1P-A, dated April 2017

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

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

9. Framatome Topical Report ANP-10332P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident (LOCA)," Revision 0, dated March 2019

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

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

12. Framatome Topical Report BAW-10255PA, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," Revision 2, dated May 2008

13. Advanced Nuclear Fuels Topical Report ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," Revision 2, Supplements 1 and 2, dated November 1990

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

15. Susquehanna letter to NRC, "Proposed License Amendment No. 285 for Unit 1 Operating License No. NPF-14 and Proposed License Amendment No. 253 for Unit 2 Operating License No. NPF-22 Constant Pressure Power Uprate Application – Supplement," dated November 30, 2007 (ADAMS Accession No. ML073450822)

16. NUREG-1433, "Standard Technical Specifications – General Electric Plants (BWR/4)," Revision 4, Volume 1, dated April 2012 (ADAMS Accession No. ML12104A192)

17. NRC letter to Susquehanna, "Issuance of Amendment Re: Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) Implementation (TAC Nos. MC9040 and MC9041)," dated March 23, 2007 (ADAMS Accession No. ML070720675)

18. Framatome letter to NRC, "Informational Transmittal of ANP-2637P, Revision 8, 'Boiling Water Reactor Licensing Compendium,'" dated June 5, 2019 (ADAMS Accession No. ML19158A093)

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

20. Framatome Topical Report EMF-93-177(P)(A), "Mechanical Design for BWR Fuel Channels," Revision 1, dated August 2005

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

22. Exxon Nuclear Company Topical Report XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors - Application of the ENC Methodology to BWR Reloads," Volume 4 Revision 1, dated April 1986

23. Exxon Nuclear Company Topical Report XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Revision 1, dated November 1982

24. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 4.2, "Fuel System Design," Revision 3, dated March 2007 (ADAMS Accession No. ML070740002)

25. Draft Regulatory Guide DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," dated November 2016 (ADAMS Accession No. ML16124A200)

26. AREVA letter to NRC, "Response to Request for Additional Information Regarding Topical Report ANP-10333P, Revision 0, 'AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA),'" dated April 6, 2017 (ADAMS Accession No. ML17100A170)

27. NRC letter to Susquehanna, "Issuance of Amendment Regarding Minimum Critical Power Ratio Safety Limit and Reference Changes (TAC No. MC9187)," dated March 20, 2006 (ADAMS Accession No. ML060730355)

28. NRC letter to Susquehanna, "Issuance of Amendment Regarding Minimum Critical Power Ratio Safety Limit and Reference Changes (TAC No. MC4431)," dated February 28, 2005 (ADAMS Accession No. ML050590044)

29. Siemens Power Corporation letter to NRC, "Revisions to Attachment 1 of Letter NRC:99:030, Request for Concurrence on SER Clarifications," dated October 12, 1999 (Legacy ADAMS Accession No. 9910190133)

30. NRC letter to Siemens Power Corporation, "Siemens Power Corporation Re: Request for Concurrence on Safety Evaluation Report Clarifications (TAC No. MA6160)", dated May 31, 2000 (ADAMS Accession No. ML003719373)

31. Framatome letter to NRC, "Informational Transmittal of ANP-3653P Revision 0, 'Fuel Design Evaluation for ATRIUM 11 BWR Reload Fuel,' and ANP-2637P Revision 7, 'Boiling Water Reactor Licensing Methodology Compendium,'" dated September 18, 2018 (ADAMS Accession No. ML18264A015)

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

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

Enclosure 2 of PLA-7783

Marked-Up Operating License Pages

Revised Operating License Pages

Unit 1 Operating License Page
18

Unit 2 Operating License Page
14

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) Neutronic Methods

(a) Not Used~~An OPRM amplitude setpoint penalty will be applied to account for a reduction in thermal neutrons around the LPRM detectors caused by transients that increase voiding. This penalty will reduce the OPRM scram setpoint according to the methodology described in Response No. 3 of the operating licensee's letter, PLA-6306, dated November 30, 2007. This penalty will be applied until NRC evaluation determines that a penalty to account for this phenomenon is not warranted.~~

(b) Not Used~~For SSES SLMCPR, a conservatively adjusted pin power distribution uncertainty and bundle power correlation coefficient will be applied as stated in Response No. 4 of the operating licensee's letter, PLA-6306, dated November 30, 2007, when performing the analyses in accordance with ANF-524(P)(A) "Critical Power Methodology for Boiling Water Reactors," using the uncertainty parameters associated With EMF-2158(P)(A) "Siemens Power Corporations Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO 4/MICROBURN-B2.~~

(22) Neutronic Methods

(a) Not Used~~An OPRM amplitude setpoint penalty will be applied to account for a reduction in thermal neutrons around the LPRM detectors caused by transients that increase voiding. This penalty will reduce the OPRM scram setpoint according to the methodology described in Response No. 3 of the operating licensee's letter, PLA-6306, dated November 30, 2007. This penalty will be applied until NRC evaluation determines that a penalty to account for this phenomenon is not warranted.~~

(b) Not Used~~For SSES SLMCPR, a conservatively adjusted pin power distribution uncertainty and bundle power correlation coefficient will be applied as stated in Response No. 4 of the operating licensee's letter, PLA-6306, dated November 30, 2007, when performing the analyses in accordance with ANF-524(P)(A) "Critical Power Methodology for Boiling Water Reactors," using the uncertainty parameters associated With EMF-2158(P)(A) "Siemens Power Corporations Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO 4/MICROBURN-B2.~~

(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 CL TP, 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 Band 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

Enclosure 3 of PLA-7783

Revised (Clean) Operating License Pages

Revised Operating License Pages

Unit 1 Operating License Page
18

Unit 2 Operating License Page
14

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

(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 CL TP, 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 Band 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

Enclosure 4 of PLA-7783

Marked-Up Technical Specification Pages

Revised Technical Specifications Pages

Unit 1 TS Pages
1.1-6, 2.0-1, 5.0-22, and 5.0-23

Unit 2 TS Pages
1.1-6, 2.0-1, 5.0-22, and 5.0-23

1.1 Definitions

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3952 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during η Surveillance Frequency intervals, where η is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~557~~ 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 ≥ ~~557~~ 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 ~~XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company.~~

10. Not Used ~~ANF-524(P)(A), "ANF Critical Power Methodology for Boiling~~

~~Water Reactors," Advanced Nuclear Fuels Corporation~~

11. Not Used ~~ANF-913(P)(A), "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation.~~

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 ~~NE-092-001A, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company.~~

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.

1.1 Definitions

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3952 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~557~~ 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 ≥ ~~557~~ 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 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 ~~XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company.~~

10. Not Used~~ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation~~

5.6 Reporting Requirements

11. Not Used~~ANF-913(P)(A), "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation.~~

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~~NE-092-001A, "Licensing Topical Report for Power Uprate with Increased Core Flow," Pennsylvania Power & Light Company~~

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.

Enclosure 5 of PLA-7783

Revised (Clean) Technical Specification Pages

Revised Technical Specifications Pages

Unit 1 TS Pages
1.1-6, 2.0-1, 5.0-22, and 5.0-23

Unit 2 TS Pages
1.1-6, 2.0-1, 5.0-22, and 5.0-23

1.1 Definitions

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3952 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during η Surveillance Frequency intervals, where η is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

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.

1.1 Definitions

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3952 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

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 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.

Enclosure 6 of PLA-7783

Marked-Up Technical Specification Bases Pages

Revised Technical Specifications Bases Pages

Unit 1 TS Bases Pages
2.0-1, 2.0-2, 2.0-3, 2.0-4, 2.0-5, 3.1-6, 3.1-38, 3.2-1, 3.2-2, 3.2-3, 3.2-4, 3.2-5, 3.2-9, 3.2-10, 3.2-11, 3.2-13, 3.4-3, 3.5-4, 3.5-8, 3.5-15, 3.10-32, and 3.10-38

Unit 2 TS Bases Pages
2.0-1, 2.0-2, 2.0-3, 2.0-4, 2.0-5, 3.1-6, 3.1-38, 3.2-1, 3.2-2, 3.2-3, 3.2-4, 3.2-5, 3.2-8, 3.2-9, 3.2-10, 3.2-12, 3.4-3, 3.5-4, 3.5-8, 3.5-16, 3.10-33, and 3.10-39

(Provided for Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for ATRIUM 10 and ATRIUM 11~~AREVA NP~~ fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

The use of the SPCB (Reference 4) correlation is valid for critical power calculations with ATRIUM 10 fuel at pressures ≥ 571.4 psia (conservatively bounded by 575 psig) and bundle mass fluxes > 0.087×10^6 lb/hr-ft^2.

The use of the ACE/ATRIUM 11 (Reference 6) correlation is valid for critical power calculations with ATRIUM 11 fuel at pressures ≥ 588.8 psia (conservatively bounded by 575 psig) with no minimum bundle mass flux.

For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition.

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

~~For the AREVA NP ATRIUM-10 design, the minimum bundle flow is > 28 x 10^3 lb/hr. For the AREVA NP ATRIUM-10 fuel design, the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various AREVA NP and GE fuel designs at pressures from 14.7 psia to 1400 psia indicate the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 23% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 2.8, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 23% RTP for reactor pressures < 557 psig is conservative and for conditions of lesser power would remain conservative.~~
For ATRIUM 10 and ATRIUM 11 fuel, the minimum bundle flow is > 28 x 10^3 lb/hr, and the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.24 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various fuel designs at pressures from 14.7 psia to 1400 psia indicate that the fuel assembly critical power at 0.24 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 23% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 2.8, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 23% RTP for reactor pressures < 575 psig is conservative and for conditions of lesser power would remain conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty in the critical power correlation. References 2, 4, 5, and ~~5~~6 describe the methodology used in determining the MCPR SL.

The SPCB and ACE/ATRIUM 11 critical power correlations ~~is~~are based on a significant body of practical test data. As long as the core pressure and flow are within the range of validity of the correlations (refer to Section B.2.1.1.1), the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate

the number of rods in boiling transition. These conservatisms and the inherent accuracy of the SPCB and ACE/ATRIUM 11 correlations provide a reasonable degree of assurance that during sustained operation at the MCPR SL there would be no transition boiling in the core.

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised.

Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

~~AREVA NP ATRIUM-10 fuel is monitored using the SPCB Critical Power Correlation.~~ The effects of channel bow on MCPR are explicitly included in the calculation of the MCPR SL. Explicit treatment of channel bow in the MCPR SL addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

Monitoring required for compliance with the MCPR SL is specified in LCO 3.2.2, Minimum Critical Power Ratio.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

BASES

SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of regulatory limits. Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ~~ANF-524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.~~ANP-10307PA, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” (as identified in the COLR).

3. ~~Deleted~~Not Used.

4. EMF-2209(P)(A), "SPCB Critical Power Correlation,” AREVA NP, ~~[See Core Operating Limits Report for Revision Level~~(as identified in the COLR)~~]~~.

5. EMF-2158(P)(A), ~~Revision 0,~~ “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2,” ~~October 1999~~(as identified in the COLR).

6. ANP-10335P-A, “ACE/ATRIUM 11 Critical Power Correlation,” (as identified in the COLR).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

the highest worth control rod is determined by analysis or testing.

Local critical tests require the withdrawal of control rods in a sequence that is not in conformance with BPWS. This testing would therefore require re-programming or bypassing of the rod worth minimizer to allow the withdrawal of control rods not in conformance with BPWS, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing - Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each planned in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Section 15.

3. ~~XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, “Exxon Nuclear Methodology for Boiling Water Reactors,” Exxon Nuclear Company, March 1983.~~ANP-10333P-A, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA),” (as identified in the COLR).

4. FSAR, Section 15.4.1.1.

BASES

ACTIONS

B.1 and B.2 (continued)

of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is periodically verified to be in compliance with the BPWS to ensure the assumptions of the CRDA analyses are met. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The RWM which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. ~~XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.~~ ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," (as identified in the COLR).

2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR 100.11.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

9. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

LOCA calculations for the ATRIUM 10 and ATRIUM 11 fuel designs were performed. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 7, 8 and 9.~~SPC performed LOCA calculations for the SPC ATRIUM™-10 fuel design. The analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 are presented in References 3, 4, 5, and 6 for the SPC analysis. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 3 through 9.~~

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the Peak Cladding Temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation limits of 10 CFR 50.46. The analyses are performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. ~~A complete discussion of the analysis codes are provided in References 3, 4, 5, and 6 for the SPC analysis.~~ The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within the assembly.

The specific analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 for the ATRIUM 10 fuel design are presented in Reference 3 and 4. The specific analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 for the ATRIUM-11 fuel design are presented in Reference 11.

APLHGR limits are developed as a function of fuel type and exposure. ~~The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g., Maximum Extended Load Line Limit Analysis (MELLLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)).~~ LOCA analyses were performed for the regions of the power/flow map bounded by the rod line that runs through 100% RTP and maximum core flow and the upper boundary of the MELLLA region.

The MELLLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required. The results of the analysis demonstrate the PCTs are within the 10 CFR 50.46 limit, and that APLHGR multipliers as a function of core flow are not required.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The SPC LOCA analyses consider the delay in Low Pressure Coolant Injection (LPCI) availability when the unit is operating in the Suppression Pool Cooling Mode. The delay in LPCI availability is due to the time required to realign valves from the Suppression Pool Cooling Mode to the LPCI mode. The results of the analyses demonstrate that the PCTs are within the 10 CFR 50.46 limit.~~

Finally, the ~~SPC~~ LOCA analyses were performed for Single-Loop Operation. The results of the ~~SPC~~ analysis for ATRIUM~~TM~~-10 fuel shows that an APLHGR limit which is 0.8 times the two-loop APLHGR limit meets the 10 CFR 50.46 acceptance criteria, and that the PCT is less than the limiting two-loop PCT. The results of the analyses for ATRIUM 11 fuel show that an APLHGR limit which is 0.8 times the two-loop APLHGR limits meets the 10 CFR 50.46 acceptance criteria, and that the PCT is less than the limiting two-loop PCT.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 10).

LCO

The APLHGR limits specified in the COLR are the result of the DBA analyses.

APPLICABILITY

The APLHGR limits are primarily derived from LOCA analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. At THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a DBA occurring simultaneously with the APLHGR out of specification.

BASES

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 24 hours after THERMAL POWER is ≥ 23% RTP and periodically thereafter. Additionally, APLHGRs must be calculated prior to exceeding 44% RTP unless performed in the previous 24 hours. APLHGRs are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the APLHGRs must be calculated prior to exceeding 44% RTP. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Not used.

2. Not used.

3. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP, (as identified in the COLR).

4. ~~ANF-CC-33(P)(A) Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.~~EMF-2292(P)(A), "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

5. Not Used ~~XN-CC-33(P)(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," November 1975.~~

BASES

References
(continued)

6. Not Used ~~XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.~~

7. FSAR, Chapter 4.

8. FSAR, Chapter 6.

9. FSAR, Chapter 15.

10. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

11. ANP-10332P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," (as identified in the COLR).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 5, 7, and 10 for ATRIUM 10 fuel design analysis and references 2, 3, 5, 7, 10, and 12 through 15 for ATRIUM 11 fuel designs. ~~References 2 through 10~~. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power

BASES

REFERENCES (continued)

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

4. Not Used ~~ANF-913(P)(A) Volume 1 Revision 1 and Volume Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.~~

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

6. Not Used ~~NE-092-001, Revision 1, "Susquehanna Steam Electric Station Units 1 & 2: Licensing Topical Report for Power Uprate with Increased Core flow," December 1992, and NRC Approval Letter: Letter from T. E. Murley (NRC) to R. G. Byram (PP&L), "Licensing Topical Report for Power Uprate With Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC Nos. M83426 and M83427)," November 30, 1993.~~

7. EMF-2209(P)(A), "SPCB Critical Power Correlation," ~~Framatome~~ AREVA NP, ~~[See Core Operating Limits Report for Revision Level~~(as identified in the COLR)~~]~~.

8. Not Used. ~~XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.~~

9. Not Used ~~XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.~~

10. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, ~~[See Core Operating Limits Report for Revision Level]~~(as identified in the COLR).

11. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

12. ANP-10300P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," (as identified in the COLR).

13. BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," (as identified in the COLR).

14. BAW-10247P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," (as identified in the COLR).

15. ANP-10335P-A, "ACE/ATRIUM-11 Critical Power Correlation," (as identified in the COLR).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, and 4. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of regulatory limits. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. Transient evaluations were also performed. Reference 4 establishes LHGR acceptance criteria on strain and fuel overheating (fuel centerline melt) for both normal operation and anticipated operational occurrences. ~~A separate evaluation was performed to determine the limits of LHGR during anticipated operational occurrences. This limit,~~

BASES

APPLICABLE SAFETY ANALYSES (continued)	~~Protection Against Power Transients (PAPT), defined in Reference 4 provides the acceptance criteria for LHGRs calculated in evaluation of the AOOs.~~ The LHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).
LCO	The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.
APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 23% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 23% RTP.
ACTIONS	<u>A.1</u> If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to

BASES

REFERENCES (continued)

4. ANF-89-98(P)(A) ~~Revision 1 and Revision 1 Supplement 1~~, "Generic Mechanical Design Criteria for BWR Fuel Designs," ~~Advanced Nuclear Fuels Corporation, May 1995~~(as identified in the COLR).

5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided that the APLHGR limits for ~~SPC ATRIUM™-10~~ATRIUM 10 and ATRIUM 11 fuel are~~is~~ modified.

The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR, LHGR, and MCPR limits for single loop operation are specified in the COLR. The APRM Simulated Thermal Power-High setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." In addition, a restriction on recirculation pump speed is incorporated to address reactor vessel internals vibration concerns and assumptions in the event analysis.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLGHR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE"), LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and APRM Simulated Thermal Power-High setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the safety analysis assumptions. Furthermore, restrictions are placed on recirculation pump speed to ensure the initial assumption of the event analysis are maintained.

BASES

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5, 6, and 7. The required analyses and assumptions are defined in Reference 8. The results of these analyses are also described in Reference 9.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 10), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

~~SPC~~ The fuel vendor performed LOCA calculations for ~~-~~the ~~SPC~~ ATRIUM~~™-~~10 and ATRIUM 11 fuel designs. The limiting single failures for the ~~SPC~~ analyses are discussed in Reference ~~11~~9. The LOCA analyses examine both recirculation pipe breaks and non-recirculation pipe breaks. For the recirculation pipe breaks, breaks on both the discharge and suction side of the recirculation pump are performed for two geometries; double-ended guillotine and split break.

~~For a large break LOCA, the SPC analyses identify the recirculation loop suction piping as the limiting break location. The SPC analysis identifies the failure of the LPCI injection valve into the intact recirculation loop as the most limiting single failure.~~

~~For a small break LOCA, the SPC analyses identify the recirculation loop discharge piping as the limiting break location, and a battery failure as the most severe single failure.~~The LOCA calculations demonstrate that the limiting fuel type (highest PCT) is ATRIUM 10 fuel. The most limiting (highest PCT) break is a double-ended guillotine break in the recirculation pump suction piping. The limiting single failure is the failure of the LPCI injection valve in the intact recirculation loop to open.

One ADS valve failure is analyzed as a limiting single failure for events requiring ADS operation. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of the NRC Policy Statement (Ref. 15).

BASES

ACTIONS (continued)

F.1

The LCO requires six ADS valves to be OPERABLE in order to provide the ADS function. Reference ~~11~~ 9 contains the results of an analysis that evaluated the effect of one ADS valve being out of service. Per this analysis, operation of only five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced, because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

G.1 and G.2

If Condition A or Condition B exists in addition to one inoperable ADS valve, adequate core cooling is ensured by the OPERABILITY of HPCI and the remaining low pressure ECCS injection/spray subsystem. However, overall ECCS reliability is reduced because a single active component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure system (ADS) and a low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS subsystem or the ADS valve to OPERABLE status. This Completion Time is based on a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

H.1 and H.2

If any Required Action and associated Completion Time of Condition D, E, F, or G is not met, or if two or more ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

I.1

When multiple ECCS subsystems are inoperable, as stated in Condition I, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.13

This SR ensures that the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is less than or equal to the maximum value assumed in the accident analysis. Response Time testing acceptance criteria are included in Reference 13. This SR is modified by a Note that allows the instrumentation portion of the response time to be assumed to be based on historical response time data and therefore, is excluded from the ECCS RESPONSE TIME testing. This is allowed since the instrumentation response time is a small part of the ECCS RESPONSE TIME (e.g., sufficient margin exists in the diesel generator start time when compared to the instrumentation response time) (Ref. 14).

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 6.3.2.2.3.
2. FSAR, Section 6.3.2.2.4.
3. FSAR, Section 6.3.2.2.1.
4. FSAR, Section 6.3.2.2.2.
5. FSAR, Section 15.2.~~4~~8.
6. FSAR, Section 15.~~2.5~~6.4.
7. FSAR, Section 15.~~2.6~~6.5.
8. 10 CFR 50, Appendix K.
9. FSAR, Section 6.3.3.
10. 10 CFR 50.46.
11. ~~FSAR, Section 6.3.3.~~Not Used
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.10.7.2 When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.
REFERENCE	1. FSAR 15.4.9 2. ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," (as identified in the COLR). ~~XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCE

1. ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," (as identified in the COLR) ~~XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.~~

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for ~~AREVA NP~~ATRIUM 10 and ATRIUM 11 fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

The use of the SPCB (Reference 4) correlation is valid for critical power calculations with ATRIUM 10 fuel at pressures ≥ 571.4 psia (conservatively bounded by 575 psig) and bundle mass fluxes > 0.087 x 10^6 lb/hr-ft^2. ~~for SPCB.~~

The use of the ACE/ATRIUM 11 (Reference 6) correlation is valid for critical power calculation with ATRIUM 11 fuel at pressures ≥ 588.8 psia (conservatively bounded by 575 psig) with no minimum bundle mass flux.

For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For ATRIUM 10 and ATRIUM 11 fuel, the minimum bundle flow is > 28 x 10^3 lb/hr and the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.24 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various fuel designs at pressures from 14.7 psia to 1400 psia indicate that the fuel assembly critical power at 0.24 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 23% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 2.8, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 23% RTP for reactor pressures < 575 psig is conservative and for conditions of lesser power would remain conservative.
~~For the AREVA NP ATRIUM-10 design, the minimum bundle flow is > 28 x 10^3 lb/hr. For AREVA NP ATRIUM-10 fuel design, the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various AREVA NP~~

~~and GE fuel designs at pressures from 14.7 psia to 1400 psia indicate the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 23% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 2.8, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 23% RTP for reactor pressures < 557 psig is conservative and for conditions of lesser power would remain the same.~~

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty in the critical power correlation. References 2, 4, 5, and ~~5~~6 describe the methodology used in determining the MCPR SL.

The SPCB and ACE/ATRIUM 11 critical power correlations ~~is~~are based on a significant body of practical test data. As long as the core pressure and flow are within the range of validity of the correlation (refer to Section B 2.1.1.1), the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the SPCB and ACE/ATRIUM 11 correlations provide a reasonable degree of assurance that during sustained operation at the MCPR SL there would be no transition boiling in the core. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised.

Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

~~AREVA NP ATRIUM-10 fuel is monitored using the SPCB Critical Power Correlation.~~ The effects of channel bow on MCPR are explicitly included in the calculation of the MCPR SL. Explicit treatment of channel bow in the MCPR SL addresses the concerns of the NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

Monitoring required for compliance with the MCPR SL is specified in LCO 3.2.2, Minimum Critical Power Ratio.

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height.

The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of regulatory limits. Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ~~ANF-524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.~~ANP-10307PA, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” (as identified in the COLR)

3. ~~Deleted~~Not Used.

4. EMF-2209(P)(A), “SPCB Critical Power Correlation,” AREVA NP,

~~[See Core Operating Limits Report for Revision Level]~~(as identified in the COLR).

BASES

REFERENCES (continued)

5. EMF-2158(P)(A) ~~Revision 0~~, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2,” ~~October 1999~~(as identified in the COLR).

6. ANP-10335P-A, “ACE/ATRIUM 11 Critical Power Correlation,” (as identified in the COLR)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

the highest worth control rod is determined by analysis or testing.

Local critical tests require the withdrawal of control rods in a sequence that is not in conformance with BPWS. This testing would therefore require re-programming or bypassing of the rod worth minimizer to allow the withdrawal of control rods not in conformance with BPWS, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each planned in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Section 15.

~~3. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.~~

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

4. FSAR, Section 15.4.1.1.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is periodically verified to be in compliance with the BPWS to ensure the assumptions of the CRDA analyses are met. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The RWM provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," (as identified in the COLR).~~XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.~~

2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR 100.11.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

9. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

LOCA calculations for the ATRIUM 10 and ATRIUM 11 fuel designs were performed. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 7, 8, and 9. ~~SPC performed LOCA calculations for the SPC ATRIUM™-10 fuel design. The analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 are presented in References 3, 4, 5, and 6 for the SPC analysis. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 3 through 9.~~

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the Peak Cladding Temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation limits of 10 CFR 50.46. The analyses are performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. ~~A complete discussion of the analysis codes are provided in References 3, 4, 5, and 6 for the SPC analysis.~~ The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within the assembly.

The specific analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 for the ATRIUM 10 fuel design are presented in References 3 and 4. The specific analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 for the ATRIUM 11 fuel design are presented in Reference 11.

APLHGR limits are developed as a function of fuel type and exposure. LOCA analyses were performed for the regions of the power/ flow map bounded by the rod line that runs through 100% RTP and maximum core flow and the upper boundary of the MELLLA region. The MELLLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required. The results of the analysis demonstrate the PCTs

are within the 10 CFR 50.46 limit, and that APLHGR multipliers as a function of core flow are not required. ~~The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g., Maximum Extended Load Line Limit Analysis (MELLLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)). LOCA analyses were performed for the regions of the power/flow map bounded by the rod line that runs through 100% RTP and maximum core flow and the upper boundary of the MELLLA region. The MELLLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required. The results of the analysis demonstrate the PCTs are within the 10 CFR 50.46 limit, and that APLHGR multipliers as a function of core flow are not required.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The SPC LOCA analyses consider the delay in Low Pressure Coolant Injection (LPCI) availability when the unit is operating in the Suppression Pool Cooling Mode. The delay in LPCI availability is due to the time required to realign valves from the Suppression Pool Cooling Mode to the LPCI mode. The results of the analyses demonstrate that the PCTs are within the 10 CFR 50.46 limit.~~

Finally, the ~~SPC~~ LOCA analyses were performed for Single-Loop Operation. The results of the ~~Framatome~~ analysis for ATRIUM~~TM-~~10 fuel shows that an APLHGR limit which is 0.8 times the two-loop APLHGR limit meets the 10 CFR 50.46 acceptance criteria, and that the PCT is less than the limiting two-loop PCT. The results of the analyses for ATRIUM 11 fuel show that an APLHGR limit which is 0.8 times the two-loop APLHGR limits meets the 10 CFR 50.46 acceptance criteria, and that the PCT is less than the limiting two-loop PCT.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 10).

LCO

The APLHGR limits specified in the COLR are the result of the DBA analyses.

APPLICABILITY

The APLHGR limits are primarily derived from LOCA analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. At THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a DBA occurring simultaneously with the APLHGR out of specification.

BASES

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 24 hours after THERMAL POWER is ≥ 23% RTP and periodically thereafter. Additionally, APLHGRs must be calculated prior to exceeding 44% RTP unless performed in the previous 24 hours. APLHGRs are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the APLHGRs must be calculated prior to exceeding 44% RTP. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Not Used

2. Not Used

3. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP (as identified in the COLR).

4. EMF-2292(P)(A) ~~Revision 0~~, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients,~~.~~" (as identified in the COLR).

5. Not Used~~XN-CC-33(P)(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," November 1975.~~

BASES

REFERENCES (continued)

6. Not Used ~~XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.~~

7. FSAR, Chapter 4.

8. FSAR, Chapter 6.

9. FSAR, Chapter 15.

10. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

11. ANP-10332P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios," (as identified in the COLR).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 5, 7, and 10 for ATRIUM 10 fuel design analysis and References 2, 3, 5, 7, 10, and 12 through 15 for ATRIUM 11 fuel designs ~~References 2 through 10~~. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency These analyses may also consider other

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.2 (continued)

Determining MCPR operating limits based on interpolation between scram insertion times is not permitted. The average measured scram times and corresponding MCPR operating limit must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3 and SR 3.1.4.4 because the effective scram times may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in average measured scram times expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

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

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

4. Not Used ~~ANF-913(P)(A) Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.~~

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

6. Not Used ~~NE-092-001, Revision 1, "Susquehanna Steam Electric Station Units 1 & 2: Licensing Topical Report for Power Uprate with Increased Core Flow," December 1992, and NRC Approval Letter: Letter from T. E. Murley (NRC) to R. G. Byram (PP&L), "Licensing Topical Report for Power Uprate With Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC Nos. M83426 and M83427)," November 30, 1993.~~

7. EMF-2209(P)(A), "SPCB Critical Power Correlation," AREVA NP (~~See Core Operating Limits Report for Revision Level.~~as identified in the COLR).

BASES

Reference (continued)

8. Not Used.~~XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.~~

9. Not Used ~~XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.~~

10. ANF-1358(P)(A)~~ Revision 3~~, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," ~~Advanced Nuclear Fuels Corporation, September 2005~~(as identified in the COLR).

11. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

12. ANP-10300P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," (as identified in the COLR).

13. BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," (as identified in the COLR).

14. BAW-10247P-A Supplement 2P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," (as identified in the COLR).

15. ANP-10335P-A, "ACE/ATRIUM-11 Critical Power Correlation," (as identified in the COLR).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, and 4. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of regulatory limits. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. Transient evaluations were also performed. Reference 4 establishes LHGR acceptance criteria on strain and fuel overheating (fuel centerline melt) for both normal operation and anticipated operational occurrences. ~~A separate evaluation was performed to determine the limits of LHGR during anticipated operational occurrences. This limit, Protection Against Power Transients (PAPT), defined in reference 4, provides the acceptance criteria for LHGRs calculated in evaluation of the AOOs.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 24 hours after THERMAL POWER is ≥ 23% RTP and periodically thereafter. Additionally, LHGRs must be calculated prior to exceeding 44% RTP unless performed in the previous 24 hours. The LHGR is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels and because the LHGRs must be calculated prior to exceeding 44% RTP. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. FSAR, Section 4.
2. FSAR, Section 5.
3. NUREG-0800, Section II.A.2(g), Revision 2, July 1981.
4. ANF-89-98(P)(A) ~~Revision 1 and Revision 1 Supplement 1~~, "Generic Mechanical Design Criteria for BWR Fuel Design," ~~Advanced Nuclear Fuels Corporation, May 1995~~(as identified in the COLR).
5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided that the APLHGR limits for ~~SPC ATRIUM™-10~~ATRIUM 10 and ATRIUM 11 fuel ~~is~~ are modified.

The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR, LHGR, and MCPR limits for single loop operation are specified in the COLR. The APRM Simulated Thermal Power-High setpoint is in LCO 3.3.1.1, “Reactor Protection System (RPS) Instrumentation.” In addition, a restriction on recirculation pump speed is incorporated to address reactor vessel internals vibration concerns and assumptions in the event analysis.

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement (Ref. 5).

LCO

Two recirculation loops are required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLGHR limits (LCO 3.2.1, “AVERAGE PLANAR LINEAR HEAT GENERATION RATE”), LHGR limits (LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR)”), MCPR limits (LCO 3.2.2, “MINIMUM CRITICAL POWER RATIO (MCPR)”), and APRM Simulated Thermal Power-High setpoint (LCO 3.3.1.1) may be applied to allow continued operation consistent with the safety analysis assumptions. Furthermore, restrictions are placed on recirculation pump speed to ensure the initial assumption of the event analysis are maintained.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5, 6, and 7. The required analyses and assumptions are defined in Reference 8. The results of these analyses are also described in Reference 9.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 10), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

~~SPC~~ The fuel vendor performed LOCA calculations for the ~~SPC~~ ATRIUM~~™~~ 10 and ATRIUM 11 fuel designs. The limiting single failures for the ~~SPC~~ analyses are discussed in Reference ~~11~~9. The LOCA ~~calculations~~ analyses examine both recirculation pipe and non-recirculation pipe breaks. For the recirculation pipe breaks, breaks on both the discharge and suction side of the recirculation pump are performed for two geometries; double-ended guillotine break and split break.

The LOCA calculations demonstrate the limiting fuel type (highest PCT) is ATRIUM 10 fuel. The ~~LOCA calculations demonstrate that the~~ most limiting (highest PCT) break is a double-ended guillotine break in the recirculation pump suction piping. The limiting single failure is the failure of the LPCI injection valve in the intact recirculation loop to open.

One ADS valve failure is analyzed as a limiting single failure for events requiring ADS operation. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of the NRC Policy Statement (Ref. 15).

BASES

ACTIONS (continued)

F.1

The LCO requires six ADS valves to be OPERABLE in order to provide the ADS function. Reference ~~11~~ 9 contains the results of an analysis that evaluated the effect of one ADS valve being out of service. Per this analysis, operation of only five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced, because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

G.1 and G.2

If Condition A or Condition B exists in addition to one inoperable ADS valve, adequate core cooling is ensured by the OPERABILITY of HPCI and the remaining low pressure ECCS injection/spray subsystem. However, overall ECCS reliability is reduced because a single active component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure system (ADS) and a low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS subsystem or the ADS valve to OPERABLE status. This Completion Time is based on a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

H.1 and H.2

If any Required Action and associated Completion Time of Condition D, E, F, or G is not met, or if two or more ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

I.1

When multiple ECCS subsystems are inoperable, as stated in Condition I, LCO 3.0.3 must be entered immediately.

BASES

REFERENCES

1. FSAR, Section 6.3.2.2.3.

2. FSAR, Section 6.3.2.2.4.

3. FSAR, Section 6.3.2.2.1.

4. FSAR, Section 6.3.2.2.2.

5. FSAR, Section 15.2.8.

6. FSAR, Section 15.6.4.

7. FSAR, Section 15.6.5.

8. 10 CFR 50, Appendix K.

9. FSAR, Section 6.3.3.

10. 10 CFR 50.46.

11. ~~FSAR, Section 6.3.3~~Not Used.

12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

13. FSAR, Section 6.3.3.3.

14. NEDO 32291-A, "System Analysis for the Elimination of Selected Response Time Testing Requirements, October 1995.

15. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.

REFERENCE

1. FSAR 15.4.9

2. ~~XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.~~ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," (as identified in the COLR).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCE

1. ~~XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.~~ANP-10333P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)," (as identified in the COLR).

Enclosure 7 of PLA-7783

List of Regulatory Commitments

Regulatory Commitments Contained in this Correspondence

The following table identifies actions committed to in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

#	Regulatory Commitment	Due Date
7783-1	Susquehanna will provide the SSES Unit 2 Cycle 21 Fuel Cycle Design Report to the NRC for information.	15 days following approval of the report
7783-2	Susquehanna will provide the SSES Unit 2 Cycle 21 Nuclear Fuel Bundle Design Report to the NRC for information.	15 days following approval of the report
7783-3	Susquehanna will provide the SSES Unit 2 Cycle 21 SLMCPR Report to the NRC for information.	15 days following approval of the report
7783-4	Susquehanna will provide the SSES Unit 2 Cycle 21 Fuel Rod Design Report to the NRC for information.	15 days following approval of the report
7783-5	Susquehanna will provide the SSES Unit 2 Cycle 21 Reload Safety Analysis Report to the NRC for information	15 days following approval of the report
7783-6	Susquehanna will provide the SSES Unit 1 Cycle 23 Reload Safety Analysis Report to the NRC for information.	Upon issuance of the SSES Unit 1 COLR

Enclosure 8b of PLA-7783

Framatome Topical Report
ANP-3753NP

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

(Non-Proprietary Version)

Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel

ANP-3753NP
Revision 0

May 2019

ATRIUM, ULTRAFLOW, FUELGUARD, Z4B, and S-RELAP5 are trademarks or registered trademarks of Framatome (formerly known as AREVA Inc.) or its affiliates, in the USA or other countries.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
3GFG	Third Generation FUELGUARD
ACE	Framatome's advanced critical power correlation [ ]
AFC	Advanced Fuel Channel
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BWR	boiling water reactor
CHF	critical heat flux
CPR	critical power ratio
DIVOM	delta-over-Initial CPR versus oscillation magnitude
EPU	extended power uprate
HPCI	High Pressure Coolant Injection
IHPCIS	Inadvertant High Pressure Coolant Injection System
KATHY	Karlstein thermal hydraulic test facility
LHGR	linear heat generation rate
LOCA	loss of coolant accident
LTP	Lower Tie Plate
MELLLA	Maximum Extended Load Line Limit Analysis
MCPR	minimum critical power ratio
NRC	Nuclear Regulatory Commission, U. S.
OLMCPR	operating limit minimum critical power ratio
PLFR	part length fuel rod
SLMCPR	safety limit minimum critical power ratio
SER	safety evaluation report
TIP	traversing incore probe
UTP	Upper Tie Plate
Z4B	Zircolay BWR material similar to Zircaloy-4
Zry-4	Zircaloy-4

1.0 INTRODUCTION

This document reviews the Framatome approved licensing methodologies to demonstrate that they are applicable to licensing and operation of the Susquehanna Nuclear Plant with ATRIUM 11 in the extended power uprate (EPU) operating domain with a representative power/flow operating map in Figure 1-1. Application of the new methods added for ATRIUM 11 (ACE ATRIUM 11, RODEX-4 for Chromia doped fuel, SLMCPR, AURORA-B AOO, CRDA* and LOCA) for EPU applications are addressed in this document or in plant specific applications of the new methodologies. These methodologies have all been approved for application to mixed core loadings as discussed in Appendix A including the ATRIUM-10 and ATRIUM 11 fuel.

The **[ ]** applied for CRDA startup range evaluation in AURORA-B CRDA and the application of **[ ]** fuel property models for UO2 and Cr-doped UO2 in STAIF and RAMONA5-FA are the only plant specific applications addressed in this report.

This document applies to both Susquehanna units since both Susquehanna BWR/4s are identical. The most significant difference between the units is the core loadings and corresponding core designs. The impact of the differences in core designs between units and cycles is addressed in the cycle specific reload report for each unit.

For the introduction of ATRIUM 11 at EPU conditions a review of the RAI's received from previous license applications was used to identify anything that needed to be addressed. Most of the issues identified in previous license applications have been addressed by the NRC approved methodologies that are being used for the licensing of ATRIUM 11 fuel in Susquehanna.

---

* For the Susquehanna ATRIUM 11 plant-specific application of CRDA, **[ ]** has been applied for the startup range evaluations.

Figure 1-1
Susquehanna Power Flow Operating Map

2.0 OVERVIEW

The introduction of ATRIUM 11 fuel coincides with the application of a new modern suite of methodologies (References 1 through 9 and 20) that also address a number of industry concerns. This is the second application of the entire suite of new and upgraded methodologies. Susquehanna currently operates with ATRIUM-10 fuel and is transitioning to ATRIUM 11. The design characteristics of the ATRIUM-10 and ATRIUM 11 are explicitly accounted for in all of the models for operation with EPU. The differences in fuel design characteristics between the ATRIUM-10 and ATRIUM 11 are discussed in Section 3.0.

The first step in determining the applicability of current licensing methods to Susquehanna operating conditions was a review of Framatome BWR topical reports listed in Table 2-1 and the Susquehanna facility operating license conditions to identify SER restrictions. This review identified penalties on Neutronic methods applied at EPU conditions for OPRM amplitude setpoint and pin power uncertainty/radial power correlation coefficient for SLMCPR analysis. Applicability of methods to EPU conditions and removal of these penalties is addressed in Sections 7.0 and 9.0 of this report. This review identified that there are no SER restrictions on core power level or core flow for the Framatome topical reports up to and including EPU. The review also indicated that the **[**

]. This is discussed in the Thermal Hydraulics section.

Based on the fundamental characteristics of the fuel designs, each of the major analysis domains thermal-mechanics, thermal-hydraulics, mechanics, core neutronics, transient analysis, LOCA and stability are assessed to determine any challenges to application.

Table 2-1 Framatome Licensing Topical Reports

Document Number	Document Title
XN-NF-79-56(P)(A) Revision 1 and Supplement 1	"Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Exxon Nuclear Company, November 1981
XN-NF-85-67(P)(A) Revision 1	"Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, July 1986
XN-NF-85-92(P)(A)	"Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986
ANF-89-98(P)(A) Revision 1 and Supplement 1	"Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995
ANF-90-82(P)(A) Revision 1	"Application of ANF Design Methodology for Fuel Assembly Reconstitution," Advanced Nuclear Fuels Corporation, May 1995
EMF-93-177(P)(A) Revision 1	"Mechanical Design for BWR Fuel Channels," Framatome ANP, August 2005
EMF-93-177P-A Revision 1 Supplement 1P-A Revision 0	"Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," AREVA Inc., September 2013
BAW-10247PA Revision 0	"Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008
BAW-10247PA, Supplement 1P-A, Revision 0	"Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding", April 2017
BAW-10247P-A, Supplement 2P-A, Revision 0	"Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods", Framatome Inc., August 2018
ANP-10340PA Revision 0	"Incorporation of Chromium-Doped Fuel in AREVA Approved Methods", Framatome Inc., May 2018

Table 2-1 Framatome Licensing Topical Reports *(Continued)*

Document Number	Document Title
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2	"Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company, March 1983
XN-NF-80-19(P)(A) Volume 4 Revision 1	"Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986
EMF-2158(P)(A) Revision 0	"Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/ MICROBURN-B2," Siemens Power Corporation, October 1999
EMF-CC-074(P)(A) Volume 1	"STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994
EMF-CC-074(P)(A) Volume 4, Revision 0	"BWR Stability Analysis Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation, August 2000
BAW-10255PA Revision 2	"Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP, May 2008
EMF-3028P-A Volume 2 Revision 4	“RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2: Theory Manual,” AREVA NP, March, 2013
XN-NF-79-59(P)(A)	"Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, November 1983
XN-NF-80-19(P)(A) Volume 3 Revision 2	"Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987
EMF-2209(P)(A) Revision 3	"SPCB Critical Power Correlation," AREVA NP, September 2009.
ANP-10335P-A Revision 0	“ACE/ATRIUM 11 Critical Power Correlation”, Framatome Inc., May 2018
ANP-10307PA Revision 0	"AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011

Table 2-1 Framatome Licensing Topical Reports *(Continued)*

Document Number	Document Title
EMF-2292(P)(A) Revision 0	"ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation, September 2000
EMF-2361(P)(A) Revision 0	"EXEM BWR-2000 ECCS Evaluation Model", Framatome ANP Richland, Inc., May 2001
ANF-1358(P)(A) Revision 3	"The Loss of Feedwater Heating Transient in Boiling Water Reactors," Framatome ANP, September 2005
ANP-10300P-A Revision 1	"AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios" Framatome Inc., January 2018
ANP-10332PA Revision 1	"AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios" Framatome Inc., March 2019
ANP-10333P-A Revision 0	"AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident Scenarios", Framatome Inc., March 2018

3.0 ATRIUM 11 FUEL ASSEMBLY DESIGN

[ ]

The fuel design utilizes a square internal water channel which occupies nine (3x3) lattice positions. The upper and lower ends of the water channel are attached to connecting hardware which provides a load chain between the upper and lower tie plates.

The 11x11 rod array is comprised of 92 full length fuel rods, 8 long part length fuel rods (PLFR) and 12 short PLFRs. The PLFRs are captured in the LTP grid to prevent axial movement.

The fuel rod pitch is slightly larger in the upper section of the assembly relative to the fuel rod pitch in the lower section of the assembly. The array of fuel rods remain orthogonal throughout the assembly.

The nine ULTRAFLOW™ spacers are [ ] and utilize [ ]

[

]

Details of the fuel design characteristics are presented in Table 3-1 and Table 3-2 along with the equivalent values for the ATRIUM-10 fuel design which is currently used and licensed in the Susquehanna units.

Table 3-1 Fuel Assembly and Component Description

Table 3-2 Fuel Channel and Fastener Description

4.0 MECHANICAL LIMITS METHODOLOGY

The LHGR limit is established to support plant operation while satisfying the fuel mechanical design criteria. The methodology for performing the fuel rod evaluation is described in References 3 through 5. The extension of these methods to fuel incorporating chromia is described in Reference 6. Fuel rod design criteria evaluated by the methodology are contained in References 3 and 11.

Fuel rod power histories are generated as part of the methodology for equilibrium cycle conditions as well as cycle-specific operation. These power histories include the impact of channel bow as described in Reference 3. A comprehensive number of uncertainties are taken into account in the categories of operating power uncertainties, code model parameter uncertainties, and fuel manufacturing tolerances. In addition, adjustments are made to the power history inputs for possible differences in planned versus actual operation. Upper limits on the analysis results are obtained for comparison to the design limits for fuel melt, cladding strain, rod internal pressure and other topics as described by the design criteria.

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

5.0 THERMAL HYDRAULICS

5.1 *ATRIUM 11 Void Fraction*

The **[ ]** void-quality correlation has been qualified by Framatome against both the FRIGG void measurements, ATRIUM-10 and ATRIUM 10XM measurements. The standard deviation for the FRIGG tests was shown to be **[ ]** while the standard deviation for the ATRIUM-10 and ATRIUM 10XM tests was found to be **[ ]** respectively. **[**

] the use of the **[ ]** correlation for ATRIUM 11 is justified.

The ATRIUM 11 **[ ]** void fraction measurements. S-RELAP5 was assessed against previous measurements based upon fundamental hydraulic characteristics. The Marviken assembly of FRIGG had a 2-sigma error of **[ ]** in void prediction. The ATRIUM-10 has a 2-sigma error of **[ ]** for void. **[ ]**; therefore, the use of a 2-sigma error of **[ ]** is justified for the ATRIUM 11.

5.2 *ACE/ATRIUM 11 Critical Power Ratio Correlation*

The critical power ratio (CPR) correlation used in MICROBURN-B2, SAFLIM3D, S-RELAP5, RAMONA5-FA, and X-COBRA is based on the ACE/ATRIUM 11 critical power correlation described in Reference 7. As with all Framatome correlations, the range of applicability is enforced in Framatome methods through automated bounds checking and corrective actions. The ATRIUM 11 bounds checking process is similar to the ATRIUM-10 as provided in Table 5-1. The ACE CPR correlation uses K-factor values to account for rod local peaking, rod location and bundle geometry effects.

The K-factor parameter is described in detail in Section 6.10 of Reference 7.

The ranges of applicability of the ACE/ATRIUM 11 and SPCB are compared in Table 5-2.

Table 5-1 SPCB Bounds Checking

Table 5-2 Comparison of the Range of Applicability for the ACE/ATRIUM 11 and SPCB Correlations

5.3 *Loss Coefficients*

Wall friction and component loss coefficients were determined for Susquehanna based on single-phase testing of a prototypic ATRIUM 11 fuel assembly in the Portable Hydraulic Test Facility (PHTF). Prototypical fuel rods, spacer grids, flow channel, upper tie plate and lower tie plate were used in the testing. A description of the PHTF facility and an overview of the process for determining the component loss coefficients are described in Reference 12.

The ATRIUM 11 PHTF tests form the basis for the single phase loss coefficients currently used for design and licensing analyses supporting U.S. BWRs. The PHTF is used by Framatome to obtain single phase loss coefficients for the spacers. The friction factor correlation is a Reynolds dependent function based on the Moody friction model and the measured surface roughness. The pressure drops across the spacers are measured in the PHTF for each new design. **[**

]

The wall friction and component loss coefficients determined from the PHTF and utilized in the validation of the MICROBURN-B2 pressure drop model for the ATRIUM 11 fuel design are provided in Table 5-3.

PHTF data was reduced to determine single phase losses for the spacers in the **[**

] of the bundle. The values have been selected because they are representative of the hydraulic characteristics of actual ATRIUM 11 fuel assemblies loaded into the reactor.

The modeling of the two-phase spacer pressure drop multiplier for the ATRIUM 11 fuel design has been confirmed with two-phase pressure drop measurements taken in the KATHY facility.

Figure 5-1 shows measured versus the MICROBURN-B2 predicted two phase pressure drop for a range of conditions. This figure confirms the applicability of the thermal-hydraulic models to predict pressure drop for the ATRIUM 11 design.

Table 5-3 Hydraulic Characteristics of ATRIUM 11 Fuel Assemblies

[ ]

* Loss coefficients are referenced to the adjacent assembly bare rod flow area.

† [ ]

Figure 5-1 Measured versus Predicted (MICROBURN-B2) Bundle Pressure Drop

5.4 *Safety Limit MCPR*

The safety limit MCPR (SLMCPR) methodology is used to determine the Technical Specification SLMCPR value that ensures that 99.9% of the fuel rods are expected to avoid boiling transition during normal reactor operation and anticipated operation occurrences. The SLMCPR methodology for Susquehanna ATRIUM 11 is described in Reference 9. The SLMCPR is determined by statistically combining calculation uncertainties and plant measurement uncertainties that are associated with the calculation of MCPR. The thermal hydraulic, neutronic, and critical power correlation methodologies are used in the calculation of MCPR. The applicability of these methodologies for Susquehanna is discussed in other sections of this report.

Framatome calculates the SLMCPR on a cycle-specific basis to protect all allowed reactor operating conditions. The analysis incorporates the cycle-specific fuel and core designs. The initial MCPR distribution of the core is a major factor affecting how many rods are predicted to be in boiling transition. The MCPR distribution of the core depends on the neutronic design of the reload fuel and the fuel assembly power distributions in the core. Framatome SLMCPR methodology specifies that analyses be performed with a design basis power distribution that "… conservatively represents expected reactor operating states which could both exist at the MCPR operating limit and produce a MCPR equal to the MCPR safety limit during an anticipated operational occurrence." (Reference 9, Section 3.3.2).

[

]

[

]. This is a plant specific extension to the Reference 9 approved methodology.

6.0 TRANSIENTS AND ACCIDENTS

6.1 *Void Quality Correlation Uncertainties*

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

The OLMCPR is determined based on the safety limit MCPR (SLMCPR) methodology and the transient analysis (ΔCPR) methodology. Void-quality correlation uncertainty is not a direct input to either of these methodologies; however, the impact of void-correlation uncertainty is inherently incorporated in both methodologies as discussed below.

The SLMCPR methodology explicitly considers important uncertainties in the Monte Carlo calculation performed to determine the number of rods in boiling transition. One of the uncertainties considered in the SLMCPR methodology is the bundle power uncertainty. This uncertainty is determined through comparison of calculated to measured core power distributions. Any miscalculation of void conditions will increase the error between the calculated and measured power distributions and be reflected in the bundle power uncertainty. Therefore, void-quality correlation uncertainty is an inherent component of the bundle power uncertainty used in the SLMCPR methodology.

The transient analyses methodology is a combination of deterministic, bounding analyses and a statistical evaluation of the impact of model uncertainties that contains conservatism in addition to uncertainties in individual phenomena. Conservatism is incorporated in the methodology in two ways: (1) computer code models are developed to produce conservative results on an integral basis relative to benchmark tests, and (2) important input parameters are biased in a conservative direction in licensing calculations.

The transient analyses methodology results in predicted power increases that are bounding relative to benchmark tests. In addition, for licensing calculations a multiplier is applied to the calculated integral power to provide additional conservatism to account for uncertainties in individual phenomena as defined in the transient analyses methodology. Therefore, uncertainty in the void-quality correlation is inherently incorporated in the transient analysis methodology.

In addition to the impact of void-quality correlation uncertainty being inherently incorporated in the analytical methods used to determine the OLMCPR, biasing of important input parameters in licensing calculations provides additional conservatism in establishing the OLMCPR. No additional adjustments to the OLMCPR are required to address void-quality correlation uncertainty.

6.2 *Assessment of the Void-Quality Correlation*

As discussed in Section 5.1, the **[ ]** is equally applicable to the ATRIUM 11 applications at Susquehanna.

6.3 [ ]

[ ]

Section 3.5.2.7 documented the NRC's review of this response as such:

> *However, the NRC staff does not agree with AREVA's third response.* **[ ]**

[

]

The result of this conclusion was Limitation and Condition 12 of AURORA-B AOO which requires plant-specific approval for any changes made to the transient coolant mixing. This section is intended to provide the description of the method used to determine [

].

6.3.1 Transient Mixing Determination

For Susquehanna, the mixing is evaluated using [

]

[ ]

[

]

Figure 6.1 [

]

6.3.2 Implementation in AURORA-B AOO Licensing

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, **[**

]

[ ]

6.4 *Control Rod Drop Accident*

[ ]

[

]

Figure 6-2 Total Enthalpy Rise with CHF Multipliers

6.5 *Loss of Coolant Accident*

The approved AURORA-B LOCA methodology, Reference 20, has been approved to be applicable to BWR/3 to BWR/6 with conditions extending up to EPU with extended flow windows. This bounds the EPU/MELLLA flow domain that is currently implemented at Susquehanna. In addition, Limitation and Condition 27 of Reference 20 addresses the application of the methodology to **[**

].

7.0 STABILITY

Stability analysis are performed using the Option III methodology described in Reference 21. This methodology was approved prior to the implementation of chromia doped fuel. The RAMONA5-FA (Reference 21) and STAIF (Reference 23) methods used in the Option III methodology have been updated to address this advanced fuel design feature using **[ ]**. The fuel property models implemented are the same models used in the Framatome generic ATWS-I methodology described in Reference 22. Susquehanna Units 1 and 2 are only implementing the fuel rod property models from Reference 22. Both Susquehanna units continue to implement stability Option III for the NRC approved EPU operating domain (Figure 1-1) which remains unchanged.

Justification of the implementation of these models is provided in the following section.

7.1 [ ] *Fuel Rod Models*

For the Susquehanna application of the Option III methodology **[**

]. For Chromia-doped pellets, modifications to the standard UO_2 thermal conductivity and **[ ]** models were necessary to account for the effects of the Chromia doping. The Chromia-doped pellet specific models presented here are **[**

].

The subsections that follow present the fuel rod material properties and the pellet-clad gap heat transfer coefficient model used in the Susquehanna application of the Option III methodology.

7.1.1 Material Properties

[ ]

7.1.2 Pellet-Clad Gap Heat Transfer Coefficient

[ ]

[ ]

7.2 *Radial Power Deposition Distributions in Fuel Pellets*

[ ]

7.3 *STAIF Reactor Benchmarks Using New Fuel Rod Property Models*

A description of the STAIF reactor benchmarking suite is given in Section 4.0 of Reference 23. All reactor benchmarks in this suite were reanalyzed with the new fuel rod property models described in Section 7.1.

Table 7-1

[ ]

[ ]

Table 7-2

[ ]

* Regional Oscillation Mode

Table 7-3
[ ]

[ ]

Table 7-4
[ ]

[ ]

Table 7-5

[ ]

7.4 *RAMONA5-FA Reactor Benchmarks Using New Fuel Rod Property Models*

A description of the RAMONA5-FA reactor benchmarking suite is given in Section 5.0 of Reference 23. All reactor benchmarks in this suite were reanalyzed with the new fuel rod property models described in Section 7.1.

A description of the benchmark analyses is given in the following sections along with the RAMONA5-FA calculated growth ratios and frequencies.

7.4.1 [ ]

[ ]

* [ ]

7.4.2 [ ]

[

]

7.4.3 [ ]

[

]

7.4.4 [ ]

[

]

* [

]

7.5 *Removal of OPRM Amplitude Setpoint Penalty*

The current Susquehanna Operating License includes licensing condition 2.C.(38)(a) and 2.C.(22)(a) for Units 1 and 2 , respectively, on the OPRM setpoint determination. This condition states:

(38) Neutronic Methods

(a) An OPRM amplitude setpoint penalty will be applied to account for a reduction in thermal neutrons around the LPRM detectors caused by transients that increase voiding. This penalty will reduce the OPRM scram setpoint according to the methodology described in Response No. 3 of the operating licensee's letter, PLA-6306, dated November 30, 2007. This penalty will be applied until NRC evaluation determines that a penalty to account for this phenomenon is not warranted.

On December 3, 2007, the ACRS performed a review of the RAMONA5-FA DIVOM methodology, Reference 21. This review led to an additional RAI being issued relating to bypass boiling. The response looked at the effect of reduced LPRM sensitivity in the upper levels on the OPRM system response. The work concluded that bypass voiding **[**

]. In addition, the NRC also conducted a full review of the RAMONA5-FA code system, Reference 34. RAI-21 of Reference 34 was issued to evaluate the transient impact of bypass boiling oscillations during power oscillations. This work confirmed that bypass voiding **[**

]. These conclusions are also summarized in Section 2.3.8 of the SE for Reference 34. Based on the NRC reviews of both the DIVOM methodology, Reference 21, and the RAMONA5-FA code system, Reference 34, no additional penalties on the OPRM setpoint are required and this license condition can be safely removed.

8.0 ATWS

8.1 *ATWS General*

The AURORA-B methodology is used for the ATWS overpressurization analysis. The ACE/ATRIUM 11 critical power correlation pressure limit is not a factor in the analysis.

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

The ATWS event is not limiting relative to acceptance criteria identified in 10 CFR 50.46. The core remains covered and adequately cooled during the event. Following the initial power increase during the pressurization phase, the core returns to natural circulation conditions after the recirculation pumps trip and fuel cladding temperatures are maintained at acceptable low levels. The ATWS event is significantly less limiting than the loss of coolant accident relative to 10 CFR 50.46 acceptance criteria.

8.2 *Void Quality Correlation Bias*

Framatome performs cycle-specific ATWS analyses of the short-term reactor vessel peak pressure using the AURORA-B methodology. The ATWS peak pressure calculation is a core-wide pressurization event that is sensitive to similar phenomenon as other pressurization transients. Bundle design is included in the development of input for the coupled neutronic and thermal-hydraulic S-RELAP5 core model. Important inputs to the S-RELAP5 system model are biased in a conservative direction.

The Framatome transient analysis methodology is a deterministic, bounding approach that contains sufficient conservatism and evaluates uncertainties in individual phenomena. As demonstrated in Section 5.1 the void-quality correlation is robust for past and present designs including the ATRIUM 11.

The reference ATWS analysis evaluation presented in the topical report (Reference 1) of the core active density response, which is closely related to the void quality correlation, showed minimal changes in the peak vessel pressure. A study was also performed for the ASME overpressure event (FWCF) with similar results.

8.3 *ATWS Containment Heatup*

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

[ ]

[

]

Table 8-1 [

]

[

]

* Boron worth is quoted as a positive value since it refers to the boron defect. The ppm boron used is 660 at 68 F. The calculation uses the equivalent boron at 349.6 F, used in SSES SLCS calculations.

9.0 NEUTRONICS

From the neutronics perspective, the ATRIUM 11 fuel design differs from the ATRIUM-10 fuel design primarily in the fuel rod diameter and pitch and position and number of the part length rods. The CASMO-4 code is designed to model a wide range of fuel rod diameters and pitches. The neutronic models have already been demonstrated to accurately model the vacant positions and this continues to be true for the ATRIUM 11 fuel design.

9.1 *Shutdown Margin*

The part length rod in the corner of the assembly improves the shutdown margin performance of the fuel design because of the flux trap that is created in the cold condition with the vacant rod position of all four assemblies in a control cell being in close proximity. The heterogeneous solution of CASMO-4 accurately models the vacant rod position and the associated reactivity. No change in predicted hot operating or cold critical eigenvalue is anticipated with the ATRIUM 11 fuel design.

9.2 *Monitoring*

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

9.3 *Removal of Pin Power Uncertainty and Bundle Power Correlation Coefficient Penalty*

No significant change in the uncertainty of the predicted detector response relative to the measurements is anticipated. The SLMCPR pin power distribution uncertainty and bundle power correlation coefficient restriction/penalty present in the current Susquehanna facility operating license (licensing condition 2.C.(38)(b) and 2.C.(22)(b) for Units 1 and 2 respectively) for EPU operation should be removed. Since the analysis and core monitoring at Susquehanna is based upon the CASMO-4/MICROBURN-B2 methodology there is no need for any restrictions/uncertainty penalties when using AURORA- B methods per section 3.3.2.4.5 of the AURORA-B safety evaluation. As noted in section 5.1 of this report, use of the Dix-Findlay correlation for ATRIUM 11 fuel is justified. In addition, since Susquehanna is currently operating within approved EPU conditions and not requesting operation with extended flow windows, operating conditions are within previously validated Power/Flow ratios.

9.4 *Bypass modeling*

The bypass behavior of the ATRIUM 11 fuel design is identical to the ATRIUM-10 fuel design, thus there is no difference in the modeling. Any differences in bypass heat deposition are treated explicitly.

10.0 REFERENCES

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

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

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

4. BAW-10247PA Revision 0 Supplement 1P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding," AREVA Inc., April 2017.

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

6. ANP-10340PA Revision 0, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," Framatome Inc., May 2018.

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

8. EMF-93-177P-A Revision 1 Supplement 1P-A Revision 0 "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," AREVA Inc., September 2013.

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

10. ANP-3762P Revision 0, "Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies," Framatome Inc., May 2019.

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

12. I.K. Madni, et al., "Development of Correlations for Pressure Loss/Drop Coefficients Obtained From Flow Testing of Fuel Assemblies In Framatome ANP’S PHTF," Paper Number 22428, Proceedings of ICONE10, Arlington, VA, April 14-18,2002.

13. Simcenter STARCCM+ Version 13.06, User Guide, 2018.

14. [
]

15. D. C. Groeneveld, L. K. H. Leung, P. L. Kirillov, V. P. Bobkov, I. P. Smogalev, V. N. Vinogradov, X. C. Huang, E. Royer, “The 1995 look-up tables for critical heat flux in tubes, “Nuclear Engineering and Design, 163, pp.1-23, 1996.

16. D. C. Groeneveld, L. K. H. Leung, Y. Guo, A. Vasic, M. El Nakla, S. W. Peng, J. Yangand S. C. Cheng, “Lookup Tables for Predicting CHF and Film Boiling Heat Transfer: Past, Present, and Future,” Nucleat Technology, 152, pp.87-104, 2005.

17. D. C. Groeneveld, J. Q. Shan, A. Z. Vasic, L. K. H. Leung, A. Durmayz, J. Yang, S. C. Cheng and A. Tanase, “The 2006 CHF look-up table,” Nuclear Engineering and Design, 237, pp. 1909-1922, 2007.

18. M. Lee, “A critical heat flux approach for square rod bundles using the 1995 Groeneveld CHF table and bundle data of heat transfer research facility,” Nuclear Engineering and Design, 197, pp. 357-374, 2000.

19. N. I. Kolev, Check of the 2005 look up table for prediction of CHF in bundles, Nuclear Engineering and Design 237(9):978-981 May 2007.

20. ANP-10332PA, AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios, March 2014.

21. BAW-10255PA Revision 2, “Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code”, May 2008.

22. ANP-10346P, “ATWS-I Analysis Methodology for BWRs Using RAMONA5-FA”, December 2017

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

24. EMF-CC-074(P)(A) Volume 1, "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain," and Volume 2 "STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain - Code Qualification Report," Siemens Power Corporation, July 1994.

25. N. Zuber and J. A. Findlay, “Average Volumetric Concentration in Two-Phase Flow Systems,” J. Heat Transfer, 1965.

26. P. Coddington and R. Macian, “A Study of the Performance of Void Fraction Correlations Used In the Context of Drift-Flux Two-Phase Flow Models,” Nuclear Engineering and Design, 215, 199-216, June 2002.

27. **[ ]**

28. K. Ohkawa and R. T. Lahey, Jr., “The Analysis of CCFL Using Drift-Flux Models,” Nuclear Engineering and Design, 61, 1980.

29. S. Misu et al., “The Comprehensive Methodology for Challenging BWR Fuel Assembly and Core Design used at FANP,” proceedings on CD-ROM, PHYSOR 2002, Seoul, Korea, October 7-10, 2002

30. XN-NF-84-105(P)(A) Volume 1 Supplement 4, XCOBRA-T: A Computer Code For BWR Transient Thermal-Hydraulic Core Analysis, Void Fraction Model Comparison to Experimental Data, Advanced Nuclear Fuels Corporation, June 1988.

31. O. Nylund et al., “Hydrodynamic and Heat Transfer Measurements on A Full-Scale Simulated 36-Rod Marviken Fuel Element with Non-Uniform Radial Heat Flux Distribution,” FRIGG-3, R-494/RL-1154, November 1969.

32. J. Skaug et al., “FT-36b, Results of void Measurements,” FRIGG-PM-15, May 1968.

33. O. Nylund et al., “Hydrodynamic and Heat Transfer Measurements on A Full-Scale Simulated 36-Rod Marviken Fuel Element with Uniform Heat Flux Distribution,” FRIGG-2, R-447/RTL-1007, May 1968.

34. EMF-3028P-A Volume 2 Revision 4, “RAMONA5-FA: A Computer Program for BWR Transient Analysis in the Time Domain Volume 2: Theory Manual,” March 2013.

Appendix A. Application of Framatome Methodology for Mixed Cores

A.1 DISCUSSION

Framatome has considerable experience analyzing fuel design transition cycles and has methodology and procedures to analyze mixed cores composed of multiple fuel types. For each core design, analyses are performed to confirm that all design and licensing criteria are satisfied. The analyses performed explicitly include each fuel type in the core. The analyses consider the cycle-specific core loading and use input data appropriate for each fuel type in the core. The mixed core analyses are performed using generically approved methodology in a manner consistent with NRC approval of the methodology. Based on results from the analyses, operating limits are established for each fuel type present in the core. During operation, each fuel type is monitored against the appropriate operating limits.

Thermal hydraulic characteristics are determined for each fuel type that will be present in the core. The thermal hydraulic characteristics used in core design, safety analysis, and core monitoring are developed on a consistent basis for both Framatome fuel and other vendor co-resident fuel to minimize variability due to methods. For Susquehanna operation, the entire core will be composed of Framatome fuel designs.

For core design and nuclear safety analyses, the neutronic cross-section data is developed for each fuel type in the core using CASMO-4. MICROBURN-B2 is used to design the core and provide input to safety analyses (core neutronic characteristics, power distributions, etc.). Each fuel assembly is explicitly modeled in MICROBURN-B2 using cross-section data from CASMO-4 and geometric data appropriate for the fuel design.

Fuel assembly thermal mechanical limits for all fuel are verified and monitored for each mixed core designed by Framatome. Framatome performs design and licensing analyses to demonstrate that the core design meets steady-state limits and that transient limits are not exceeded during anticipated operational occurrences.

The critical power ratio (CPR) is evaluated for each fuel type in the core using calculated local fluid conditions and an appropriate critical power correlation. Fuel type specific correlation coefficients for Framatome fuel are based on data from the Framatome critical power test facility. The SPCB critical power correlation will be used for monitoring ATRIUM-10 fuel present during the transition to operation with ATRIUM 11 at Susquehanna. The critical power ratio (CPR) correlation used for the ATRIUM 11 fuel is the ACE/ATRIUM 11 critical power correlation described in Reference 7. The ACE CPR correlation uses K-factor values to account for rod local peaking, rod location and bundle geometry effects.

In the safety limit MCPR analysis each fuel type present in the core is explicitly modeled using appropriate geometric data, thermal hydraulic characteristics, and power distribution information (from CASMO-4 and MICROBURN-B2 analyses). CPR is evaluated for each assembly using fuel type specific correlation coefficients. Plant and fuel type specific uncertainties are considered in the statistical analysis performed to determine the safety limit MCPR. The safety limit MCPR analysis is performed each cycle and uses the cycle specific core configuration.

An operating limit MCPR is established for each fuel type in the core. For fast transients the AURORA-B code (Reference 1) is used to determine the overall system and hot channel response. The core nuclear characteristics used in AURORA-B are obtained from MICROBURN-B2 and reflect the actual core loading pattern. Critical power performance is evaluated using local fluid conditions and fuel type specific CPR correlation coefficients. The transient CPR response is used to establish an operating limit MCPR for each fuel type.

For transient events that are sufficiently slow such that the heat transfer remains in phase with changes in neutron flux during the transient, evaluations are performed with steady state codes such as MICROBURN-B2 in accordance with NRC approval. Such slow transients are modeled by performing a series of steady state solutions with appropriate boundary conditions using the cycle specific design core loading plan.

Each fuel assembly type in the core is explicitly modeled. The change in CPR between the initial and final condition after the transient is determined, and if the CPR change is more severe than those determined from fast transient analyses, the slow transient result is used to determine the MCPR operating limit.

Stability analyses to establish OPRM setpoints and backup stability exclusion regions are performed using the cycle-specific core loading pattern. The stability analyses performed with RAMONA5-FA and STAIF explicitly model each fuel type in the core. Each fuel type is modeled using appropriate geometric, thermal hydraulic and nuclear characteristics determined as described above. The stability OPRM setpoints and exclusion region boundaries are established based on the predicted performance of the actual core composition.

MAPLHGR operating limits are established and monitored for each fuel type in the core to ensure that 10 CFR 50.46 acceptance criteria are met during a postulated LOCA. The S-RELAP5 code is used to determine the overall system and hot channel response during a postulated LOCA. While system analyses are typically performed on an equilibrium core basis, the thermal hydraulic characteristics of all fuel assemblies in the core are considered to ensure the LOCA analysis results are applicable to mixed core configurations.

The core monitoring system will monitor each fuel assembly in the core. Each assembly is modeled with geometric, thermal hydraulic, neutronic, and CPR correlation input data appropriate for the specific fuel type. Each assembly in the core will be monitored relative to thermal limits that have been explicitly developed for each fuel type.

In summary, Framatome methodology is used consistent with NRC approval to perform design and licensing analyses for mixed cores. The cycle design and licensing analyses explicitly consider each fuel type in mixed core configurations. Limits are established for each fuel type and operation within these limits is verified by the monitoring system during operation.

Enclosure 8c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3753P, Applicability of Framatome
BWR Methods to Susquehanna with ATRIUM 11 Fuel Report

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3753P Revision 0, "Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel," dated May 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 31st

day of May, 2019.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY
PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

Enclosure 9b of PLA-7783

Framatome Topical Report
ANP-3762NP

Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies

(Non-Proprietary Version)

framatome

Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies

ANP-3762NP
Revision 0

Licensing Report

May 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
AFC	Advanced fuel channel
AOO	Anticipated operational occurrences
ASME	American Society of Mechanical Engineers
BOL	Beginning of life
B&PV	Boiler and pressure vessel
BQ	Beta-quench
BWR	Boiling water reactor
EOL	End of life
LOCA	Loss-of-coolant accident
LTP	Lower tie plate
MWd/kgU	Megawatt-days per kilogram of Uranium
NRC	U. S. Nuclear Regulatory Commission
OLC	Optimized load chain
PLFR	Part-length fuel rods
SHP1	Servo hydraulic test facility
SRP	Standard review plan
UTP	Upper tie plate
Z4B	Proprietary Zircaloy BWR material similar to Zircaloy-4
Zry-4	Zircaloy-4
3GFG	3rd Generation FUELGUARD

1.0 INTRODUCTION

This report documents the successful completion of all licensing analyses and related testing necessary to verify that the mechanical design criteria are met for the ATRIUM 11 Fuel Assemblies supplied by Framatome Inc. (Framatome) for insertion into Susquehanna Units. This report also provides a description of the mechanical design and licensing methods. The scope of this report is limited to an evaluation of the mechanical design of the fuel assembly and fuel channel.

The ATRIUM 11 design is a Framatome advanced boiling water reactor (BWR) fuel design that builds on the history of proven ATRIUM family of fuel designs. The design uses an 11x11 fuel array, a [ ] fuel rod, a central water channel that displaces a 3x3 array of rods and is made from an advanced Zirconium alloy Z4B material, a modular lower tie plate with a 3rd generation FUELGUARD and nine ULTRAFLOW spacer grids [ ] .

The fuel assembly design was evaluated according to the Framatome BWR generic mechanical design criteria (Reference 1). The fuel channel design was evaluated to the criteria given in the fuel channel topical reports (References 2 and 3). The generic design criteria have been approved by the U.S. Nuclear Regulatory Commission (NRC) and the criteria are applicable to the subject fuel assembly and channel design. Mechanical analyses have been performed using NRC-approved design analysis methodology (References 1, 2, 3 and 4). The methodology permits maximum licensed assembly and fuel channel exposures of [ ] (Reference 4, Section 1.0).

The fuel assembly and fuel channel meet all mechanical compatibility requirements for use in Susquehanna Units. This includes compatibility with both co-resident fuel and the reactor core internals.

2.0 DESIGN DESCRIPTION

This section provides a design description of the ATRIUM 11 fuel assembly and fuel channel. Reload-specific design information is available in the design package provided by Framatome for each reload delivery.

2.1 *Overview*

Susquehanna has successfully operated for several cycles with reload quantities of ATRIUM-10 fuel assemblies. Susquehanna will operate with ATRIUM 11 fuel assemblies in reload quantities starting with Susquehanna Unit 2 Cycle 21. The ATRIUM 11 bundle consists of an 11x11 fuel lattice with a square internal water channel that displaces a 3x3 array of rods.

The ATRIUM 11 incorporates key design features relative to previous ATRIUM designs as described in Reference 5.

Table 2-1 lists the key design parameters of the ATRIUM 11 fuel assembly.

2.1.1 Fuel Assembly

Figure 2-1 provides an illustration of the fuel assembly, and Table 2-1 lists the main fuel assembly attributes. The fuel assembly is accompanied by a fuel channel, as described later in this section.

[ ]

2.1.2 Upper Tie Plate and Connecting Hardware

Figure 2-2 provides an illustration of the UTP and connecting hardware.

2.1.3 Water Channel

Figure 2-2 provides an illustration of the water channel, and Table 2-1 lists the main water channel attributes.

[ ]

2.1.4 Spacer Grid

Figure 2-3 provides illustration of the spacer grid, and Table 2-1 lists the main spacer grid attributes.

2.1.5 Lower Tie Plate

Figure 2-4 provides an illustration of the 3GFG FUELGUARD.

2.1.6 Fuel Rods

This mechanical design report documents the fuel structural analyses. The fuel rod thermal-mechanical report provides fuel rod design description detail. Figure 2-5 provides an illustration of the full-length and the two part-length fuel rods.

2.2 *Fuel Channel and Components*

Figure 2-6 provides an illustration of the fuel channel and components, and Table 2-2 lists the fuel channel component attributes.

Table 2-1
Fuel Assembly and Component Description

[ ]

Table 2-1
Fuel Assembly and Component Description
(Continued)

Table 2-2
Fuel Channel and Channel Spacer Assembly Description

[ ]

3.0 FUEL DESIGN EVALUATION

This section provides a summary of the mechanical methodology and results from the structural design evaluations. Results from the mechanical design evaluation demonstrate that the design satisfies the mechanical criteria to the analyzed exposure limit. Sections 3.1 through 3.4 correspond to the fuel assembly criteria sections within Section 3.0 of Reference 1. Section 3.5 and Table 3-2 corresponds to the advanced fuel channel criteria sections within Table 1.1 and 1.2 of Reference 2.

3.1 *Objectives*

The objectives of designing fuel assemblies (systems) to specific criteria are to provide assurance that:

- The fuel assembly (system) shall not fail as a result of normal operation and anticipated operational occurrences (AOOs). The fuel assembly (system) dimensions shall be designed to remain within operational tolerances, and the functional capabilities of the fuel shall be established to either meet or exceed those assumed in the safety analysis.
- Fuel assembly (system) damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.
- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from handling and shipping.

The first four objectives are those cited in the Standard Review Plan (SRP). The latter two objectives are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel. To satisfy these objectives, the criteria are applied to the fuel rod and the fuel assembly (system) designs. Specific component criteria are also necessary to assure compliance. The criteria established to meet these objectives include those given in Chapter 4.2 of the SRP.

3.2 *Fuel Rod Evaluation*

The mechanical design report documents the fuel structural analyses only. The fuel rod evaluation will be documented in the fuel rod thermal-mechanical report. However, the fuel rod mechanical fracturing (Reference 1, Section 3.2.7) is evaluated in Section 3.4.4 *Structural Deformations.*

3.3 *Fuel System Evaluation*

The detailed fuel system design evaluation is performed to ensure the structural integrity of the design under normal operation, AOO, faulted conditions, handling operations, and shipping. The analysis methods are based on fundamental mechanical engineering techniques, often employing finite element analysis, prototype testing, and correlations based on in-reactor performance data. Summaries of the major assessment topics and associated testing are described in the sections that follow.

Prototype testing is an essential element of Framatome methodology for demonstrating compliance with structural design requirements. Results from design verification testing may directly demonstrate compliance with criteria or may be used as input to design analyses.

Testing performed to qualify the mechanical design or evaluate assembly characteristics includes:

- Fuel assembly axial load structural strength
- Fuel assembly fretting
- Fuel assembly static lateral deflection
- Fuel assembly lateral vibration
- Fuel assembly impact
- Spacer grid lateral impact strength
- Tie plate lateral load strength

3.3.1 Stress, Strain, or Loading Limits on Assembly Components

The structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various handling, AOOs, and accident or faulted loads. Framatome uses Section III of the American Society of Mechanical Engineers (ASME) boiler and pressure vessel (B&PV) code as a guide to establish acceptable stress, deformation, and load limits for standard assembly components. These limits are applied to the design and evaluation of the UTP, LTP, spacer grids, springs, and load chain components, as applicable.

All significant loads experienced during normal operation, AOOs, and under faulted conditions are evaluated to confirm the structural integrity of the fuel assembly components. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. See Section 3.3.9 for a discussion of fuel handling loads and Section 3.4.4 for the structural evaluation of faulted conditions. Although normal operation and AOO loads are often not limiting for structural components, a stress evaluation may be performed to confirm the design margin and to establish a baseline for adding accident loads. The stress calculations use conventional, open-literature equations. A general-purpose, finite element stress analysis code, such as ANSYS, may be used to calculate component stresses.

3.3.2 Fatigue

Section addressed in the fuel rod thermal-mechanical report.

3.3.3 Fretting Wear

Fuel rod failures due to grid-to-rod fretting shall not occur. [

] .

Fretting wear is evaluated by testing, as described in Section 3.3.3.1. The testing is conducted by [

] . The inspection measurements for wear are documented. The lack of significant wear demonstrates adequate rod restraint geometry at the contact locations. Also, the lack of significant wear at the spacer cell locations [ ] provides further assurance that no significant fretting will occur at higher exposure levels.

3.3.3.1 Fuel Assembly Fretting Test

A fretting test was conducted on a full-size test assembly to evaluate the ATRIUM 11 fuel rod support design. [

] . After the test, the assembly was inspected for signs of fretting wear. No significant wear was found on fuel rods in contact with spacer springs [

] . The results agree with past test results on BWR designs where no noticeable wear was found on the fuel rods or other interfacing components following exposure to coolant flow conditions.

3.3.4 Oxidation, Hydriding, and Crud Buildup

Section addressed in the fuel rod thermal-mechanical report.

3.3.5 Rod Bow

The predicted rod-to-rod gap closure due to bow is assessed by thermal hydraulics group for impact on thermal margins.

Differential expansion between the fuel rods and cage structure, and lateral thermal and flux gradients can lead to lateral creep bow of the rods in the spans between spacer grids. This lateral creep bow alters the pitch between the rods and may affect the peaking and local heat transfer. The Framatome design criterion for fuel rod bowing is [

].

Visual exams on ATRIUM 11 have not revealed any unusual fuel rod bow behavior for exposures up to [ ] based on the latest experience from Lead Test Assembly post-irradiation exams. This exposure is beyond the threshold where increasing rod bow had been observed on other designs. Therefore, the ATRIUM 11 has been shown to have minimal rod bow. A rod gap closure ratio curve is provided in Reference 4.

3.3.6 Axial Irradiation Growth

Reference 4 requires [

] .

The fuel rod growth correlation is established from [

] .

Assembly growth is established from ATRIUM 10x10 and 11x11 arrayed fuel assemblies with water channels made of Z4B material. It is based on the ATRIUM fuel assembly growth data only and excludes designs with load bearing tie rods as well as the European bundle-in-basket designs. [

] .

The fuel rod and assembly growth approved correlations are described within Reference 4 along with the respective tolerance limits.

3.3.7 Rod Internal Pressure

Section addressed in the fuel rod thermal-mechanical report.

3.3.8 Assembly Lift-off

Fuel assembly lift-off is evaluated under both normal operating conditions (including AOOs) and under faulted conditions. The fuel shall not levitate under normal operating or AOO conditions. Under postulated accident conditions, the fuel shall not become disengaged from the fuel support. These criteria assure control blade insertion is not impaired.

For normal operating conditions, the net axial force acting on the fuel assembly is calculated by adding the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy. The calculated net force is confirmed to be in the downward direction, indicating no assembly lift-off. [

] .

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

For faulted conditions, [

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

3.3.9 Fuel Assembly Handling

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

The fuel assembly structural components are assessed for axial fuel handling loads by analyses and testing. To demonstrate compliance with the criteria, the tests and analyses are performed by loading a test assembly or the individual components of the load chain to an axial tensile force greater than [ ]. An acceptable test and analysis demonstrates no yielding after loading.

Handling requirements for the fuel rod plenum spring are addressed in the fuel rod thermal-mechanical report.

3.3.9.1 Fuel Assembly Axial Load Tests

Each test is used in support of analytical or Finite Element Analysis to demonstrate that no significant permanent deformation occurs for loads [ ] .

Descriptions of tests:

[ ]

3.3.10 Miscellaneous Components

3.3.10.1 Compression Spring Forces

The compression spring force shall support the weight of the upper tie plate and channel throughout the design life of the fuel. The ATRIUM 11 has a single large compression spring mounted on the central water channel. The compression spring serves the same function as previous ATRIUM family of fuel designs by providing support for the UTP and fuel channel. The spring force is calculated based on the installed deflection and specified spring force requirements to meet support criteria. Irradiation-induced relaxation is taken into account for EOL conditions. The minimum compression spring force at EOL is greater than the combined weight of the UTP assembly and fuel channel assembly. Since the compression spring design of that ATRIUM family of fuel assemblies load chain designs do not interact with the fuel rods, no consideration is required for fuel rod buckling loads.

3.3.10.2 LTP Seal Spring

The LTP seal spring shall limit the bypass coolant leakage rate between the LTP and fuel channel. The seal spring shall accommodate expected channel deformation while remaining in contact with the fuel channel. Also, the seal spring shall have adequate corrosion resistance and be able to withstand the operating stresses without yielding.

Flow testing is used to confirm acceptable bypass flow characteristics. The seal spring is designed with adequate deflection to accommodate the maximum expected channel bulge while maintaining acceptable bypass flow. [ ] is selected as the material because of its high strength at elevated temperature and its excellent corrosion resistance. Seal spring stresses are analyzed using a finite element method.

3.4 *Fuel Coolability*

For accidents in which severe fuel damage might occur, core coolability and the capability to insert control blades are essential. Chapter 4.2 of the SRP provides several specific areas important to fuel coolability, as discussed below.

3.4.1 Cladding Embrittlement

Section addressed in the thermal hydraulic reload safety analysis report.

3.4.2 Violent Expulsion of Fuel

Section addressed in the thermal hydraulic reload safety analysis report.

3.4.3 Fuel Ballooning

Section addressed in the thermal hydraulic reload safety analysis report.

3.4.4 Structural Deformations

ATRIUM 11 structural component deformations or stresses from postulated accidents are limited according to requirements contained in the ASME B&PV Code, Section III, Division 1, Appendix F, and SRP Section 4.2, Appendix A.

The methodology for analyzing the fuel under the influence of accident loads is described in the Mechanical Designs for BWR Fuel Channels Topical Report (Reference 2) and is further discussed in Section 3.5.2. Evaluations performed for the fuel under accident conditions include [ ].

[ ].

Dynamic properties of the ATRIUM 11 fuel assembly are provided to Susquehanna in support of evaluations assessing the impact of the introduction of the ATRIUM 11 to the reactor pressure vessel, internal reactor components and other applicable evaluations.

3.4.4.1 Test Verifications

Fuel assemblies are tested with, and without, a fuel channel as described in Appendix C of Reference 2. Testing is preformed to obtain the dynamic characteristics of the fuel assembly and spacer grids. The stiffness, natural frequencies and damping values derived from the tests are used as inputs for analytical models of the fuel assembly and fuel channel. In general, the testing and analyses have shown the dynamic response of ATRIUM 11 to be similar to ATRIUM-10 fuel assemblies.

3.4.4.1.1 Fuel Assembly Static Lateral Deflection Test

A lateral deflection test is performed to determine the fuel assembly stiffness, both with and without a fuel channel. The stiffness is obtained by supporting the fuel assembly at the two ends in a vertical position, applying a side displacement at the central spacer location, and measuring the corresponding force.

3.4.4.1.2 Fuel Assembly Lateral Vibration Tests

The lateral vibration testing consists of both a free vibration test and a forced vibration test
[ ] .

The test setup for the free vibration test [

] .

The forced vibration test [

] .

3.4.4.1.3 Fuel Assembly Impact Tests

Impact testing was performed in a similar manner to the lateral deflection tests. The unchanneled assembly is supported in a vertical position with both ends fixed. The assembly is displaced a specified amount and then released. [

] .

3.4.4.1.4 Spacer Grid Lateral Impact Strength Test

Spacer grid impact strength is determined by a [

] .

The maximum force prior to the onset of buckling was determined from the tests. The results were adjusted to reactor operating temperature conditions to establish an allowable lateral load.

3.4.4.1.5 Tie Plate Strength Tests

In addition to the axial tensile tests described in Section 3.3.9.1, a lateral load test is performed on the UTP and LTP.

The UTP lateral load test was conducted on a test machine which applied [

] . This provides a limiting lateral load for accident conditions.

To determine a limiting lateral load for accident conditions for the 3GFG LTP, a lateral load test was conducted by attaching the grid of the tie plate to a rigid vertical plate [

] .

The results were adjusted to reactor operating temperature conditions to establish an allowable lateral load per Reference 1, Section 3.3.1.

3.5 *Fuel Channel and Components*

The fuel channel assembly design criteria are summarized below, and evaluation results are summarized in Table 3-2. The analysis methods are described in detail in Reference 2.

3.5.1 Design Criteria for Normal Operation

Stress due to Pressure Differential. The stress limits during normal operation are obtained from the ASME B&PV Code, Section III, Division 1, Subsection NG for Service Level A. The calculated stress intensities are due to the differential pressure across the fuel channel wall. The pressure loading includes the normal operating pressure plus the increase during AOO. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation (Reference 7). As an alternative to the elastic analysis stress intensity limits, a plastic analysis may be performed as permitted by paragraph NB-3228.3 of the ASME B&PV Code.

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

Fatigue. Cyclic changes in power and flow during operation impose a duty loading on the fuel channel. The cyclic duty from pressure fluctuations is limited to less than the fatigue lifetime of the fuel channel. The fatigue life is based on the O’Donnell and Langer curve (Reference 6), which includes a factor of 2 on stress amplitude or a factor of 20 on the number of cycles, whichever is more conservative.

Oxidation and Hydriding. Oxidation reduces the material thickness and results in less load-carrying capacity. The fuel channels have thicker walls than other components (e.g., fuel rods), and the normal amounts of oxidation and hydrogen pickup are not limiting provided: the alloy composition and impurity limits are carefully selected; the heat treatments are also carefully chosen; and the water chemistry is controlled. [

] .

Long-Term Deformation. Changes to the geometry of the fuel channel occur due to creep deformation during the long term exposure in the reactor core environment. Overall deformation of the fuel channel occurs from a combination of bulging and bowing. Bulging of the side walls occurs because of the differential pressure across the wall. Lateral bowing of the channel is caused primarily from the neutron flux and thermal gradients. Too much deflection may prevent normal control blade maneuvers and it may increase control blade insertion time above the Technical Specification limits. The total channel deformation must not stop free movement of the control blade. [

] .

3.5.2 Design Criteria for Accident Conditions

Fuel Channel Stresses, Load Limit, and Vertical Acceleration. The criteria are based on the ASME B&PV Code, Section III, Appendix F, for faulted conditions (Service Level D). Component support criteria for elastic system analysis are used as defined in paragraphs F-1332.1 and

F-1332.2. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation (Reference 7). [

] . Vertical acceleration produces a membrane stress in the axial direction due to a postulated impact of the channeled fuel assembly impacting the fuel support after liftoff.

The amount of bulging remains limited to that value which will permit control blade insertion.

Channel Bending from Combined Horizontal Excitations. [

] .

Fuel Channel Gusset Strength. [

] .

Table 3-1
Results for ATRIUM 11 Fuel Assembly Criteria

Criteria Section	Description	Criteria	Results
ANF-89-98(P)(A) (Reference 1) Associated Mechanical Design Criteria Sections			
3.3	**Fuel System Criteria**		
3.3.1	Stress, strain and loading limits on assembly components	The ASME B&PV Code Section III is used to establish acceptable stress levels or load limits for assembly structural components. The design limits for accident conditions are derived from Appendix F of Section III.	[ ].
3.3.3	Fretting wear	[ ].	[ ].
3.3.5	Rod bow	Protect thermal limits	[ ]
3.3.6	Axial irradiation growth		
	Upper end cap clearance	Clearance always exists	[ ]
3.3.8	Assembly lift-off		
	Normal operation (including AOOs)	No lift-off from fuel support	[ ]
	Postulated accident	No disengagement from fuel support	[ ].

Table 3-1
Results for ATRIUM 11 Fuel Assembly Criteria
(Continued)

Criteria Section	Description	Criteria	Results
3.3	**Fuel System Criteria** (Continued)		
3.3.9	Fuel assembly handling	[ ]	Verified by testing and Analyses to meet requirement
3.3.10	Miscellaneous components		
3.3.10.1	Compression spring forces	Support weight of UTP and fuel channel throughout design life	The design criteria are met
3.3.10.2	LTP seal spring	Accommodate fuel channel deformation, adequate corrosion, and withstand operating stresses	The design criteria are met
3.4	**Fuel Coolability**		
3.4.4	Structural deformations	Maintain coolable geometry and ability to insert control blades. SRP 4.2, App. A, and ASME Section III, App. F.	[ ].

Table 3-2
Results for ATRIUM 11 Advanced Fuel Channel Criteria

Criteria Section	Description	Criteria	Results
EMF-93-177(P)(A) (Reference 2) Associated Fuel Channel (FC) Criteria Sections			
FC 3.2	**ATRIUM 11 Advanced Fuel Channel – Normal Operation**		
FC 3.2.1	Stress due to pressure differential	The pressure load including AOO is limited to [ ] according to ASME B&PV Code, Section III. The pressure load is also limited such that [ ].	The deformation during AOO remains within functional limits for normal control blade operation and the [ ] is met. There is no significant plastic deformation.
FC 3.2.2	Fatigue	Cumulative cyclic loading to be less than the design cyclic fatigue life for Zircaloy.	Expected number of cycles is less than allowable
FC 3.2.3	Oxidation and hydriding	Oxidation shall be accounted for in the stress and fatigue analyses	The maximum expected oxidation is low in relation to the wall thickness. Oxidation was accounted for in the stress and fatigue analyses.
FC 7.0	Long-term deformation (bulge creep and bow)	Bulge and bow shall not interfere with free movement of the control blade	Margin to a stuck control blade remains positive

Table 3-2
Results for ATRIUM 11 Advanced Fuel Channel Criteria
(Continued)

Criteria Section	Description	Criteria	Results
FC 3.3	**ATRIUM 11 Advanced Fuel Channel – Accident Conditions**		
FC 3.3.1	Fuel channel stresses and load limit and vertical accelerations	The pressure load is limited to [ ]. The pressure load is also limited such that [ ].	The deformation during blowdown does not interfere with control blade insertion. This also satisfies the less restrictive [ ].
FC 3.3.1 (continued)	Channel bending from combined horizontal excitations	Allowable bending moment based on ASME Code, Section III, Appendix F [ ].	[ ]
FC 3.3.2	Fuel channel gusset strength	Vertical load must be less than ASME allowable load rating based on testing.	[ ].

4.0 REFERENCES

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

2. EMF-93-177(P)(A) Revision 1, "Mechanical Design for BWR Fuel Channels," Framatome ANP Inc., August 2005.

3. EMF-93-177P-A Revision 1, Supplement 1P-A, Revision 0, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," AREVA NP Inc., September 2013.

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

5. ANP-3653P, Revision 0, "Fuel Design Evaluation for ATRIUM 11 BWR Reload Fuel," Information Design Report, Framatome Inc., September 2018.

6. W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering, Volume 20, January 1964.

7. Huang, P. Y., Mahmood, S. T., and Adamson, R. B. "Effects of Thermomechanical Processing on In-Reactor Corrosion and Post-Irradiation Properties of Zircaloy-2," *Zirconium in the Nuclear Industry: Eleventh International Symposium*, ASTM STP 1295, E. R. Bradley and G. P. Sabol, Eds., American Society for Testing and Materials, 1996, pp. 726-757.

Enclosure 9c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3762P, Mechanical Design Report
for Susquehanna ATRIUM 11 Fuel Assemblies

A F F I D A V I T

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle Elliott. I am Deputy Director, Licensing & Regulatory Affairs, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in Licensing Report ANP-3762P, Revision 0, entitled, "Mechanical Design Report for Susquehanna ATRIUM 11 Fuel Assemblies," dated May 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 24th

day of May, 2019.

ELLA FRANCES CARR-PAYNE
NOTARY PUBLIC
REG# 309873
MY COMMISSION EXPIRES
08/31/2021
COMMONWEALTH OF VIRGINIA

Enclosure 10b of PLA-7783

Framatome Topical Report
ANP-3761NP

Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies

(Non-Proprietary Version)

Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies

ANP-3761NP
Revision 0

May 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

Tables

Figures

Nomenclature

AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
LOCA	loss-of-coolant accident
LTP	lower tie plate
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
NRC	Nuclear Regulatory Commission, U.S.
PLFR	part-length fuel rod
RPF	radial peaking factor
UTP	upper tie plate

1.0 Introduction

The results of Susquehanna thermal-hydraulic analyses are presented to demonstrate that Framatome ATRIUM 11 fuel is hydraulically compatible with the previously loaded ATRIUM-10 fuel design. This report also provides the hydraulic characterization of the ATRIUM 11 and the coresident ATRIUM-10 design for Susquehanna.

The generic thermal-hydraulic design criteria applicable to the design have been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) in the topical report ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1). In addition, thermal-hydraulic criteria applicable to the design have also been reviewed and approved by the NRC in the topical report XN-NF-80-19(P)(A) Volume 4 Revision 1 (Reference 2).

2.0 Summary and Conclusions

ATRIUM 11 fuel assemblies have been determined to be hydraulically compatible with the coresident ATRIUM-10 fuel design at Susquehanna Units 1 and 2 for the entire range of the licensed power-to-flow operating map. Detailed calculation results supporting this conclusion are provided in Section 3.2 and Tables 3.4–3.8.

The ATRIUM 11 fuel design is geometrically different from the coresident ATRIUM-10 design, but the designs are hydraulically compatible. **[**

]

Core bypass flow (defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface) is not adversely affected by the introduction of the ATRIUM 11 fuel design. Analyses at rated conditions show a core bypass flow (excluding water rod flow) of **[ ]** of rated core flow for a full core of ATRIUM-10 fuel and **[ ]** for transition core configurations and a full core of ATRIUM 11 fuel.

Analyses demonstrate the thermal-hydraulic design and compatibility criteria discussed in Section 3.0 are satisfied for the Susquehanna Units 1 and 2 cores consisting of ATRIUM-10 fuel with ATRIUM 11 fuel for the expected core power distributions and core power/flow conditions encountered during operation.

3.0 Thermal-Hydraulic Design Evaluation

Thermal-hydraulic analyses are performed to verify that design criteria are satisfied and to help establish thermal operating limits with acceptable margins of safety during normal reactor operation and anticipated operational occurrences (AOOs). The design criteria that are applicable to the ATRIUM 11 fuel design is described in Reference 1. To the extent possible, these analyses are performed on a generic fuel design basis. However, due to reactor and cycle operating differences, many of the analyses supporting these thermal-hydraulic operating limits are performed on a plant- and cycle-specific basis and are documented in plant- and cycle-specific reports.

The thermal-hydraulic design criteria are summarized below:

- **Hydraulic compatibility.** The hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to the existing fuel in the reactor such that there is no significant impact on total core flow or the flow distribution among assemblies in the core. This criterion evaluation is addressed in Sections 3.1 and 3.2.
- **Thermal margin performance.** Fuel assembly geometry, including spacer design and rod-to-rod local power peaking, should minimize the likelihood of boiling transition during normal reactor operation as well as during AOOs. The fuel design should fall within the bounds of the applicable empirically based boiling transition correlation approved for Framatome reload fuel. Within other applicable mechanical, nuclear, and fuel performance constraints, the fuel design should achieve good thermal margin performance. The thermal-hydraulic design impact on steady-state thermal margin performance is addressed in Section 3.3. Additional thermal margin performance evaluations dependent on the cycle-specific design are addressed in the reload licensing report.
- **Fuel centerline temperature.** Fuel design and operation shall be such that fuel centerline melting is not projected for normal operation and AOOs. This criterion evaluation is addressed in the fuel rod thermal and mechanical design report.
- **Rod bow.** The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margin requirements. This criterion evaluation is addressed in Section 3.4.
- **Bypass flow.** The bypass flow characteristics of the reload fuel assemblies shall not differ significantly from the existing fuel in order to provide adequate flow in the bypass region. This criterion evaluation is addressed in Section 3.5.
- **Stability.** Reactors fueled with new fuel designs must be stable in the power and flow operating region. The stability performance of new fuel designs will be equivalent to, or better than, existing (approved) Framatome fuel designs. This criterion evaluation is addressed in Section 3.6. Additional core stability evaluations dependent on the cycle-specific design are addressed in the reload licensing report.

- **Loss-of-coolant accident (LOCA) analysis.** LOCAs are analyzed in accordance with Appendix K modeling requirements using NRC-approved models. The criteria are defined in 10 CFR 50.46. LOCA analysis results are presented in the break spectrum and MAPLHGR report.
- **Control rod drop accident (CRDA) analysis.** The deposited enthalpy must be less than the applicable criteria. This criterion evaluation is addressed in the reload licensing report.
- **ASME overpressurization analysis.** ASME pressure vessel code requirements must be satisfied. This criterion evaluation is addressed in the reload licensing report.
- **Seismic/LOCA liftoff.** Under accident conditions, the assembly must remain engaged in the fuel support. This criterion evaluation is addressed in the mechanical design report.

A summary of the thermal-hydraulic design evaluations is given in Table 3.1.

3.1 *Hydraulic Characterization*

Basic geometric parameters for the ATRIUM 11 and ATRIUM-10 fuel designs are summarized in Table 3.2. Component loss coefficients for the fuels mentioned are based on tests and are presented in Table 3.3. These loss coefficients include modifications to the test data reduction process **[**

]. The bare rod friction, ULTRAFLOW spacer, UTP and LTP losses for ATRIUM 11 and ATRIUM-10 are based on tests performed at Framatome's Portable Hydraulic Test Facility. **[**

]

The primary resistance for the leakage flow through the LTP flow holes is **[**

]. The resistances for the leakage paths are shown in Table 3.3.

3.2 ***Hydraulic Compatibility***

The thermal-hydraulic analyses were performed in accordance with the Framatome thermal-hydraulic methodology for BWRs. The methodology and constitutive relationships used by Framatome for the calculation of pressure drop in BWR fuel assemblies are presented in Reference 3 and are implemented in the XCOBRA code. The XCOBRA code predicts steady-state thermal-hydraulic performance of the fuel assemblies of BWR cores at various operating conditions and power distributions. XCOBRA received NRC approval in Reference 4. The NRC reviewed the information provided in Reference 5 regarding inclusion of water rod models in XCOBRA and accepted the inclusion in Reference 6.

Hydraulic compatibility, as it relates to the relative performance of the ATRIUM 11 and coresident ATRIUM-10 fuel designs, has been evaluated. Detailed analyses were performed for full cores of each fuel design present herein. Analyses for mixed cores with ATRIUM 11 and ATRIUM-10 fuel were also performed to demonstrate that the thermal-hydraulic design criteria are satisfied for transition core configurations, and thus the fuel assemblies are compatible.

The hydraulic compatibility analysis is based on **[**

]

Table 3.4 summarizes the input conditions for the analyses. These conditions reflect two of the state points considered in the analyses: 100% power/100% flow and 57% power/40% flow. Table 3.4 also defines the core loading for the transition core configurations. Input for other core configurations is similar in that core operating conditions remain the same and the same axial power distribution is used. Evaluations were made with the bottom-, middle-, and top-peaked axial power distributions presented in Figure 3.1. Results presented in this report are for the middle-peaked power distribution. Results for bottom- and top-peaked axial power distributions show similar trends.

Table 3.5 and Table 3.6 provide a summary of calculated thermal-hydraulic results using the first transition core configuration. Table 3.7 and Table 3.8 provide a summary of results for all

core configurations evaluated. Core average results and the differences between the ATRIUM 11 and ATRIUM-10 results at rated power are within the range which is considered compatible. Similar agreement occurs at lower power levels. As shown in Table 3.5, **[**

]. Differences in assembly flow between the ATRIUM 11 and ATRIUM-10 fuel designs as a function of assembly power level are shown in Figure 3.2 and Figure 3.3.

Core pressure drop and core bypass flow fraction are also provided for the configurations evaluated. Based on the reported changes in pressure drop and assembly flow caused by the introduction of ATRIUM 11, the ATRIUM 11 design is considered hydraulically compatible with the coresident fuel designs since the thermal-hydraulic design criteria are satisfied.

3.3 *Thermal Margin Performance*

Relative thermal margin analyses were performed in accordance with the thermal-hydraulic methodology for Framatome's XCOBRA code. The calculation of the fuel assembly critical power ratio (CPR) (thermal margin performance) is established by means of an empirical correlation based on results of boiling transition test programs. The CPR methodology is the approach used by Framatome to determine the margin to thermal limits for BWRs.

CPR values for ATRIUM 11 are calculated with the ACE/ATRIUM 11 critical power correlation (Reference 7) while the CPR values for the ATRIUM-10 are calculated with the SPCB critical power correlation (Reference 8). Assembly design features are incorporated in the CPR calculation through the K-factor term in the ACE correlation and F-eff term in the SPCB correlation. The K-factors and F-eff are based on the local power peaking for the nuclear design and on additive constants determined in accordance with approved procedures. The local peaking factors are a function of assembly void fraction and exposure.

For the compatibility evaluation, steady-state analyses evaluated ATRIUM 11 and ATRIUM-10 assemblies with radial peaking factors (RPFs) between **[**

]. Table 3.5 and Table 3.6 show CPR results of the ATRIUM 11 and ATRIUM-10

fuel. Table 3.7 and Table 3.8 show similar comparisons of CPR and assembly flow for the various core configurations evaluated. Analysis results indicate ATRIUM 11 fuel will not cause thermal margin problems for the coresident ATRIUM-10 fuel design.

3.4 *Rod Bow*

The bases for rod bow are discussed in the mechanical design report. Rod bow magnitude is determined during the fuel-specific mechanical design analyses and confirmed on a cycle-specific basis.

[

]

3.5 *Bypass Flow*

Total core bypass flow is defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface. Table 3.7 shows that total core bypass flow (excluding water rod flow) fraction at rated conditions is **[ ]** of rated core flow for the core configurations presented (middle-peaked power shape). In summary, adequate bypass flow will be available with the introduction of the ATRIUM 11 and applicable design criteria are met.

3.6 *Stability*

Each new fuel design is analyzed to demonstrate that the stability performance is equivalent to or better than an existing Framatome fuel design. The stability performance is a function of the core power, core flow, core power distribution, and to a lesser extent, the fuel design. **[**

] A comparative stability analysis was performed with the NRC-approved STAIF code (Reference 9). The study shows that the ATRIUM 11 fuel design has decay ratios equivalent to or better than other Framatome fuel designs.

As stated above, the stability performance of a core is strongly dependent on the core power, core flow, and power distribution in the core. Therefore, core stability is evaluated on a cycle-specific basis and addressed in the reload licensing report.

Table 3.1 Design Evaluation of Thermal and Hydraulic Criteria for the ATRIUM 11 Fuel Assembly

Report Section	Description	Criteria	Results or Disposition
	Thermal and Hydraulic Criteria		
3.1 / 3.2	Hydraulic compatibility	Hydraulic flow resistance shall be sufficiently similar to existing fuel such that there is no significant impact on total core flow or flow distribution among assemblies.	Verified on a plant-specific basis. ATRIUM 11 demonstrated to be compatible with ATRIUM-10 fuel.
3.3	Thermal margin performance	Fuel design shall be within the limits of applicability of an approved CHF correlation.	ACE/ATRIUM 11 critical power correlation is applied to the ATRIUM 11 fuel. SPCB critical power correlation is applied to the ATRIUM-10 fuel.
		< 0.1% of rods in boiling transition.	Verified on cycle-specific basis for Chapter 15 analyses.
	Fuel centerline temperature	No centerline melting.	Plant- and fuel-specific analyses are performed.
3.4	Rod bow	Rod bow must be accounted for in establishing thermal margins.	The lateral displacement of the fuel rods due to fuel rod bowing is not of sufficient magnitude to impact thermal margins. Verified on a cycle-specific basis.
3.5	Bypass flow	Bypass flow characteristics shall be similar among assemblies to provide adequate bypass flow.	Verified on a plant-specific basis. Analysis results demonstrate that adequate bypass flow is provided.

Table 3.1 Design Evaluation of Thermal and Hydraulic Criteria for the ATRIUM 11 Fuel Assembly *(Continued)*

Report Section	Description	Criteria	Results or Disposition
	Thermal and Hydraulic Criteria *(Continued)*		
3.6	Stability	New fuel designs are stable in the approved power and flow operating region, and stability performance will be equivalent to (or better than) existing (approved) Framatome fuel designs.	ATRIUM 11 channel and core decay ratios have been demonstrated to be equivalent to or better than other approved Framatome fuel designs. Core stability behavior is evaluated on a cycle-specific basis.
	LOCA analysis	LOCA analyzed in accordance with Appendix K modeling requirements. Criteria defined in 10 CFR 50.46.	Plant- and fuel-specific analysis is performed with Appendix K LOCA models and verified with cycle specific calculations.
	CRDA analysis	Applicable criteria	Cycle-specific analysis is performed.
	ASME over-pressurization analysis	ASME pressure vessel core requirements shall be satisfied.	Cycle-specific analysis is performed.
	Seismic/LOCA liftoff	Assembly remains engaged in fuel support.	Plant- and fuel-specific analyses are performed.

Table 3.2 Comparative Description of Susquehanna ATRIUM 11 and ATRIUM-10 Fuel Types

Fuel Parameter	ATRIUM-10	ATRIUM 11
Number of fuel rods		
Full-length fuel rods	83	92
PLFRs	8	
Short PLFRs		12
Long PLFRs		8
Fuel clad OD, in	0.3957	0.3701
Number of spacers	8	9
Active fuel length, ft		
Full-length fuel rods	12.45	12.50
PLFRs	7.50	
Short PLFRs		4.66
Long PLFRs		7.34
Hydraulic resistance characteristics	Table 3.3	Table 3.3
Number of water rods	1	1
Water rod OD, in	1.378*	1.300*

* Square water channel outer width.

Table 3.3 Hydraulic Characterization Comparison Between Susquehanna ATRIUM-10 and ATRIUM 11 Fuel

[

]

[

]

Table 3.4 Susquehanna
Thermal-Hydraulic Design Conditions

Reactor Conditions	100%P / 100%F	57%P / 40%F
Core power level, MWt	3952.0	2252.6
Core exit pressure, psia	1062.1	988.1
Core inlet enthalpy, Btu/lbm	523.7	495.0
Total core coolant flow, Mlbm/hr	100.0	40.0
Axial power shape	Middle-peaked (Figure 3.1)	Middle-peaked (Figure 3.1)

	Number of Assemblies Central Region	Number of Assemblies Peripheral Region
First Transition Core Loading		
[		]
[		]
Second Transition Core Loading		
[		]
[		]

Table 3.5 Susquehanna First Transition Core Thermal-Hydraulic Results at Rated Conditions (100%P / 100%F)

[

]

[

]

**Table 3.6 Susquehanna
First Transition Core Thermal-Hydraulic Results at
Off-Rated Conditions (57%P / 40%F)**

[

]

[

]

Table 3.7 Susquehanna Thermal-Hydraulic Results at Rated Conditions (100%P / 100%F) for Transition to ATRIUM 11 Fuel

[

]

Table 3.8 Susquehanna Thermal-Hydraulic Results at Off-Rated Conditions (57%P / 40%F) for Transition to ATRIUM 11 Fuel

[

]

[

]

Figure 3.1 Axial Power Shapes

[

]

**Figure 3.2 First Transition Core:
Hydraulic Demand Curves 100%P/100%F**

[

]

**Figure 3.3 First Transition Core:
Hydraulic Demand Curves 57%P/40%F**

4.0 References

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

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

3. XN-NF-79-59(P)(A), *Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies*, Exxon Nuclear Company, November 1983.

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

5. Letter, R.A. Copeland (ANF) to R.C. Jones (USNRC), "Explicit Modeling of BWR Water Rod in XCOBRA," RAC:002:90, January 9, 1990.

6. Letter, R.C. Jones (USNRC) to R.A. Copeland (ANF), no subject (regarding XCOBRA water rod model), February 1, 1990.

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

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

9. EMF-CC-074(P)(A) Volume 1, *STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain;* and Volume 2, *STAIF – A Computer Program for BWR Stability Analysis in the Frequency Domain – Code Qualification Report*, Siemens Power Corporation, July 1994.

Enclosure 10c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3761P, Susquehanna Units 1 and 2
Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3761P Revision 0, "Susquehanna Units 1 and 2 Thermal-Hydraulic Design Report for ATRIUM 11 Fuel Assemblies," dated May 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 30th

day of May, 2019.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

Enclosure 11b of PLA-7783

Framatome Topical Report ANP-3745NP

ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Susquehanna LAR

(Non-Proprietary Version)

ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Susquehanna LAR

ANP-3745NP
Revision 0

Licensing Report

March 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
3GFG	3rd generation FUELGUARD
AOO	anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BOL	beginning of life
BWR	boiling water reactor
CRWE	control rod withdrawal error
CUF	cumulative usage factor
EOL	end of life
FDL	fuel design limit
ID	inside diameter
LAR	License Amendment Request
LHGR	linear heat generation rate
LTP	lower tie plate
MWd/kgU	megawatt days per kilogram of initial uranium
NRC	Nuclear Regulatory Commission, U.S.
OD	outside diameter
PCI	pellet-to-cladding-interaction
PLFR	part length fuel rod
ppm	parts per million
SRA	stress relieved annealed
S-N	stress amplitude versus number of cycles
UTL	upper tolerance limit

1.0 INTRODUCTION

This document reports the results of thermal-mechanical analyses for the performance of ATRIUM 11 fuel assemblies inserted into to an equilibrium cycle for the Susquehanna units and demonstrates that the design criteria relevant to these limits are satisfied. This report is intended to support a License Amendment Request (LAR) for the approval to use the Framatome advanced analysis methods that will be deployed coincident with the implementation of the ATRIUM 11 fuel assembly design. These analyses assume the use of chromia additive in the enriched urania portions of the fuel. Both the design criteria and the analysis methodology have been approved by the U.S. NRC (NRC).

The analysis results are evaluated according to the generic fuel rod thermal and mechanical design criteria contained in ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1) along with design criteria provided in the RODEX4 fuel rod thermal-mechanical topical report (Reference 2). The cladding external oxidation limit defined by Reference 2 is **[**

]. Approved methodology for the inclusion of chromia additive in the fuel pellets is also used (Reference 3).

The RODEX4 fuel rod thermal-mechanical analysis code is used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue and external oxidation. The code and application methodology are described in the RODEX4 topical report (Reference 2). The cladding steady-state stress and plenum spring design methodology are summarized in Reference 1.

The following sections describe the fuel rod design, design criteria and methodology with reference to the source topical reports. Results from the analyses are summarized for comparison to the design criteria.

2.0 SUMMARY AND CONCLUSIONS

Key results are compared against each design criterion in Table 3-2. Results are presented for the limiting cases. Additional RODEX4 results are given in Section 3.0.

The analyses support a maximum fuel rod discharge exposure of 62 MWd/kgU.

Fuel rod criteria applicable to the design are summarized in Section 3.0. Analyses show the criteria are satisfied when the fuel is operated at or below the LHGR (linear heat generation rate) limit (Fuel Design Limit – FDL) presented in Figure 2-1.

Table 2-1 Summary of Fuel Rod Design Evaluation Results

Criteria Section*	Description	Criteria	Result, Margin† or Comment
3.2	**Fuel Rod Criteria**		
3.2.1	Internal hydriding	**[**	**]**
(3.1.1)	Cladding collapse	**[**	**]**
(3.1.2)	Overheating of fuel pellets	No fuel melting margin to fuel melt > 0. °C	**[ ]**
3.2.5	Stress and strain limits		
(3.1.1) (3.1.2)	Pellet-cladding interaction	**[**	**]**
3.2.5.2	Cladding steady-state stresses	**[**	**]**
3.3	**Fuel System Criteria**		
(3.1.1)	Fatigue	**[**	**]**
(3.1.1)‡	Oxidation, hydriding, and crud buildup	**[**	**]**
(3.1.1) (3.1.2)	Rod internal pressure	**[**	**]**
3.3.9	Fuel rod plenum spring (fuel handling)	Plenum spring to **[**	**]**

* Numbers in the column refer to paragraph sections in the generic design criteria document, ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1). A number in parentheses is the paragraph section in the RODEX4 fuel rod topical report (Reference 2).

† Margin is defined as (limit – result).

‡ The cladding external oxidation limit is restricted to the reduced value of **[ ]** based on the NRC review of the RODEX4 first implementation in the U.S.

[

]

Figure 2-1 LHGR Limit (Normal Operation)

3.0 FUEL ROD DESIGN EVALUATION

Summaries of the design criteria and methodology are provided in this section along with analysis results in comparison to criteria. Both the fuel rod criteria and fuel system criteria as directly related to the fuel rod analyses are covered.

The fuel rod analyses cover normal operating conditions and AOOs (anticipated operational occurrences). The fuel centerline temperature analysis (overheating of fuel) and cladding strain analysis take into account slow transients at rated operating conditions.

Other fuel rod-related topics on overheating of cladding, cladding rupture, fuel rod mechanical fracturing, rod bow, axial irradiation growth, cladding embrittlement, violent expulsion of fuel and fuel ballooning are evaluated as part of the respective fuel assembly structural analysis, thermal hydraulic analyses, or LOCA analyses and are reported elsewhere. The evaluation of fast transients and transients at off-rated conditions also are reported separately from this report.

3.1 *Fuel Rod Design*

The ATRIUM 11 fuel rod is conventional in design configuration and very similar to past designs such as the ATRIUM 10XM and ATRIUM-10 fuel rods.

[

] 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. The longer is **[ ]** long, while the shorter is **[ ]** long. **[**

] .

[

]

As on previous ATRIUM fuel designs that incorporated the 3rd generation FUELGUARD (3GFG) Lower Tie Plate (LTP), the PLFR's have a **[**

]

Table 3-1 lists the main parameters for the fuel rod and components.

3.2 *RODEX4 and Statistical Methodology Summary*

RODEX4 evaluates the thermal-mechanical response of the fuel rod surrounded by coolant. The fuel rod model considers the fuel column, gap region, cladding, gas plena and the fill gas and released fission gases. The fuel rod is divided into axial and radial regions with conditions computed for each region. The operational conditions are controlled by the **[**

].

The heat conduction in the fuel and clad is **[**

].

Mechanical processes include **[**

].

As part of the methodology, fuel rod power histories are generated **[**

].

Since RODEX4 is a best-estimate code, uncertainties are taken into account by a [

]. Uncertainties taken into account in the analysis are summarized as:

- Power measurement and operational uncertainties – [ ].
- Manufacturing uncertainties – [ ].
- Model uncertainties – [ ].

[ ].

3.3 *Summary of Fuel Rod Design Evaluation*

Results from the analyses are listed in Table 3-2. Summaries of the methods and codes used in the evaluation are provided in the following paragraphs. The design criteria also are listed along with references to the sections of the design criteria topical reports (References 1 and 2).

The fuel rod thermal and mechanical design criteria are summarized as follows.

- **Internal Hydriding**. The fabrication limit **[ ]** to preclude cladding failure caused by internal sources of hydrogen (Section 3.2.1 of Reference 1).

- **Cladding Collapse**. Clad creep collapse shall be prevented. **[ ]** (Section 3.1.1 of Reference 2).

- **Overheating of Fuel Pellets**. The fuel pellet centerline temperature during anticipated transients shall remain below the melting temperature (Section 3.1.2 of Reference 2).

- **Stress and Strain Limits**. **[ ]** during normal operation and during anticipated transients (Sections 3.1.1 and 3.1.2 of Reference 2).

  Fuel rod cladding steady-state stresses are restricted to satisfy limits derived from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Section 3.2.5.2 of Reference 1).

- **Cladding Fatigue**. The fatigue cumulative usage factor for clad stresses during normal operation and design cyclic maneuvers shall be below **[ ]** (Section 3.1.1 of Reference 2).

- **Cladding Oxidation, Hydriding and Crud Buildup**. Section 3.1.1 of Reference 2 limits the maximum cladding oxidation to less than **[ ]** to prevent clad corrosion failure. The oxidation limit is further reduced to **[ ]**.

- **Rod Internal Pressure**. The rod internal pressure is limited **[ ]** to ensure that significant outward clad creep does not occur and unfavorable hydride reorientation on cooldown does not occur (Section 3.1.1 of Reference 2).

- **Plenum Spring Design (Fuel Handling)**. The rod plenum spring must maintain a force against the fuel column stack **[ ]** (Section 3.3.9 of Reference 1).

Cladding collapse, overheating of fuel, cladding transient strain, cladding cyclic fatigue, cladding oxidation, and rod pressure are evaluated **[ ]**. Cladding stress and the plenum spring are evaluated **[ ]**.

3.3.1 Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. Careful moisture control during fuel fabrication reduces the potential for hydrogen absorption on the inside of the cladding. The fabrication limit **[** **]** is verified by quality control inspection during fuel manufacturing.

3.3.2 Cladding Collapse

Creep collapse of the cladding and the subsequent potential for fuel failure is avoided in the design by limiting the gap formation due to fuel densification subsequent to pellet-clad contact. The size of the axial gaps which may form due to densification following first pellet-clad contact shall be less than **[** **]**.

The evaluation is performed using the RODEX4 code and methodology. RODEX4 takes into account the **[** **]**.

Table 3-2 lists the results for an equilibrium cycle.

3.3.3 Overheating of Fuel Pellets

Fuel failure from the overheating of the fuel pellets is not allowed. The centerline temperature of the fuel pellets must remain below melting during normal operation and AOOs. The melting point of the fuel includes adjustments for **[** **]**. Framatome establishes an LHGR limit to protect against fuel centerline melting during steady-state operation and during AOOs.

Fuel centerline temperature is evaluated using the RODEX4 code and methodology for both normal operating conditions and AOOs.

Table 3-2 lists the results for an equilibrium cycle.

3.3.4 Stress and Strain Limits

3.3.4.1 Pellet/Cladding Interaction

Cladding strain caused by transient-induced deformations of the cladding is calculated using the RODEX4 and methodology. [

]. The strain limit is 1%.

Table 3-2 lists the results for an equilibrium cycle.

3.3.4.2 Cladding Stress

Cladding stresses are calculated using solid mechanics elasticity solutions and finite element methods. The stresses are conservatively calculated for the individual loadings and are categorized as follows:

Category	Membrane	Bending
Primary	[	]
Secondary	[	]

Stresses are calculated at the cladding outer and inner diameter in the three principal directions for both beginning of life (BOL) and end of life (EOL) conditions. At EOL, the stresses due to mechanical bow and contact stress are decreased due to irradiation relaxation. The separate stress components are then combined, and the stress intensities for each category are compared to their respective limits.

The cladding-to-end cap weld stresses are evaluated for loadings from differential pressure, differential thermal expansion, rod weight, and plenum spring force.

The design limits are derived from the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel (B&PV) Code Section III (Reference 4) and the minimum specified material properties.

Table 3-3 lists the results in comparison to the limits for Beginning-of-Life (BOL) Hot conditions and End-of-Life (EOL) at both Hot and Cold conditions.

3.3.5 Fuel Densification and Swelling

Fuel densification and swelling are limited by the design criteria for fuel temperature, cladding strain, cladding collapse, and rod internal pressure criteria. Although there are no explicit criteria for fuel densification and swelling, the effect of these phenomena are included in the RODEX4 code and methodology.

3.3.6 Fatigue

Fuel rod cladding fatigue is calculated using the RODEX4 code and methodology. **[**

]. The CUF (cumulative usage factor) is summed for each of the axial regions of the fuel rod using Miner's rule. The axial region with the highest CUF is used in the subsequent **[**

]. The maximum CUF for the cladding must remain below **[ ]** to satisfy the design criterion. Table 3-2 lists the results for an equilibrium cycle.

3.3.7 Oxidation, Hydriding, and Crud Buildup

Cladding external oxidation is calculated using the RODEX4 code and methodology. The corrosion model includes an enhancement factor that is derived from poolside measurement data to obtain a fit of the expected oxide thickness. An uncertainty value for the model enhancement factor also is determined from the data. The model uncertainty is included as part of the **[ ]**.

[

]

[

].

In the event abnormal crud is observed at a plant, a specific analysis is required to address the higher crud level. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25°C above the design basis calculation. The formation of crud is not calculated within RODEX4. Instead, an upper bound of expected crud based on plant observations is input by the use of the crud heat transfer coefficient. The corrosion model also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate. A higher corrosion rate is therefore included as part of the abnormal crud evaluation. A similar specific analysis is required if an abnormal corrosion layer is observed instead of crud.

In the case of the Susquehanna units, no additional crud is taken into account in the calculations because an abnormal crud or corrosion layer (beyond the design basis) has not been observed at the Susquehanna units.

[

].

Currently, [

].

The oxide limit is evaluated such that greater than [

].

Table 3-2 lists the results for an equilibrium cycle.

3.3.8 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX4 code and methodology. The maximum rod pressure is calculated under steady-state conditions and also takes into account slow transients. Rod internal pressure is limited to [ ]. The expected upper bound of rod pressure [ ] is calculated for comparison to the limit.

Table 3-2 lists the results for an equilibrium.

3.3.9 Plenum Spring Design (Fuel Assembly Handling)

The plenum spring must maintain a force against the fuel column to prevent [ ]. This is accomplished by designing and verifying the spring force in relation to the fuel column weight. The plenum spring is designed such that the [ ].

Table 3-1 Key Fuel Rod Design Parameters, ATRIUM 11 for Susquehanna LAR

[

]

* The theoretical density of enriched UO_2-Cr is **[ ]** g/cm^3 while that for UO_2-Gd_2O_3 and naturally enriched UO_2 is **[ ]** g/cm^3.

Table 3-1 Key Fuel Rod Design Parameters, ATRIUM 11 for Susquehanna LAR (cont'd)

[

]

Table 3-2 RODEX4 Fuel Rod Results Equilibrium Cycle*

[

]

* Note that the results are provided up to fuel assembly discharge.
† Margin is defined as (limit – result).

Table 3-3 Cladding and Cladding-End Cap Steady-State Stresses

Description, Stress Category	Criteria	Result		
		BOL Cold	BOL Hot	EOL Hot
Cladding stress				
P_m (primary membrane stress)	[			]
$P_m + P_b$ (primary membrane + bending)	[			]
P + Q (primary + secondary)	[			]
Cladding-End Cap stress				
$P_m + P_b$	[			]

4.0 REFERENCES

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

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

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

4. *ASME Boiler and Pressure Vessel Code*, Section III, “Rules for Construction of Nuclear Power Plant Components,” 1977.

5. O’Donnell, W.J., and B. F. Langer, “Fatigue Design Basis for Zircaloy Components,” *Nuclear Science and Engineering*, Vol. 20, 1964.

Enclosure 11c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3745P, ATRIUM 11 Fuel Rod
Thermal-Mechanical Evaluation for Susquehanna LAR

A F F I D A V I T

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Philip A. Opsal. I am Manager, Product Licensing, for Framatome Inc., (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome Inc., to determine whether certain Framatome Inc. information is proprietary. I am familiar with the policies established by Framatome Inc. to ensure the proper application of these criteria.

3. I am familiar with the Framatome Inc. (formally AREVA Inc.) information contained in the following Document (herein referred to as This Document): ATRIUM 11 Fuel Rod Thermal-Mechanical Evaluation for Susquehanna LAR, Licensing Report ANP-3745P, Revision 0

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome Inc. and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome Inc. to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome Inc.'s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome Inc.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome Inc. in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome Inc., would be helpful to competitors to Framatome Inc., and would likely cause substantial harm to the competitive position of Framatome Inc.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c), 6(d) and 6(e) above.

7. In accordance with Framatome Inc.'s policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside Framatome Inc. only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome Inc.'s policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

____

SUBSCRIBED before me this 19

day of March, 2019.

Heidi H Elder

Heidi Hamilton Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/2022
Reg. # 7777873

HEIDI HAMILTON ELDER
COMMONWEALTH
REGISTRATION NO.
7777873
MY COMM. EXPIRES
12/31/2022
OF VIRGINIA
NOTARY PUBLIC

Enclosure 12b of PLA-7783

Framatome Topical Report
ANP-3727NP

Susquehanna ATRIUM 11
Equilibrium Cycle Fuel Cycle Design Report

(Non-Proprietary Version)

0414-12-F04 (Rev. 002, 01/15/2018)

framatome

Susquehanna ATRIUM 11 Equilibrium Cycle

ANP-3727NP
Revision 0

Fuel Cycle Design Report

October 2018

Framatome Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

CONTENTS

LIST OF TABLES

LIST OF FIGURES

Nomenclature

Acronym	**Definition**
ACE	Framatome critical power correlation
BOC	beginning of cycle
BOL	beginning of life
BWR	boiling water reactor
CSDM	cold shutdown margin
EOC	end of cycle
EOFP	end of full power capability
FFTR	final feedwater temperature reduction
GWd/MTU	gigawatt days per metric ton of initial uranium
HEXR	hot excess reactivity
LHGR	linear heat generation rate
MCPR	minimum critical power ratio
MICROBURN-B2	Framatome Inc. advanced BWR core simulator methodology with PPR capability
MWd/MTU	megawatt days per metric ton of initial uranium
NRC	(United States) Nuclear Regulatory Commission
PPR	Pin Power Reconstruction. The PPR methodology accounts for variation in local rod power distributions due to neighboring assemblies and control state. The local rod power distributions are reconstructed based on the actual flux solution for each statepoint.
R Value	the larger of zero or the shutdown margin at BOC minus the minimum calculated shutdown margin in the cycle
SLC	standby liquid control

1.0 INTRODUCTION

This report documents the Framatome Inc. equilibrium cycle design and the results from a representative Cycle N for the Susquehanna BWRs. This design analysis utilizes the ATRIUM 11 fuel design and has been performed with the approved Framatome Inc. neutronics methodology (References 1 and 4).

The CASMO-4 lattice depletion code was used to generate nuclear data including cross sections and local power peaking factors. The MICROBURN-B2 version 2 three dimensional core simulator code, combined with the application of the ACE critical power correlation (Reference 4), was used to model the core. The following MICROBURN-B2 version 2 modeling features were also used in the analyses supporting this report:

- Pin power reconstruction (PPR) to determine thermal margins
- **[ ]**
- **[ ]**
- **[ ]**
- **[ ]**

Design results including projected control rod patterns and evaluations of thermal and reactivity margins for the representative equilibrium Cycle N, hereafter identified at Cycle 25, are presented in this report.

2.0 SUMMARY

The equilibrium fresh fuel batch size [ ] and batch average enrichment [ ] were determined to meet the energy requirements approved by Talen Energy in Reference 3. The loading of the Cycle 25 fuel as described in this report results in a projected full power energy capability of [ ]. Beyond the nominal full power capability, Cycle 25 has been designed to achieve [ ] of additional energy via power coastdown operation.

In order to obtain optimum operating flexibility, the projected control rod patterns were developed with acceptable margin to thermal limits. The equilibrium cycle design calculations also demonstrate adequate hot excess reactivity and cold shutdown margin throughout the cycle. Key results from the Cycle 25 analysis are summarized in Table 2.1. Table 2.2 summarizes the assembly identification range for Cycle 25 by nuclear fuel type batch. Tables 2.3, 2.4, and 2.5 contain the assumed thermal limits for the equilibrium design. Figures 2.1 and 2.2 provide a summary of the Cycle 25 step-through projection.

Table 2.1 Cycle 25 Energy and Key Results Summary

Table 2.2 Cycle 25 Assembly ID Range by Nuclear Fuel Type

[ ]

Table 2.3 Susquehanna ATRIUM 11 Equilibrium Cycle Design - Assumed MCPR Operating Limit

[ ]

Table 2.4 Susquehanna ATRIUM 11 Equilibrium Cycle Design - Assumed LHGR Limit

[ ]

Table 2.5 Susquehanna ATRIUM 11 Equilibrium Cycle Design - Assumed APLHGR Limit

[

]

Figure 2.1 Cycle 25 Step-through k-eff versus Cycle Exposure

Figure 2.2 Cycle 25 Margin to Thermal Limits versus Cycle Exposure

3.0 CYCLE 25 FUEL CYCLE DESIGN

3.1 *General Description*

Elevation views of the equilibrium fuel design axial enrichment and gadolinia distributions are shown in Appendix B, Figures B.1 through B.3, and originate from Reference 5. The loading pattern maintains quarter core symmetry within a scatter load fuel management scheme. This loading, in conjunction with the control rod patterns presented in Appendix A, shows acceptable power peaking and associated margins to limits. The analyses supporting this equilibrium cycle design were based on the core parameters shown in Table 3.1. Figures 3.1 and 3.2, along with Table 3.1, define the reference loading pattern used in the representative equilibrium Cycle 25.

3.2 *Control Rod Patterns and Thermal Limits*

Projected control rod patterns and resultant key operating parameters including thermal margins from Cycle 25 are shown in Appendix A. The thermal margins presented in this report were determined using the MICROBURN-B2 3D core simulator PPR model to provide adequate margin to thermal limits. A detailed summary of the core parameters resulting from the step-through projection analysis for Cycle 25 is provided in Tables A.1 and A.2. Limiting results from the Cycle 25 step-through are summarized in Table 2.1 and in Figure 2.2. The hot operating target k-eff versus cycle exposure which was determined to be appropriate for this evaluation is shown in Table 3.2. The k-eff and margin to limits results from the Cycle 25 depletion are presented graphically in Figures 2.1 and 2.2. The k-eff values presented in Figure 2.1 and in Appendix A are not bias corrected. Selected exposure and radial power distributions from the Cycle 25 step-through are presented in Appendix C

3.3 *Hot Excess Reactivity and Cold Shutdown Margin*

The Cycle 25 calculations demonstrate adequate hot excess reactivity, SLC shutdown margin, and cold shutdown margin throughout the cycle. Key shutdown margin and R-Value results are presented in Table 2.1. The shutdown margin is in conformance with the Technical Specification limit of R + 0.38 %Δk/k at BOC. The cold target k-eff versus exposure determined to be appropriate for calculation of cold shutdown margin is shown in Table 3.3. The core hot excess reactivity was calculated **[**

] . Table 3.4 summarizes the reactivity margins versus cycle exposure, including the SLC shutdown margin for Cycle 25.

Table 3.1 Cycle 25 Core Composition and Susquehanna ATRIUM 11 Equilibrium Cycle Design Parameters

Table 3.2 Susquehanna ATRIUM 11 Equilibrium Cycle Design Hot Operating Target k-eff Versus Cycle Exposure

[ ]

Table 3.3 Susquehanna ATRIUM 11 Equilibrium Cycle Design Cold Critical Target k-eff Versus Cycle Exposure

[ ]

* [ ]

Table 3.4 Cycle 25 Reactivity Margin Summary

Table 3.4 Cycle 25 Reactivity Margin Summary *(Continued)*

Figure 3.1 Cycle 25 Reference Loading Pattern

Figure 3.1 Cycle 25 Reference Loading Pattern *(Continued)*

Figure 3.2 Cycle 25 Upper Left Quarter Core Layout by Fuel Type

Figure 3.2 Cycle 25 Upper Right Quarter Core Layout by Fuel Type
(Continued)

Figure 3.2 Cycle 25 Lower Left Quarter Core Layout by Fuel Type
(Continued)

Figure 3.2   Cycle 25 Lower Right Quarter Core Layout by Fuel Type
(Continued)

4.0 REFERENCES

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

2. FS1-0034669, Revision 3.0, Brunswick Unit 1 LAR Equilibrium Cycle Specific ATRIUM 11 Fuel Assembly Mechanical Data for Core Engineering, April 2018.

3. FS1-0040324, Revision 1.0, Confirmation of Proposed ATRIUM 11 Equilibrium Cycle Input Parameters, October 2018.

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

5. ANP-3724P, Revision 0, Susquehanna ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report, Framatome Inc., October 2018.

Appendix A Susquehanna Representative Equilibrium Cycle 25 Step-through Depletion Summary, Control Rod Patterns, Core Average Axial Power and Exposure Distributions

Table A.1 Cycle 25 Design Depletion Summary

Table A.2 Cycle 25 Design Depletion Thermal Margin Summary

[

]

[ ]

[

]

[ ]

Appendix B Elevation Views of the Susquehanna Equilibrium Cycle Design Fuel Assemblies

Appendix C Susquehanna Representative Equilibrium Cycle 25 Radial Exposure and Power Distributions

Figure C.1 Cycle 25 BOC Exposure Distribution (GWd/MTU)

Figure C.1 Cycle 25 BOC Exposure Distribution (GWd/MTU) *(Continued)*

Figure C.2 Susquehanna Unit 2 Cycle 25 EOC Exposure Distribution (18.8 GWd/MTU)

Figure C.2 Cycle 25 EOC Exposure Distribution (18.8 GWd/MTU) *(Continued)*

Figure C.3 Cycle 25 Radial Power Distribution at 0.0 MWd/MTU

Figure C.3 Cycle 25 Radial Power Distribution at 0.0 MWd/MTU *(Continued)*

Figure C.4 Cycle 25 Radial Power Distribution at 17,752.9 MWd/MTU (EOFP)

Figure C.4 25 Radial Power Distribution at 17,752.9 MWd/MTU (EOFP) *(Continued)*

Enclosure 12c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3727P, Susquehanna ATRIUM 11
Equilibrium Cycle Fuel Cycle Design Report

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3727P, Revision 0, "Susquehanna ATRIUM 11 Equilibrium Cycle Fuel Cycle Design Report," dated October 2018 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 24th

day of October, 2018.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY
PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

Enclosure 13b of PLA-7783

Framatome Topical Report
ANP-3724NP

Susquehanna ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report

(Non-Proprietary Version)

framatome

Susquehanna ATRIUM 11 Equilibrium Fuel

ANP-3724NP
Revision 0

Nuclear Fuel Design Report

October 2018

Framatome Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
BOL	beginning of life
BWR	boiling water reactor
EVC	plenum region in a fuel pin modeled as an evacuated section
kg/MTU	kilograms per metric ton of initial uranium
LHGR	linear heat generation rate
LPF	local peaking factor
MCPR	minimum critical power ratio
MWd/MTU	megawatt days per metric ton of initial uranium
NRC	(United States) Nuclear Regulatory Commission

1.0 INTRODUCTION

This report provides results from the neutronic design analyses performed by Framatome Inc. for the Susquehanna ATRIUM 11 equilibrium fuel design. The methodology, design criteria, and general assumptions used in the fuel design are also provided.

Applicable neutronic design criteria are provided in the approved topical report ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 2). Neutronic design analysis methodology used to determine conformance to design criteria has been reviewed and approved by the NRC in the topical report EMF-2158(P)(A) (Reference 3).

The fuel design general assumptions include **[**

] . The neutronic component of this fuel design includes axially-varying enrichment and gadolinia with natural UO_2 blankets at the top and bottom of the assembly. Mechanical design parameters for the fuel design are from Reference 1 and are shown in Table 2.1. Other pertinent fuel and reactor core design information is given in Section 2.0 and in Appendices A through D.

2.0 NEUTRONIC DESIGN

The results of the neutronic design analyses are presented in this section. The fuel was designed to meet applicable design criteria, as well as reactivity and control requirements. Applicable neutronic design criteria outlined in Reference 2 are summarized below:

- **Power Distribution.** The local power distribution in the fuel assembly combined with the core power distribution shall result in Linear Heat Generation Rate (LHGR) and Minimum Critical Power Ratio (MCPR) values that are within the limits established for each fuel design.
- **Kinetics Parameters.** The moderator void reactivity coefficient due to boiling in the active channels and the Doppler fuel temperature reactivity coefficient shall be negative. The negative void and Doppler reactivity coefficients ensure a negative power coefficient during reactor operation. (Calculation results show that the assembly average Doppler and void reactivity coefficients remain negative for the life of the assembly. These results demonstrate that the Reference 2 Section 5 kinetics criteria are met on a bundle average basis.)
- **Control Blade Reactivity.** The design of the fuel assembly and the reactor core loading shall be such that the technical specification shutdown margin requirement is met for all reactor conditions.

2.1 *Neutronic Design Description*

The neutronic design parameters for these ATRIUM 11 assemblies are presented in Table 2.1. The key nuclear design characteristics are summarized below:

- The fuel assembly contains **[ ]** .
- Each fuel assembly has top and bottom natural uranium blankets.
- The enrichments are designed to yield a local power distribution which results in a balanced design relative to MCPR, LHGR, and other reactor operating requirements, e.g., power peaking.
- Gadolinia (Gd_2O_3 blended with UO_2) rods are designed to control assembly reactivity in order to meet reactivity control requirements in the reactor, e.g. cold shutdown margin.
- Fuel assembly designs utilize axially varying enrichment and/or gadolinia. The axial distributions of the lattices in the assemblies are shown in Figures 2.1, 2.2, and 2.3. The fuel rod distribution and axial descriptions are presented in Figures 2.4 through 2.8. The enrichment and gadolinia distribution maps for each of the assembly lattices are displayed in Appendix D.
- The fuel assembly incorporates an advanced fuel channel which improves uranium utilization.

2.2 *Lattice Control Blade Worths and Kinetics Parameters*

Beginning of life (BOL) lattice reactivities (k_{∞}) have been calculated for moderator and fuel conditions ranging from cold to hot operating conditions. From these reactivities, BOL control blade worths and kinetics parameters have been determined based on Original Equipment Blades (OEB), and Duralife-160C (D16) control blades (Reference 4).

Kinetics parameters are calculated for fuel temperature (Doppler), moderator void, and moderator temperature. **[**

] The results of these calculations are presented in Table 2.2 through Table 2.58.

2.3 *Enriched Lattice Uncontrolled Reactivities and Isotopic Data*

The enriched lattice exposure-dependent uncontrolled reactivities **[** **]** are presented graphically in Appendix A, and in tabular format in Appendix B. The enriched lattice exposure-dependent isotopic data **[** **]** are presented in Appendix C.

2.4 *Criticality Compliance*

The spent fuel storage and new fuel vault criticality compliance is not addressed in this report because the fuel design herein is meant for demonstration of methods, but the criticality compliance will be explicitly addressed in the Susquehanna ATRIUM 11 transition.

Table 2.1 Neutronic Design Parameters

Table 2.1 Neutronic Design Parameters *(Continued)*

Table 2.1 Neutronic Design Parameters *(Continued)*

Parameter	Design Value
Control Blade Data for OEB	
Total span, inch	General Electric Proprietary
Total support span, inch	“
Total thickness, inch	“
Total face-to-face internal dimension, inch	“
B_4C rod absorber (wing absorber zone 1) Number of rods Diameter of rod, inch Diameter of sheath, inch Theoretical density B_4C, % B_4C zone span, inch	“
Blade stiffener (wing absorber zone 2) Width, inch Total thickness, inch Distance from center support, inch Stiffener zone span, inch	“
B_4C rod absorber (wing absorber zone 3) Number of rods Diameter of rod, inch Diameter of sheath, inch Theoretical density B_4C, % B_4C zone span, inch	“
Control Blade Data for D16	
Total span, inch	“
Total support span, inch	“
Total thickness, inch	“
Total face-to-face internal dimension, inch	“
B_4C rod absorber (wing absorber zone 1) Number of rods Diameter of rod, inch Diameter of sheath, inch Theoretical density B_4C, % B_4C zone span, inch	“
Hafnium rod absorber (wing absorber zone 2) Number of rods Diameter of rod, inch Diameter of sheath, inch Hafnium rod zone span, inch	“

Table 2.1 Neutronic Design Parameters *(Continued)*

Parameter	Design Value
Core Data*	
Number of fuel assemblies in the core	764
Rated thermal power level, MWt	3,952
Rated core flow, Mlbm/hr	100.0
Inlet subcooling, Btu/lbm	26.4
Dome pressure, psia	1,050
[ ]	[ ]
[ ]	[ ]
[ ]	[ ]

* Some values are representative of rated conditions and may vary depending on the core statepoint.

Table 2.2 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.3 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.4 Kinetics Parameters at BOL

[ ]

Table 2.5 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.6 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.7 Kinetics Parameters at BOL

[ ]

Table 2.8 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.9 Control Blade Worths at BOL for Control Blade Type D16

Table 2.10 Kinetics Parameters at BOL

Table 2.11 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.12 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.13 Kinetics Parameters at BOL

[ ]

Table 2.14 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.15 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.16 Kinetics Parameters at BOL

[ ]

Table 2.17 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.18 Control Blade Worths at BOL for Control Blade Type D16

Table 2.19 Kinetics Parameters at BOL

Table 2.20 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.21 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.22 Kinetics Parameters at BOL

[ ]

Table 2.23 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.24 Control Blade Worths at BOL for Control Blade Type D16

Table 2.25 Kinetics Parameters at BOL

Table 2.26 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.27 Control Blade Worths at BOL for Control Blade Type D16

Table 2.28 Kinetics Parameters at BOL

Table 2.29 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.30 Control Blade Worths at BOL for Control Blade Type D16

Table 2.31 Kinetics Parameters at BOL

Table 2.32 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.33 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.34 Kinetics Parameters at BOL

[ ]

Table 2.35 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.36 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.37 Kinetics Parameters at BOL

[ ]

Table 2.38 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.39 Control Blade Worths at BOL for Control Blade Type D16

Table 2.40 Kinetics Parameters at BOL

Table 2.41 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.42 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.43 Kinetics Parameters at BOL

[ ]

Table 2.44 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.45 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.46 Kinetics Parameters at BOL

[ ]

Table 2.47 Control Blade Worths at BOL for Control Blade Type OEB

Table 2.48 Control Blade Worths at BOL for Control Blade Type D16

Table 2.49 Kinetics Parameters at BOL

Table 2.50 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.51 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.52 Kinetics Parameters at BOL

[ ]

Table 2.53 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.54 Control Blade Worths at BOL for Control Blade Type D16

Table 2.55 Kinetics Parameters at BOL

Table 2.56 Control Blade Worths at BOL for Control Blade Type OEB

[ ]

Table 2.57 Control Blade Worths at BOL for Control Blade Type D16

[ ]

Table 2.58 Kinetics Parameters at BOL

[ ]

Figure 2.1 Assembly Map

Figure 2.2 Assembly Map

Figure 2.3 Assembly Map

Figure 2.4 Fuel Rod Distribution

[ ]

Figure 2.5 Fuel Rod Distribution

Figure 2.6 Fuel Rod Distribution

[ ]

Figure 2.7 Fuel Rod Axial Description

Figure 2.8 Fuel Rod Axial Description

3.0 REFERENCES

1. FS1-0034669, Revision 3.0, Brunswick Unit 1 LAR Equilibrium Cycle Specific ATRIUM 11 Fuel Assembly Mechanical Data for Core Engineering, April 2018.

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

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

4. FS1-0039032, Revision 1.0, Susquehanna ATRIUM 11 Equilibrium Cycle Cross-section Library Generation and FUELRQ Uranium Requirements, August 2018.

Appendix A Enriched Lattice Hot Uncontrolled Reactivity and LPF Plots

[ ]

Appendix B Enriched Lattice Hot Uncontrolled Reactivity and LPF Tables

[ ]

[

]

[ ]

[ ]

Appendix C Enriched Lattice Isotopic Data Tables

[ ]

[ ]

[ ]

[ ]

[

]

Appendix D Lattice Enrichment Distribution Maps

[ ]

[ ]

[ ]

[

]

Enclosure 13c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3724P, Susquehanna ATRIUM 11
Equilibrium Fuel Nuclear Fuel Design Report

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Morris Byram. I am Manager, Product Licensing, for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3724P, Revision 0, entitled "Susquehanna ATRIUM 11 Equilibrium Fuel Nuclear Fuel Design Report" referred to herein as "Document." Information contained in this document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this document as proprietary and confidential.

5. This document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in this document is considered proprietary for the reasons set forth in paragraphs 6(c) and 6(d) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this document has been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 19th

day of October, 2018.

HAILEY M SIEKAWITCH
Commission Expires
NOTARY
PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

Enclosure 14b of PLA-7783

Framatome Topical Report
ANP-3783NP

Susquehanna ATRIUM 11 Transient Demonstration

(Non-Proprietary Version)

Susquehanna ATRIUM 11 Transient Demonstration

ANP-3783NP
Revision 0

June 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

Tables

Figures

Nomenclature

AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
ATWS-RPT	anticipated transient without scram recirculation pump trip
BOC	beginning of cycle
BT	boiling transition
BWR	boiling water reactors
CPR	critical power ratio
EFPH	effective full power hours
EM	evaluation model
EOFP	end of full power
FoM	figure of merit
FSAR	final safety analysis report
FWCF	feedwater controller failure
HPCI	high-pressure coolant injection
IHPCIS	inadvertent startup of the HPCI system
LAR	license amendment request
LHGR	linear heat generation rate
$LHGRFAC_p$	power-dependent LHGR multiplier
LPRM	local power range monitor
LRNB	generator load rejection no bypass
LTR	licensing topical report
MCPR	minimum critical power ratio
MDNBR	minimum departure from nucleate boiling ratio
MELLLA	maximum extended load line limit analysis
MSIV	main steam isolation valve
NSS	nominal scram speed
OLMCPR	operating limit minimum critical power ratio

Nomenclature
(continued)

Pbypass	power below which direct scram on TSV/TCV closure is bypassed
PRFO	pressure regulator failure open
SLMCPR	safety limit minimum critical power ratio
SLO	single-loop operation
SRV	safety/relief valve
TCV	turbine control valve
TIP	traversing in-core probe
TLO	two-loop operation
TSSS	technical specification scram speed
TSV	turbine stop valve
TTNB	turbine trip no bypass
ΔMCPR	change in minimum critical power ratio

1.0 Introduction

This report summarizes the results of a subset of transient analyses performed to support the Susquehanna Framatome* Advanced Methods license amendment request (LAR) to include the Reference 1, 2, 3, and 4 Licensing Topical Reports (LTR) into the Susquehanna Steam Electric Plant Technical Specifications.

For a typical reload, a full assessment of the power/flow map, cycle exposure, and scram speed are done on a cycle specific basis for the actual core configuration to develop thermal limits. The intention of this report is to demonstrate the applicability of the AURORA-B AOO methodology (Reference 1) to Susquehanna for the transient analyses that are typically limiting on a cycle-specific basis. Therefore, this document is a subset of transient analyses typically performed for each cycle.

The analyses presented in Section 4.0 of this document are based upon a representative equilibrium cycle of ATRIUM 11 fuel, Reference 5. A variety of power/flow state points are performed at a cycle exposure and scram speed discussed in each subsection of Section 4.0.

The AURORA-B AOO analysis is used to calculate the change in the minimum critical power ratio (ΔMCPR) during the anticipated operational occurrence (AOO). The ΔMCPR is combined with the safety limit MCPR (Reference 3) to establish or confirm the plant operating limits for MCPR.

Power-dependent linear heat generation rate (LHGR) multipliers ($LHGRFAC_p$), applied directly to the LHGR limits to protect against fuel melting and overstraining of the cladding during an AOO, are determined using the RODEX4 thermal-mechanical methodology (Reference 4). For the AURORA-B AOO methodology, the applicable figure of merit for the $LHGRFAC_p$ calculation is the time-dependent nodal power.

* Framatome Inc. formerly known as AREVA Inc.

The AURORA-B AOO analysis is also used to calculate the maximum reactor vessel pressure and the maximum dome pressure during the ASME and ATWS events. The calculated maximum reactor vessel pressure is compared to the ASME acceptance criterion (110% of vessel design pressure) and the calculated maximum steam dome pressure is compared to the pressure safety limit in the plant Technical Specifications. For the ATWS event, the calculated maximum reactor vessel pressure is compared to ASME Service Level C (120% of design pressure) to demonstrate that the event acceptance criterion is met. Meeting the acceptance criteria confirms that the plant safety valve performance (number of valves available, capacity per valve, and setpoints) is acceptable.

The ACE/ATRIUM 11 critical power correlation (Reference 2) is used to evaluate the thermal margin of the ATRIUM 11 fuel.

2.0 MCPR Fuel Cladding Integrity Safety Limit

2.1 *Methodology*

The two-loop operation (TLO) and single-loop operation (SLO) safety limit minimum critical power ratios (SLMCPR) were determined using the methodology presented in Reference 3. The SLMCPR is defined as the minimum value of the critical power ratio which ensures at least 99.9% of the fuel rods in the core avoid boiling transition (BT) during normal operation or an anticipated operational occurrence (AOO). The SLMCPR is determined using a statistical analysis employing a Monte Carlo process that perturbs the input parameters used in the calculation of minimum critical power ratio (MCPR). The set of uncertainties used in the statistical analysis includes both fuel-related and plant-related uncertainties.

The SLMCPR analysis is performed with a power distribution that conservatively represents expected reactor operating states that could both exist at the operating limit MCPR (OLMCPR) and produce a MCPR equal to the SLMCPR during an AOO. **[**

]

In the Framatome methodology, the effects of channel bow on the critical power performance are accounted for in the SLMCPR analysis. [

] This adjustment is a plant specific extension of the Reference 3 approved methodology (Reference 6).

2.2 *Analysis*

The core loading and cycle depletion from the Reference 5 representative equilibrium cycle of ATRIUM 11 fuel were used as the basis of the SLMCPR analysis. Analyses were performed for the minimum and maximum core flow conditions associated with rated power for the Susquehanna power/flow map. The SLO calculations used a core flow of 52% of rated and a core power of 67.1% of rated.

The ACE/ATRIUM 11 critical power correlation (Reference 2) is used to evaluate the thermal margin of the ATRIUM 11 fuel.

The uncertainties used in the SLMCPR analysis are presented in Table 2.1. The radial and nodal power uncertainties used in the analysis include the combined effects of up to 42% of the traversing in-core probe (TIP) channels out-of-service, up to 50% of the local power range monitors (LPRM) out-of-service, and an LPRM calibration interval of up to 2,500 effective full power hours (EFPH). For the representative equilibrium cycle of ATRIUM 11 fuel (Reference 5), [ ]

[

]

The SLMCPR analysis supports a TLO SLMCPR of 1.07 and an SLO SLMCPR of 1.09. Table 2.2 and Table 2.3 present a summary of the analysis results including the SLMCPR and the percentage of rods expected to experience BT.

Table 2.1 Safety Limit Uncertainties

Parameter	Standard Deviation	Reference
Reactor System Related Uncertainties		
Feedwater flow rate	1.76%	7
Feedwater temperature	0.76%	7
Core Pressure	0.50%	7
Total core flow rate		7
Two-loop	2.5%	
Single-loop	6.0%	
Fuel Related Uncertainties		
[		
		]

* [
]

† [
]

Table 2.2 TLO Safety Limit Results

% Rated Power	% Rated Flow	SLMCPR	Number of Rods in BT	% of Rods in BT
100	108	1.07	55	0.0643
100	99	1.07	57	0.0666

Table 2.3 SLO Safety Limit Results

% Rated Power	% Rated Flow	SLMCPR	Number of Rods in BT	% of Rods in BT
67.1	52	1.09	56	0.0654

3.0 Anticipated Operational Occurrences

3.1 *AURORA-B AOO Evaluation Model*

AURORA-B is a comprehensive evaluation model developed for predicting the dynamic response of boiling water reactors (BWRs) during transient, postulated accident, and beyond design-basis accident scenarios. The evaluation model (EM) contains a multi-physics code system with flexibility to incorporate all the necessary elements for analysis of the full spectrum of BWR events that are postulated to affect the nuclear steam supply system of the BWR plant. Deterministic analysis principles are applied to satisfy plant operational and Technical Specification requirements through the use of conservative initial conditions and boundary conditions.

The foundation of AURORA-B AOO is built upon three computer codes, S-RELAP5, MB2-K, and RODEX4. Working together as a system, they make up the multi-physics evaluation model that provides the necessary systems, components, geometries, processes, etc. to assure adequate predictions of the relevant BWR event characteristics for its intended applications. The three codes making up the foundation of the code system are;

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

- MB2-K – This code uses advanced nodal expansion methods to solve the three-dimensional, two-group, neutron kinetics equations. The MB2-K code is consistent with the MICROBURN-B2 steady state core simulator. MB2-K receives a significant portion of its input from the steady state core simulator.

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

3.2 *Description of [ ] Analysis Process*

The AURORA-B AOO methodology (Reference 1) includes an evaluation of the impact of code uncertainties on Figures of Merit (FoM) (e.g. ΔMCPR, time dependent nodal power, peak pressure) **[**

] that has wide acceptance in the nuclear industry.

[

]

Table 3.1 [

]

[

]

3.2.1 Sampled Parameters [ ]

The set of code and modeling uncertainty parameters to be sampled for AOO calculations is shown in Table 3.2.

[

]

[

]

3.2.2 Sampling Ranges

The sampling ranges shown in Table 3.2 are applicable to Susquehanna Units 1 and 2. Per the approved methodology (Reference 1), the sampling ranges address uncertainties inherent in the S-RELAP5 models **[**

]

[

]

A description of the basis for the sampling ranges used for each of the above sampled variables is found in Reference 1 Safety Evaluation, Sections 3.6.4.1 – 3.6.4.17.

Table 3.2 Sampling Ranges for Uncertainty Parameters

3.3 *Application [ ] for Demonstration Cases*

The statistical analysis process presented in the previous sections will be used to determine the **[ ]** values for FoMs associated with the nominal transient simulations performed to demonstrate the methodology application to the equilibrium ATRIUM 11 core. Section 3.6.5 of the Safety Evaluation (Reference 1) allows for subsequent analyses to utilize the **[ ]** to determine base conservative measures to be applied for calculation of the key FoM in future reload licensing. **[**

]

4.0 Analysis of Plant Transients

Framatome's licensing methodology is based upon core conditions established by a detailed step-through calculation. In support of demonstrating the AURORA-B AOO method to the Susquehanna units, plant transients are analyzed for a small subset of power and flow conditions at a cycle exposure and scram speed discussed in each subsection. The transient analyses, presented in this section, are performed using plant parameters provided by the utility for a full core of ATRIUM 11 fuel.

The transient events chosen to demonstrate the application of the AURORA-B AOO method are generally limiting events for Susquehanna as determined from previous cycle analyses and a review of Chapter 15 of the final safety analysis report (FSAR).

4.1 *Transient Events*

4.1.1 Load Rejection Without Bypass / Turbine Trip Without Bypass

The generator load rejection without bypass (LRNB) and the turbine trip without bypass (TTNB) events were combined as one event. The combined LRNB/TTNB event causes closure of the turbine stop valves and fast closure of the turbine control valves. The resulting compression wave travels through the steam lines into the vessel and creates a rapid pressurization. The increase in pressure causes a decrease in core voids, which in turn causes a rapid increase in power. Closure of the turbine stop valves and fast closure of the turbine control valves causes a reactor scram and a recirculation pump trip which helps mitigate the pressurization effects. Turbine bypass system operation, which also mitigates the consequences of the event, is not credited. The excursion of the core power due to the void collapse is terminated primarily by the reactor scram and revoiding of the core.

To demonstrate the AURORA-B AOO transient methodology models the combined LRNB/TTNB event appropriately, analyses were performed for the following range of conditions within the approved MELLLA power/flow map:

- 100% core power, with 108% and 99% core flow
- 80% core power, with 108% core flow
- 40% core power, with 108% core flow
- 26% core power, with 108% core flow (direct scram)
- 26% core power, with 108% core flow (non-direct scram)

Table 4.1 presents the change in MCPR and $LHGRFAC_p$ for the combined LRNB/TTNB event. The transient analyses are performed at the end of full power (EOFP) cycle exposure, utilizing the NSS scram speeds. Table 4.2 presents the sequence of event timing for the combined LRNB/TTNB event at 100% power with 108% core flow. Figure 4.1 - Figure 4.3 show the responses of various reactor and plant parameters during the limiting combined LRNB/TTNB event initiated at 100% of rated power and 108% of rated core flow with NSS insertion times.

4.1.2 Feedwater Controller Failure (FWCF)

The increase in feedwater flow due to a failure of the feedwater control system to maximum demand results in an increase in the water level and a decrease in the coolant temperature at the core inlet. The increase in core inlet subcooling causes an increase in core power. As the feedwater flow continues at maximum demand, the water level continues to rise and eventually reaches the high water level trip setpoint. The initial water level is conservatively assumed to be at the low-level normal operating range to delay the high-level trip and maximize the core inlet subcooling that results from the FWCF. Reaching the high water level trip setpoint will trip the main turbine and the reactor feed pump turbines. The main turbine trip causes the turbine stop valves to close in order to prevent damage to the turbine from excessive liquid inventory in the steam line. The valve closure creates a compression wave that travels to the core causing a void collapse and subsequent rapid power excursion. The closure of the turbine stop valves also initiates a reactor scram and a recirculation pump trip. Four of the five installed turbine bypass valves are assumed operable and provide pressure relief. The core power excursion is mitigated in part by the pressure relief, but the

primary mechanism for termination of the event is reactor scram and revoiding of the core.

To demonstrate the AURORA-B AOO transient methodology models the FWCF event appropriately, analyses were performed for the following range of conditions within the approved MELLLA power/flow map:

- 100% core power, with 108% and 99% core flow
- 80% core power, with 108% core flow
- 40% core power, with 108% core flow
- 26% core power, with 108% core flow (direct scram)
- 26% core power, with 108% core flow (non-direct scram)

Table 4.1 presents the change in MCPR and $LHGRFAC_p$ for the FWCF event. The transient analyses are performed at the EOFP cycle exposure, utilizing the NSS scram speeds. Table 4.3 presents the sequence of event timing for the FWCF event at 100% power with 108% core flow. Figure 4.4 - Figure 4.6 show the responses of various reactor and plant parameters during the limiting FWCF event initiated at 100% of rated power and 108% of rated core flow with NSS insertion times.

4.1.3 Inadvertent Startup of the HPCI Pump

The inadvertent startup of the HPCI system (IHPCIS) results in the injection of cold water to the reactor vessel from the HPCI pump through the feedwater sparger. Injection of this subcooled water increases the subcooling at the inlet to the core and results in an increase in the core power. The feedwater control system will attempt to control the water level in the reactor by reducing the feedwater flow. As long as the mass of steam leaving the reactor through the steam lines is more than the mass of HPCI water being injected, the water level will be controlled and a new steady-state condition will be established. In this situation, the event is similar to a loss of feedwater heating event. At low power, the HPCI flow can become more than the steam flow, and

the water level can increase until the high water level setpoint is reached. In this situation, the event is similar to an FWCF.

The HPCI flow in the Susquehanna units is only injected into one of the two feedwater lines and thus through the feedwater sparger on only one side of the reactor vessel, resulting in an asymmetric flow distribution of the injected HPCI flow. This asymmetric injection of the HPCI flow may cause an asymmetric core inlet enthalpy distribution and a larger enthalpy decrease for part of the core. **[**

]

To demonstrate the AURORA-B AOO transient methodology models the inadvertent startup of the HPCI event appropriately, analyses were performed for the following range of conditions within the approved MELLLA power/flow map:

- 100% core power, with 108% and 99% core flow
- 80% core power, with 108% core flow
- 40% core power, with 108% core flow
- 26% core power, with 108% core flow

Table 4.1 presents the change in MCPR and $LHGRFAC_p$ for the inadvertent startup of the HPCI pump event. The transient analyses are performed at the EOFP cycle exposure. Table 4.4 presents the sequence of event timing for the IHPCIS event at 100% power with 108% core flow. Figure 4.7 - Figure 4.9 show the responses of various reactor and plant parameters during the limiting IHPCIS event initiated at 100% of rated power and 108% of rated core flow.

4.1.4 ASME Overpressurization Analysis

This section describes the maximum overpressurization analyses performed to demonstrate compliance with the ASME Boiler and Pressure Vessel Code. The

analysis shows that the safety valves at Susquehanna have sufficient capacity and performance to prevent the reactor vessel pressure from reaching the safety limit of 110% of the design pressure.

To demonstrate the applicability of the AURORA-B AOO (Reference 1) methodology for ASME overpressurization analyses, MSIV, TSV, and TCV closure analyses were performed for 102% power and 108% flow and 102% power and 99% flow at the latest exposure in the cycle design. The valve closure results in a rapid pressurization of the core. The increase in pressure causes a decrease in void which in turn causes a rapid increase in power. The following assumptions were made in the analysis:

- No credit for direct scram on MSIV or TSV valve position or TCV fast closure (scram is delayed until the second safety-grade signal for high neutron flux or high dome pressure).
- No credit for RPT on TSV position or TCV motion (RPT delay until high dome pressure signal).
- No credit for opening of the turbine bypass valves.
- No credit for the SRVs opening at the relief setpoints (open at safety setpoints).
- The 2 lowest setpoint SRVs were assumed inoperable.
- TSSS insertion times were used.
- The initial dome pressure was set at the maximum allowed by the Technical Specifications, 1064.7 psia (1050 psig).
- A fast MSIV closure time of 2 seconds was used for the MSIV closure case.

Results of the TSV closure overpressurization analysis are presented in Table 4.5. Table 4.6 presents the sequence of event timing for the ASME event at 102% power with 99% core flow. Figure 4.10 - Figure 4.13 show the response of various reactor plant parameters during the TSV closure event. The maximum pressure of 1319 psig occurs in the lower plenum. The maximum dome pressure for the same event is 1290 psig. The results demonstrate that the maximum vessel pressure limit of 1375 psig and dome pressure limit of 1325 psig are not exceeded.

4.1.5 ATWS Overpressurization Analysis

This section describes the analyses performed to demonstrate that the peak vessel pressure for the limiting ATWS event is less than the ASME Service Level C limit of 120% of the design pressure (1500 psig). To demonstrate the applicability of the AURORA-B AOO (Reference 1) methodology for ATWS overpressurization analyses, the ATWS event analyses were performed at 100% power at 108% and 99% flow at the beginning of cycle (BOC) exposure based on historically limiting analyses. The MSIV closure and pressure regulator failure open (PRFO) events were evaluated. Failure of the pressure regulator in the open position causes the turbine control and turbine bypass valves to open such that steam flow increases until the maximum combined steam flow limit is attained. The system pressure decreases until the low pressure setpoint is reached, resulting in the closure of the MSIVs. The resulting pressurization wave causes a decrease in core voids and an increase in core pressure thereby increasing the core power.

The following assumptions were made in the analyses:

- The analytical limit ATWS-RPT setpoint and function were assumed.
- The 2 lowest setpoint SRVs were assumed inoperable.
- All scram functions were disabled.
- The initial dome pressure was set to the nominal pressure (1050.4 psia).
- An MSIV closure time of 2.0 seconds is used for the MSIV closure event. An MSIV closure time of 5.0 seconds is used for the PRFO event.

Results of the ATWS overpressurization analyses are presented in Table 4.5. Table 4.7 presents the sequence of event timing for the ATWS MSIV closure event at 100% power with 99% core flow. Figure 4.14 - Figure 4.17 show the response of various reactor plant parameters during the ATWS MSIV closure event, the event which results in the maximum vessel pressure. The maximum lower plenum pressure is 1381 psig and the maximum dome pressure is 1361 psig. The results demonstrate that the ATWS maximum vessel pressure limit of 1500 psig is not exceeded.

Table 4.1 Base Case Transient Results

State Point Power / Flow (% of rated)	ATRIUM 11 ΔMCPR	ATRIUM 11 $LHGRFAC_p$
	Combined LRNB/TTNB	
100 / 108	[ ]	[ ]
100 / 99	[ ]	[ ]
80 / 108	[ ]	[ ]
40 / 108	[ ]	[ ]
26 / 108	[ ]	[ ]
26 / 108 below Pbypass	[ ]	[ ]
	Feedwater Controller Failure	
100 / 108	[ ]	[ ]
100 / 99	[ ]	[ ]
80 / 108	[ ]	[ ]
40 / 108	[ ]	[ ]
26 / 108	[ ]	[ ]
26 / 108 below Pbypass	[ ]	[ ]
	Inadvertent Startup of the HPCI Pump	
100 / 108	[ ]	[ ]
100 / 99	[ ]	[ ]
80 / 108	[ ]	[ ]
40 / 108	[ ]	[ ]
26 / 108	[ ]	[ ]

Table 4.2 Sequence of Events Timing for the Combined LRNB/TTNB Event

Event	Time (sec)
TCV Closure Event	0.005
TSV Closure Event	0.005
Reactor Scram	0.075
Recirculation Pump Trip	0.185
Peak Power	0.675
Peak Heat Flux	0.775
Time of MDNBR	0.830
Peak Vessel Pressure (1289.20 psia)	2.090
Peak Steam Line Pressure (1296.49 psia)	2.124
Peak Dome Pressure (1263.26 psia)	2.312

Table 4.3 Sequence of Events Timing for the FWCF Event

Event	Time (sec)
FWCF Event Initiator	0.000
Level 8 – High Water Level – Trip	14.315
Level 8 – TSV Closure Signal	18.370
Reactor Scram	18.445
Turbine Bypass Valves Open	18.475
Recirculation Pump Trip	18.555
Feedwater Pump Trip	18.970
Peak Power	19.035
Peak Heat Flux	19.140
Time of MDNBR	19.230
Peak Steam Line Pressure (1249.08 psia)	20.952
Peak Vessel Pressure (1275.80 psia)	20.972
Peak Dome Pressure (1250.98 psia)	20.982

Table 4.4 Sequence of Events Timing for the Inadvertent Startup of the HPCI Pump Event

Event	Time (sec)
IHPCIS Event Initiator	0.000
Peak Power	22.105
Peak Steam Line Pressure (1055.20 psia)	23.160
Peak Dome Pressure (1063.37 psia)	23.280
Peak Vessel Pressure (1107.94 psia)	23.285
Time of MDNBR	28.140

Table 4.5 ASME and ATWS Overpressurization Analysis Results

Event	Maximum Vessel Pressure Lower Plenum (psig)	Maximum Dome Pressure (psig)
ASME Overpressurization		
TSV closure (102P/108F)	1319	1288
TSV closure (102P/99F)	1318	1290
ATWS Overpressurization		
MSIV closure (100P/108)	1370	1349
MSIV closure (100P/99F)	1381	1361

Table 4.6 Sequence of Events Timing for the ASME Overpressurization Event

Event	Time (sec)
TSV Closure Event Initiator (0.1 sec full closure time)	0.000
High Neutron Flux Setpoint	0.445
Reactor Scram	0.565
Recirculation Pump Trip Setpoint – High Pressure	0.570
Peak Power	0.705
Recirculation Pump Trip – High Pressure	1.125
SRV Actuation	1.400
Peak Steam Line Pressure (1327.97 psia)	2.190
Peak Dome Pressure (1304.63 psia)	2.595
Peak Vessel Pressure (1332.35 psia)	2.595

Table 4.7 Sequence of Events Timing for the ATWS Overpressurization Event

Event	Time (sec)
MSIV Closure Event Initiator (2.0 sec full closure time)	0.000
Recirculation Pump Trip Setpoint – High Pressure	1.950
Recirculation Pump Trip – High Pressure	2.505
SRV Actuation	2.680
Peak Power	2.730
Peak Vessel Pressure (1394.74 psia)	7.125
Peak Dome Pressure (1374.73 psia)	7.300
Peak Steam Line Pressure (1371.00 psia)	7.420

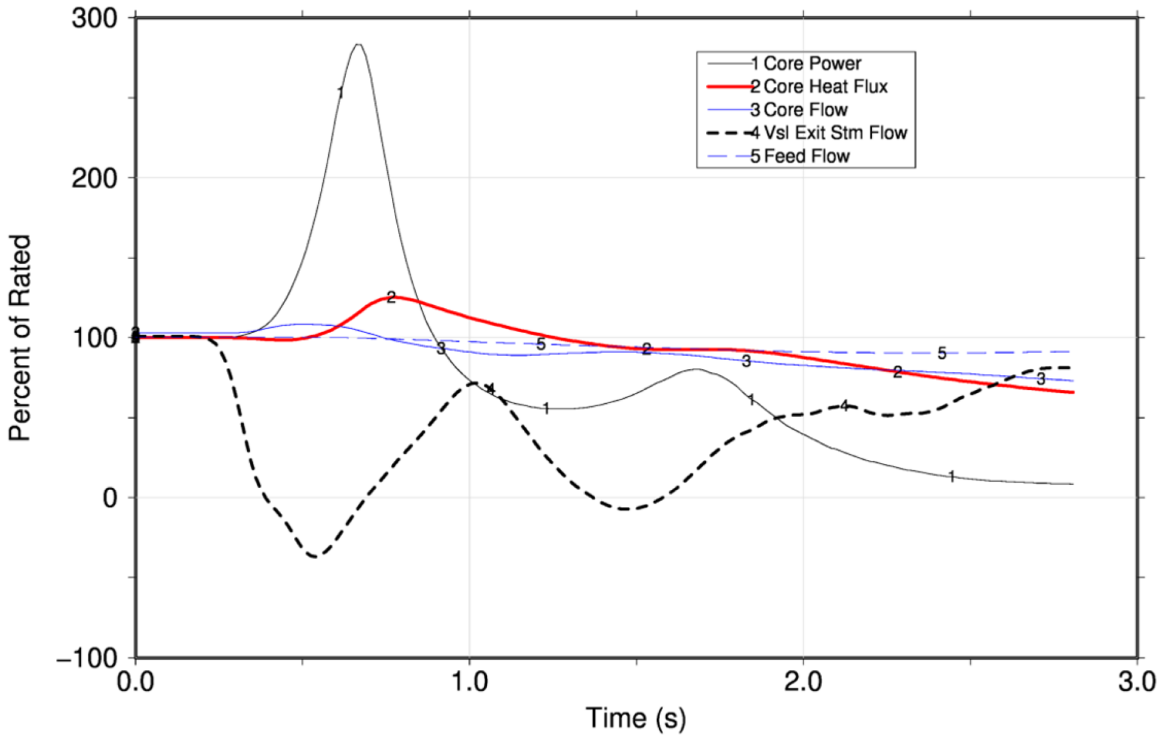

Figure 4.1 EOFP LRNB/TTNB at 100P/108F – NSS Key Parameters

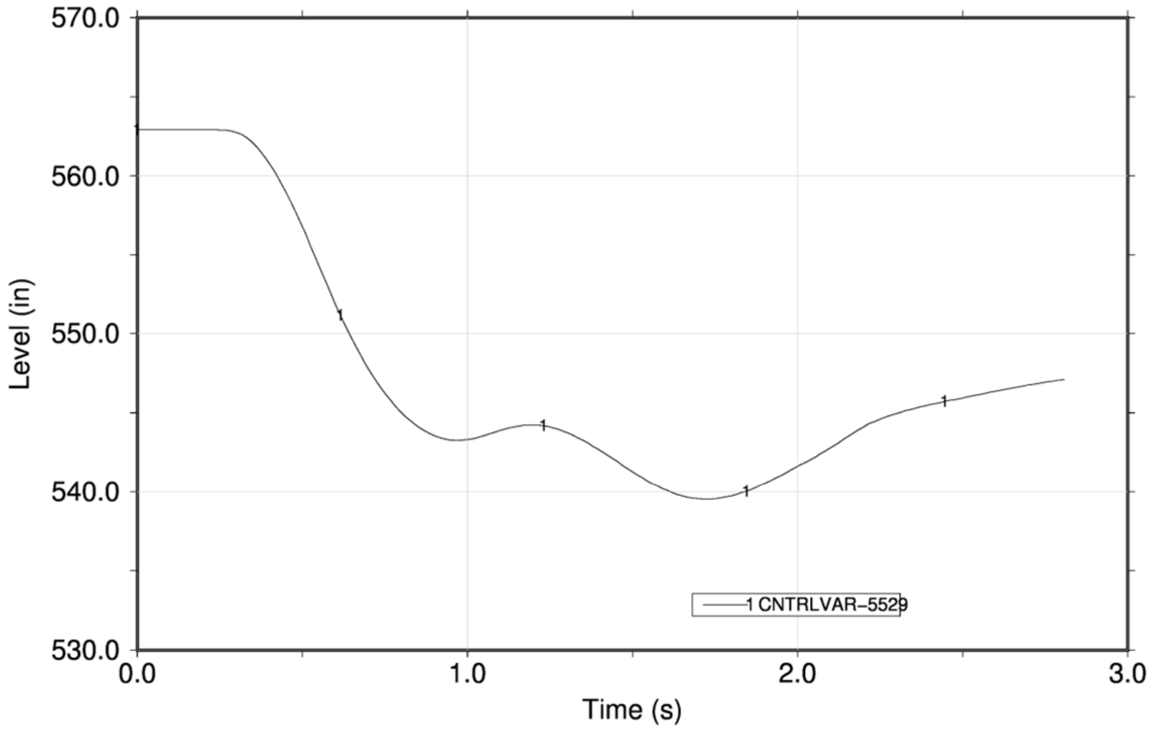

Figure 4.2 EOFP LRNB/TTNB at 100P/108F – NSS Water Level

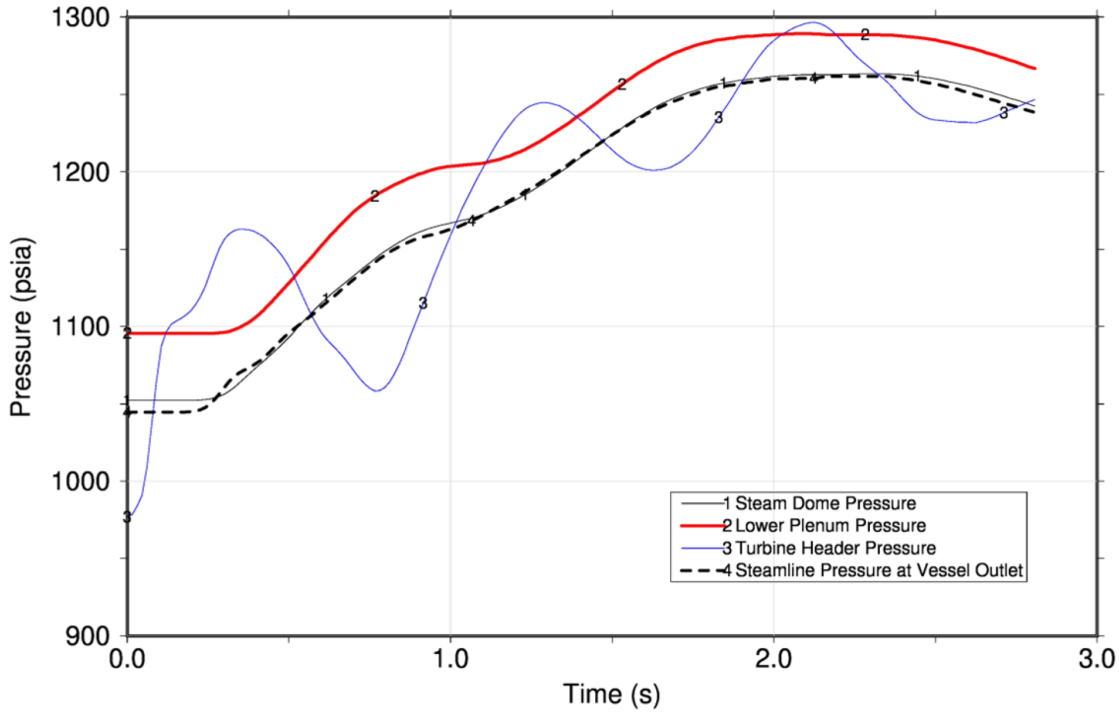

Figure 4.3 EOFP LRNB/TTNB at 100P/108F – NSS Pressures

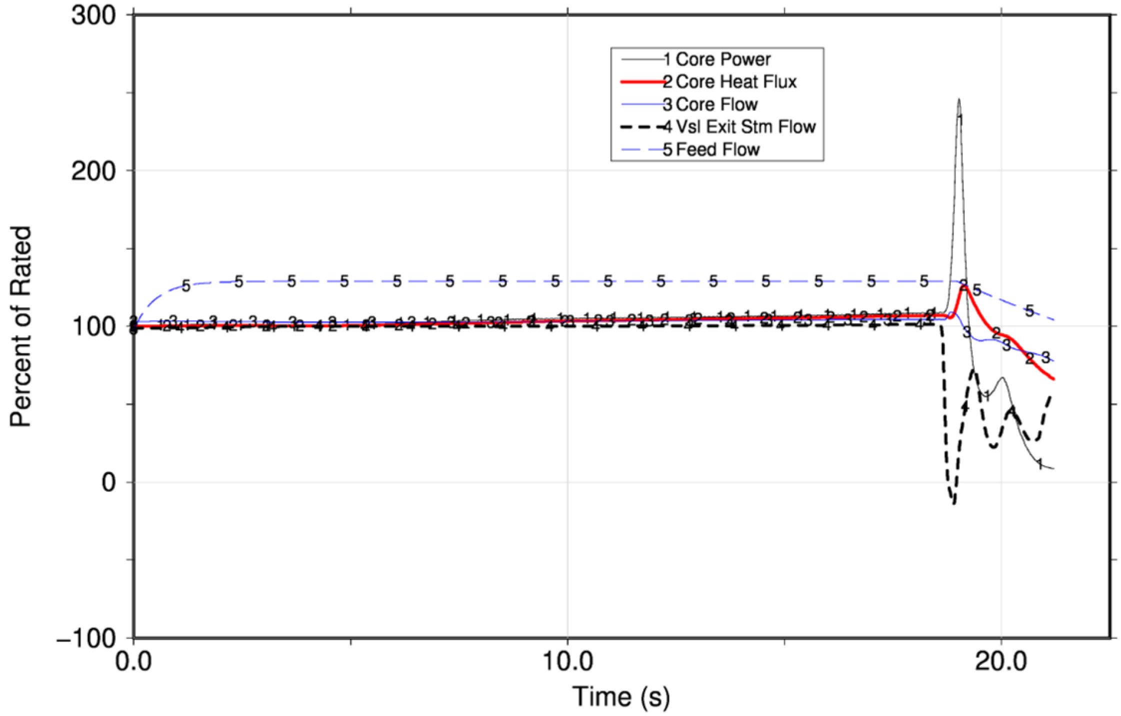

Figure 4.4 EOFP FWCF at 100P/108F – NSS Key Parameters

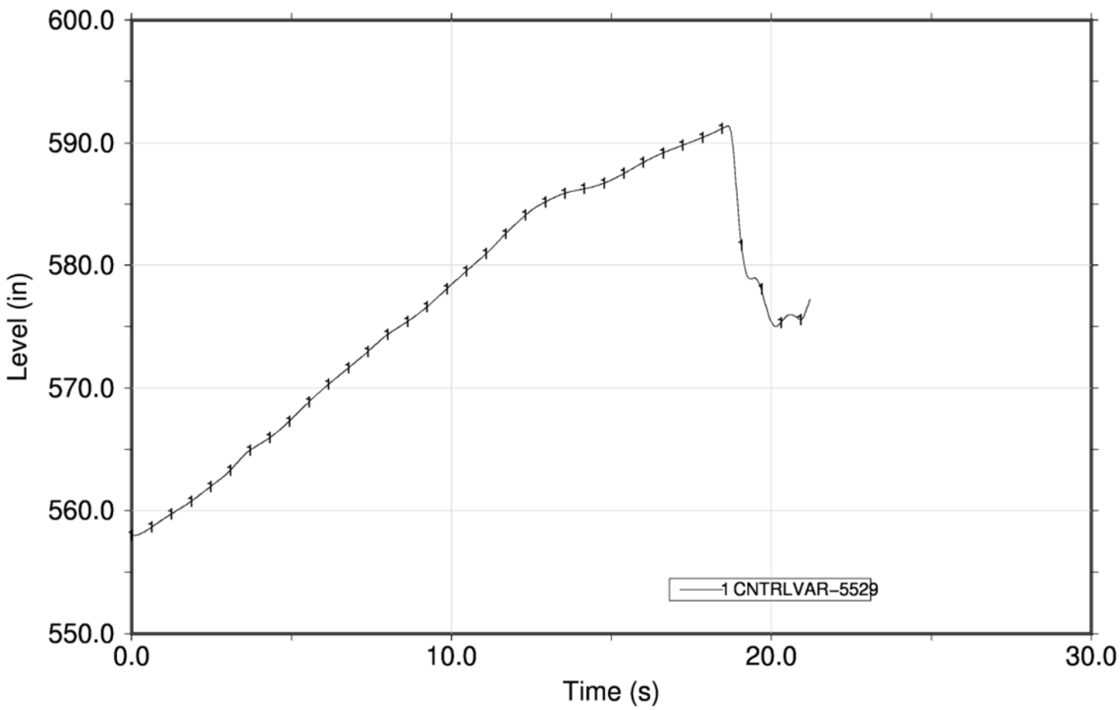

Figure 4.5 EOFP FWCF at 100P/108F – NSS Water Level

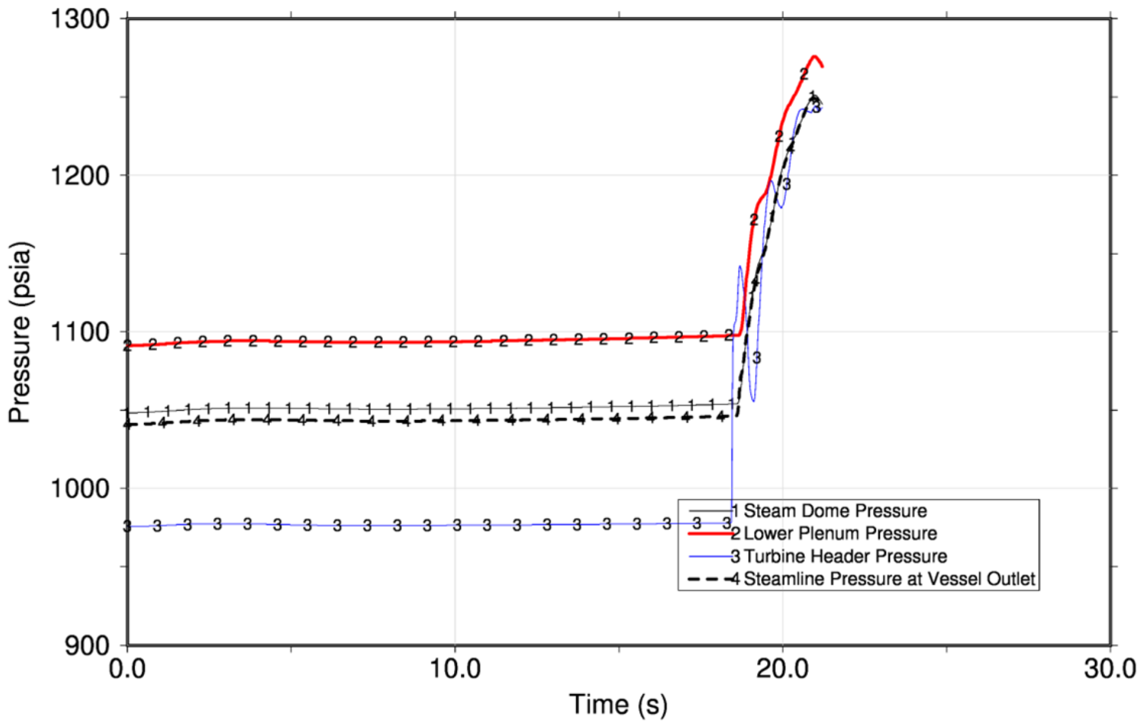

Figure 4.6 EOFP FWCF at 100P/108F – NSS Pressures

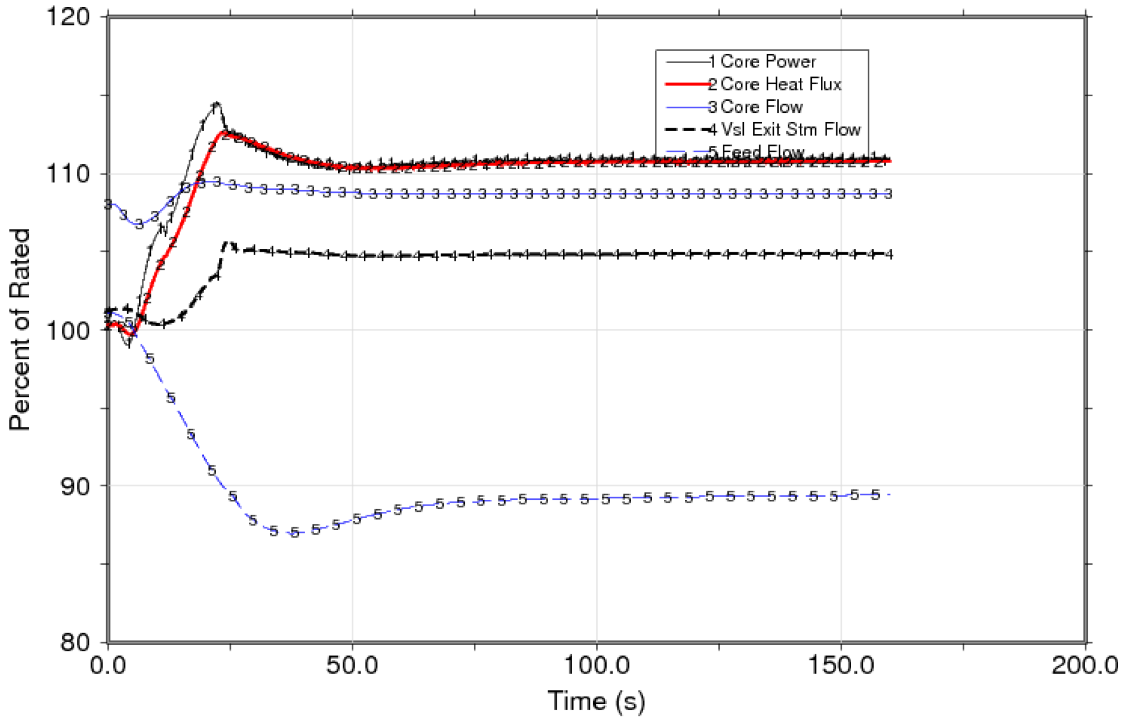

Figure 4.7 EOFP IHPCIS at 100P/108F – NSS Key Parameters

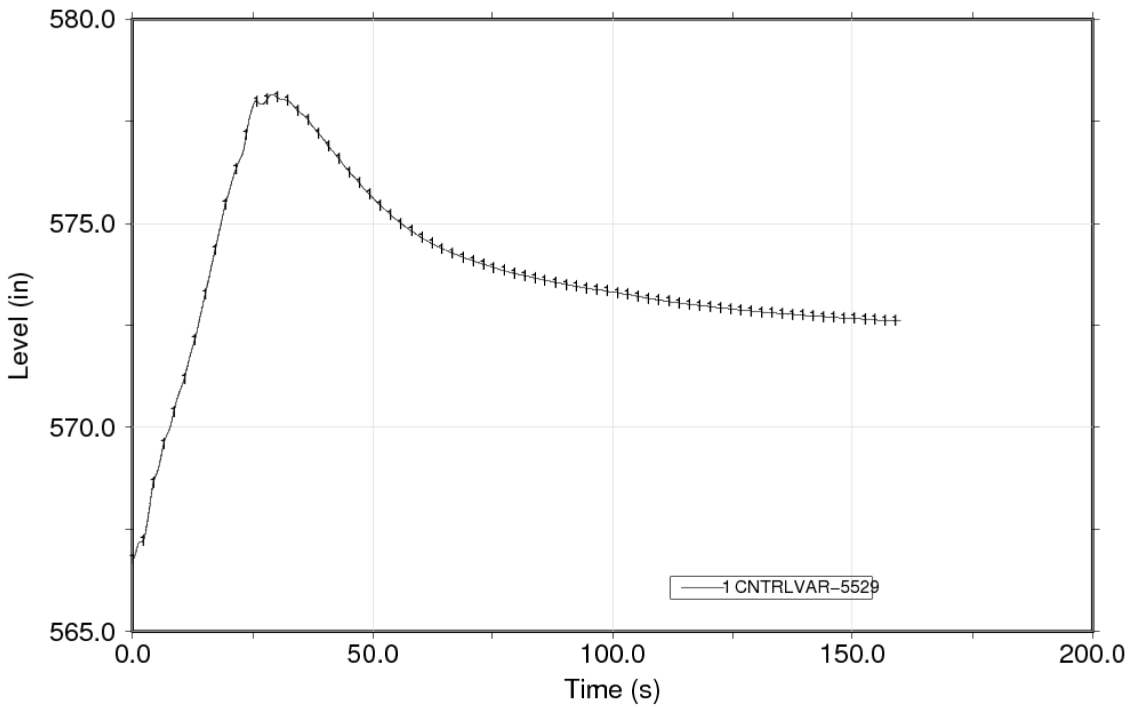

Figure 4.8 EOFP IHPCIS at 100P/108F – NSS Water Level

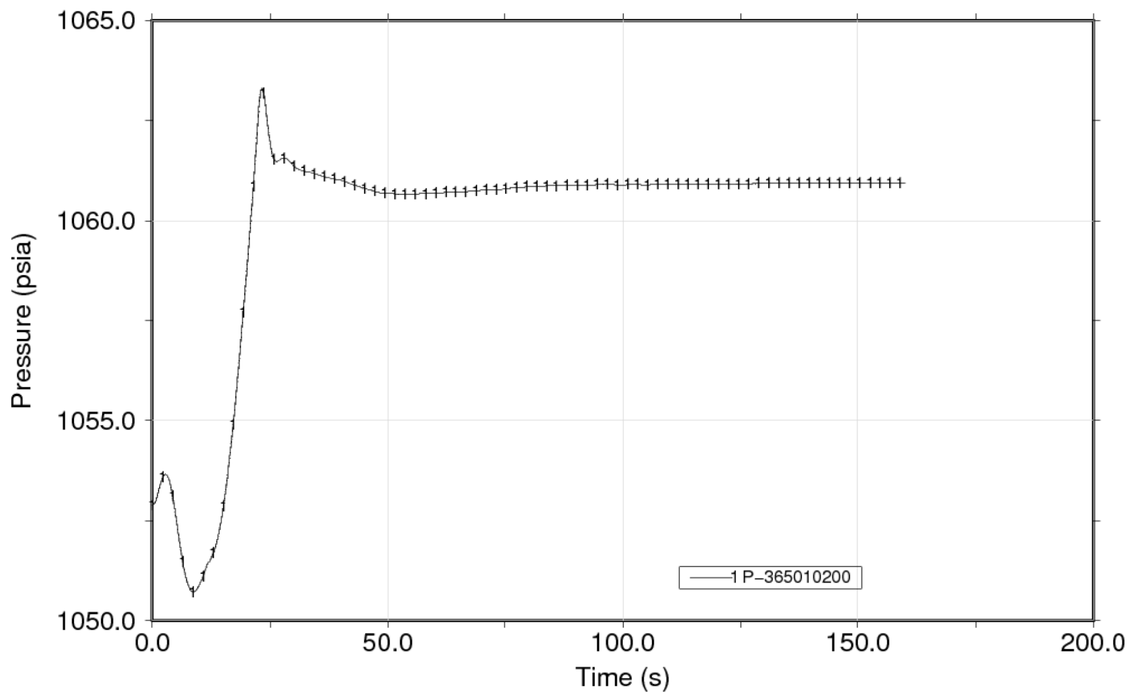

Figure 4.9 EOFP IHPCIS at 100P/108F – NSS Dome Pressure

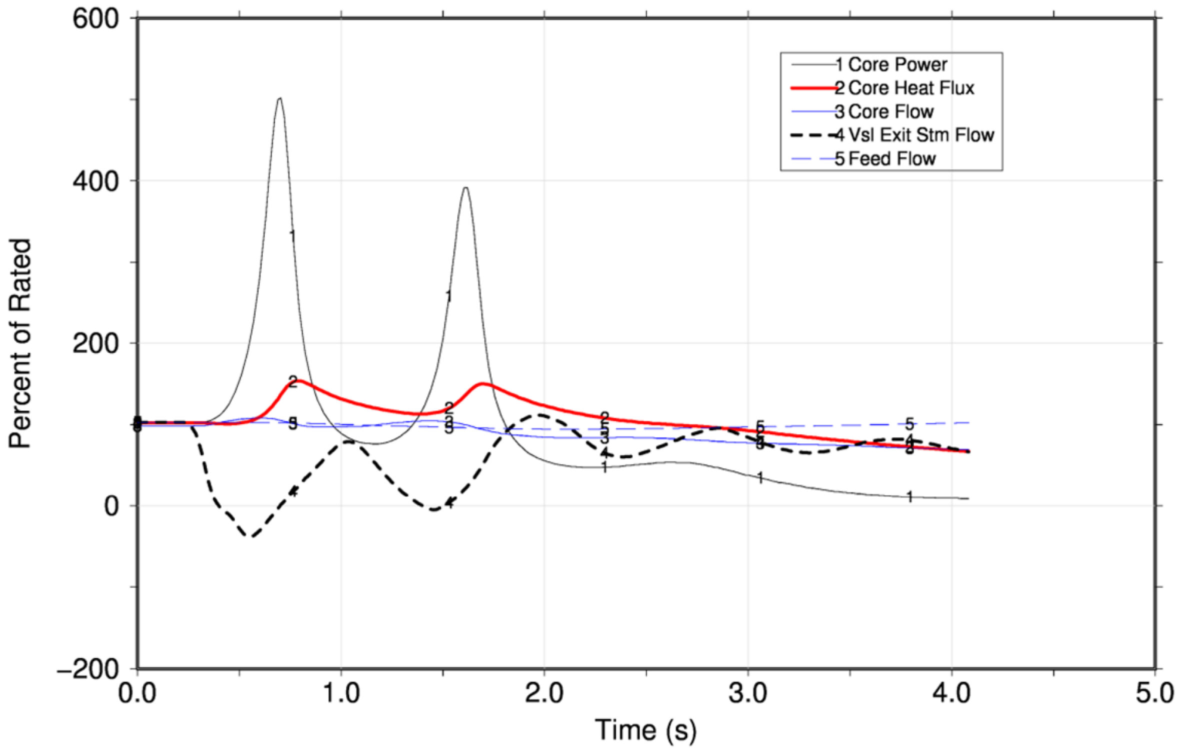

Figure 4.10 TSV Overpressurization Event at 102P/99F – Key Parameters

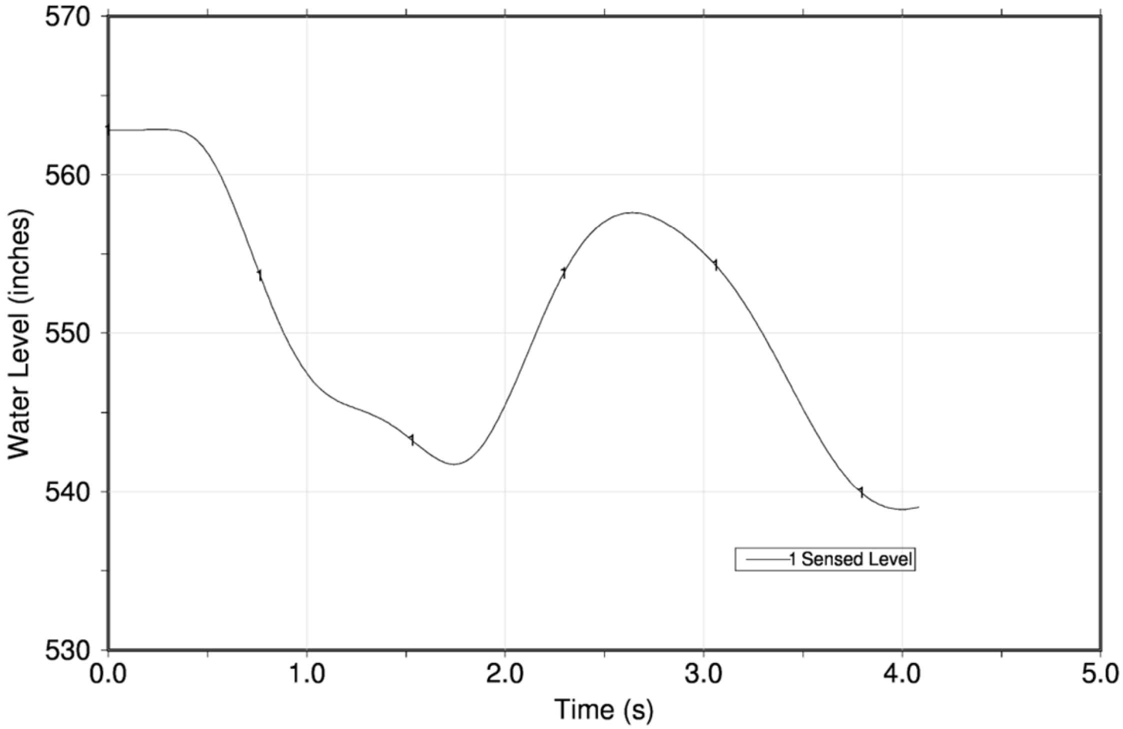

Figure 4.11 TSV Overpressurization Event at 102P/99F – Sensed Water Level

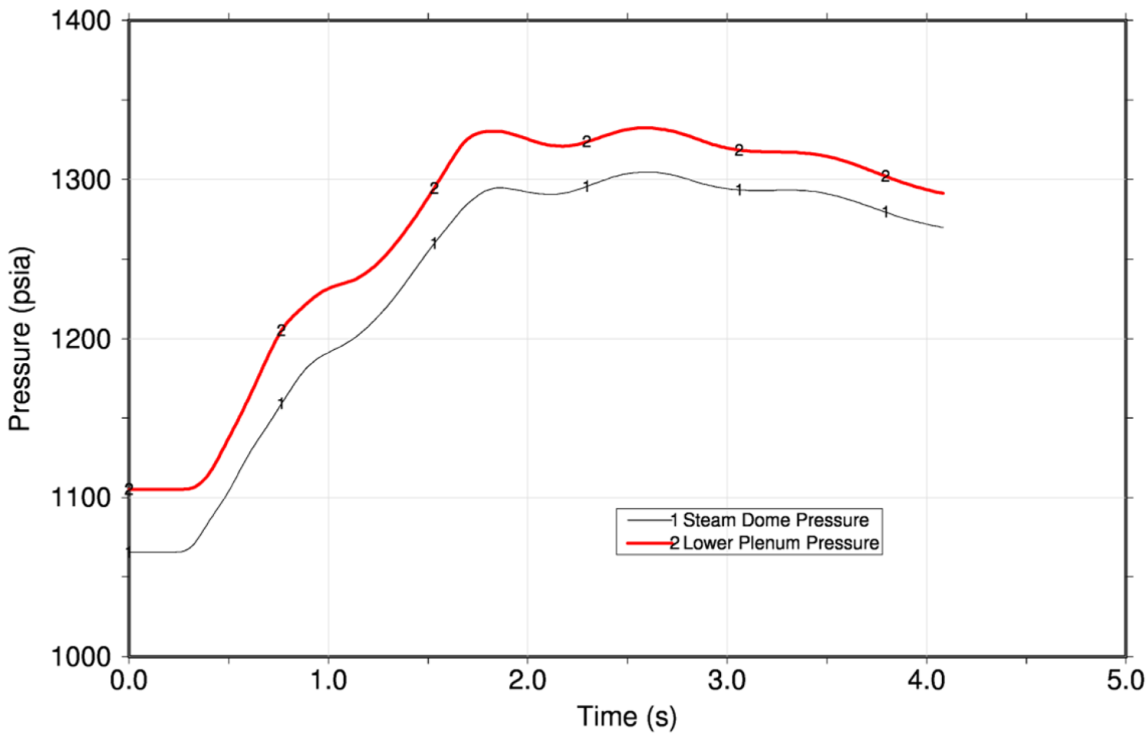

Figure 4.12 TSV Overpressurization Event at 102P/99F – Vessel Pressures

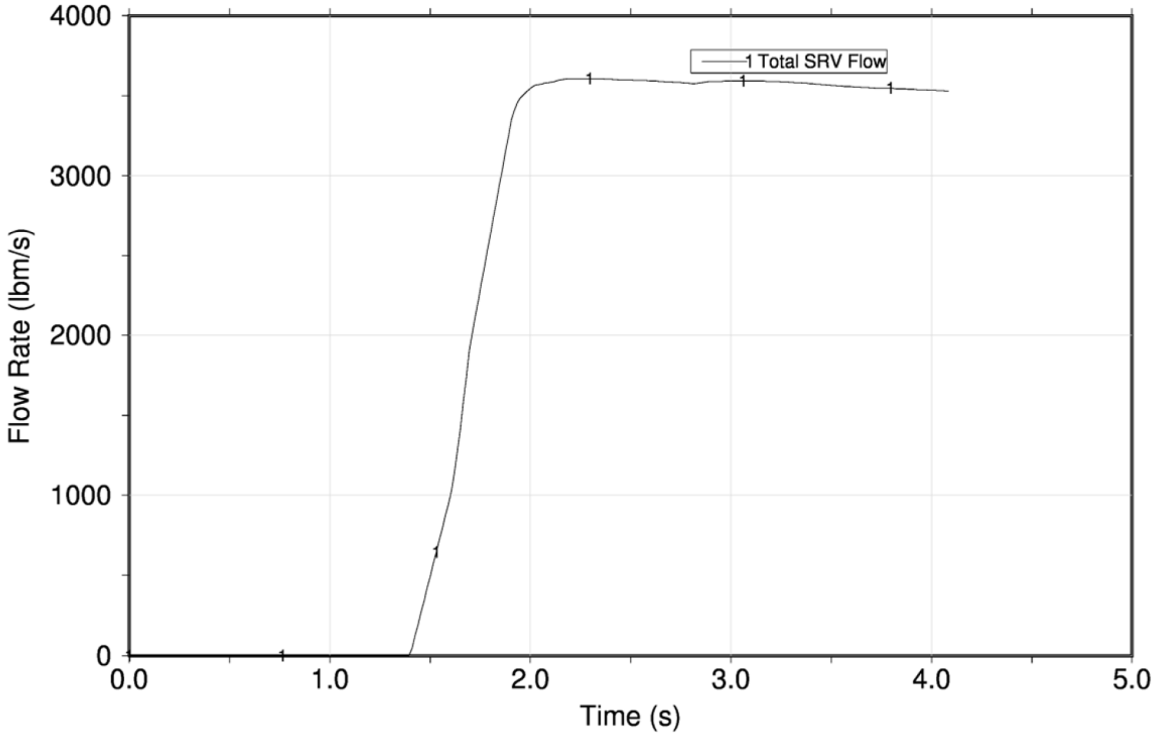

Figure 4.13 TSV Overpressurization Event at 102P/99F – Safety/Relief Valve Flow Rates

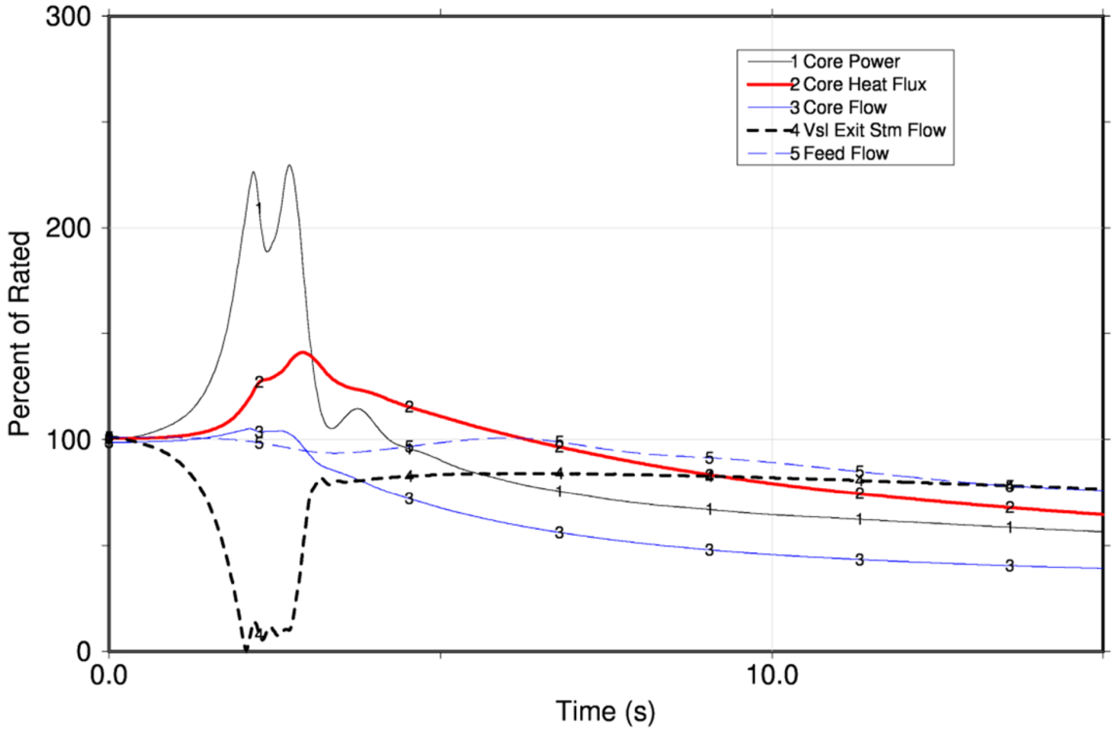

Figure 4.14 MSIV ATWS Overpressurization Event at 100P/99F – Key Parameters

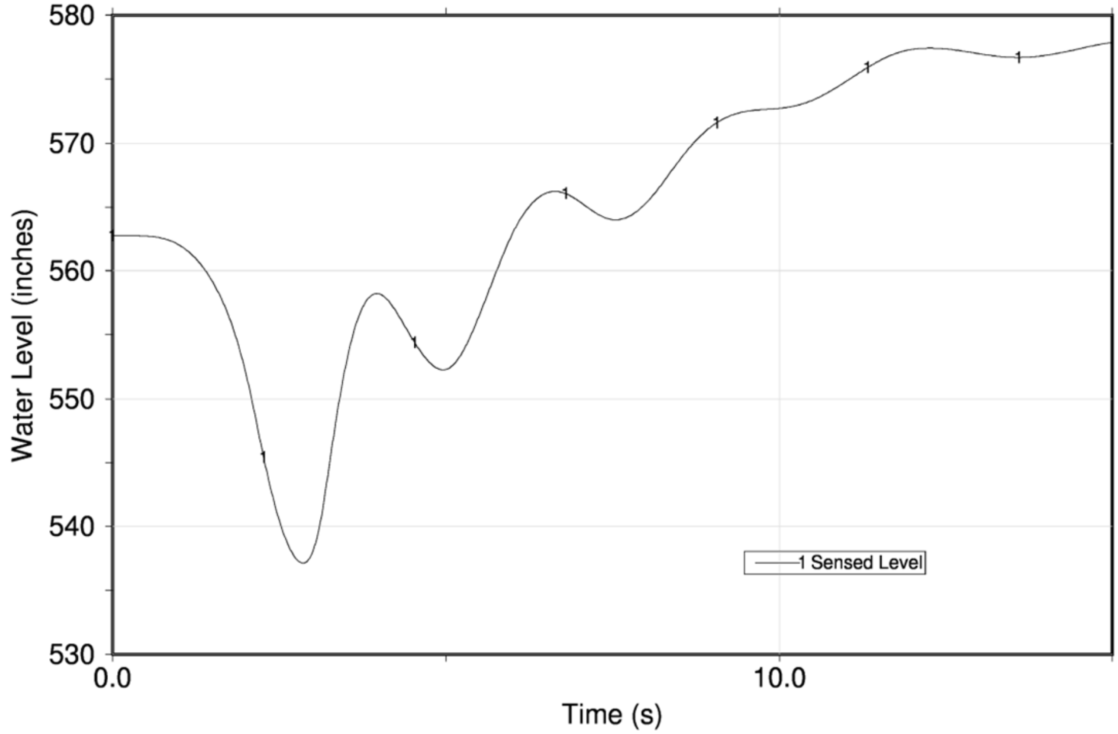

Figure 4.15 MSIV ATWS Overpressurization Event at 100P/99F – Sensed Water Level

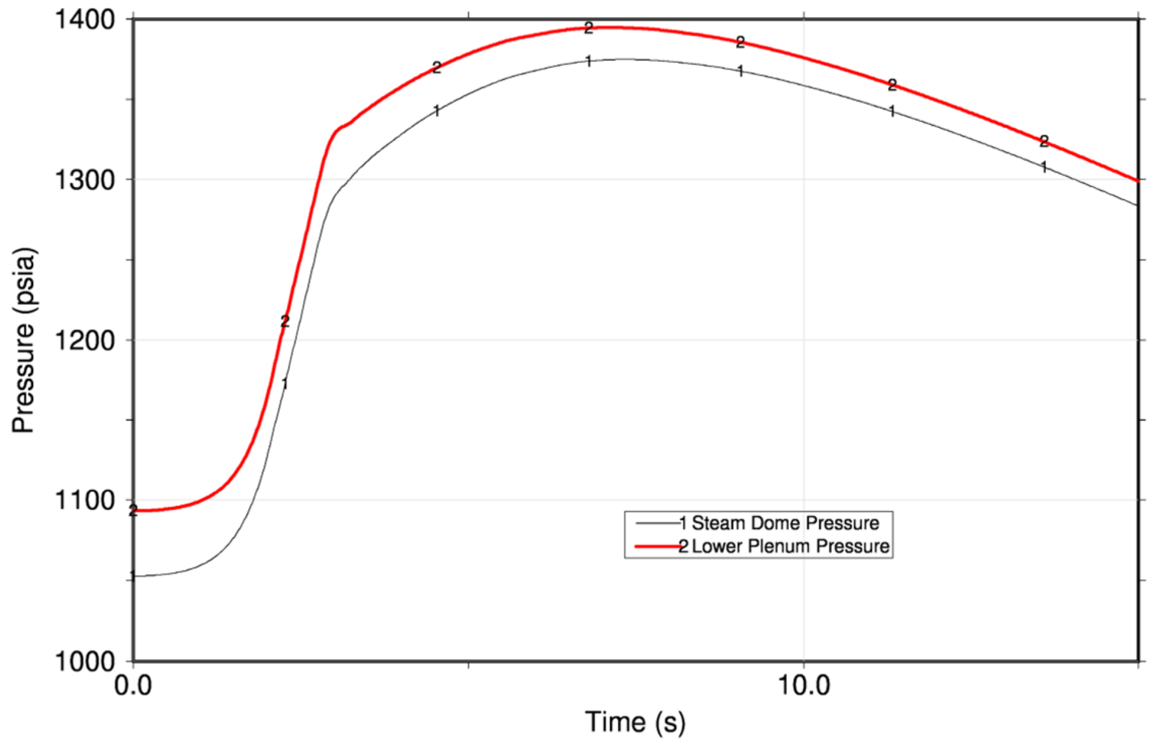

Figure 4.16 MSIV ATWS Overpressurization Event at 100P/99F – Vessel Pressures

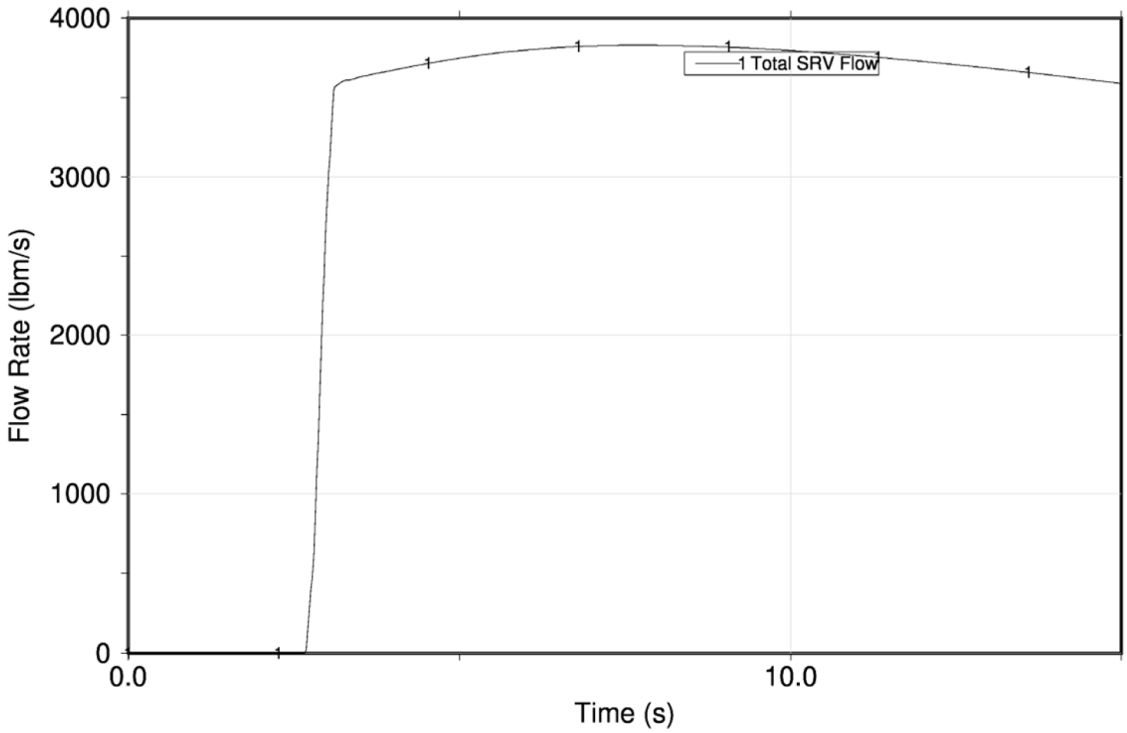

Figure 4.17 MSIV ATWS Overpressurization Event at 100P/99F – Safety/Relief Valve Flow Rates

5.0 References

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

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

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

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

5. ANP-3727P Revision 0, *Susquehanna ATRIUM 11 Equilibrium Cycle Fuel Cycle Design*, Framatome, October 2018.

6. ANP-3753P Revision 0, *Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel*, Framatome, May 2019.

7. ANP-3592P Revision 0, *Susquehanna Unit 1 Cycle 21 Principal Plant Parameters*, AREVA, June 2017.

8. EMF-2493(P) Revision 0, *MICROBURN-B2 Based Impact of Failed/Bypassed LPRMs and TIPs, Extended LPRM Calibration Interval, and Single Loop Operation on Measured Radial Bundle Power Uncertainty*, Siemens Power Corporation, December 2000.

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

Enclosure 14c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3783P, Susquehanna ATRIUM 11 Transient Demonstration

A F F I D A V I T

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3783P Revision 0, "Susquehanna ATRIUM 11 Transient Demonstration," dated June 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 7th day of June, 2019.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY
PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

Enclosure 15b of PLA-7783

Framatome Topical Report
ANP-3784NP

Susquehanna Units 1 and 2
LOCA Analysis for ATRIUM 11 Fuel

(Non-Proprietary Version)

framatome

Susquehanna Units 1 and 2 LOCA Analysis for ATRIUM 11 Fuel

ANP-3784NP
Revision 0

June 2019

ATRIUM, AURORA-B and S-RELAP5 are trademarks or registered trademarks of Framatome or its affiliates, in the USA or other countries.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1.	All	Initial Issue

Contents

Tables

Figures

Nomenclature

ADS	automatic depressurization system
BOL	beginning of life
BWR	boiling-water reactor
CFR	Code of Federal Regulations
CMWR	core average metal-water reaction
DC	direct current
DEG	double-ended guillotine
DG	diesel generator
ECCS	emergency core cooling system
FHOOS	feedwater heaters out-of-service
FSAR	final safety analysis report
HPCI	high-pressure coolant injection
ICF	increased core flow
ID	inside diameter
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MSIV	main stream isolation valve
MWR	metal-water reaction
NSSS	nuclear steam supply system
NRC	Nuclear Regulatory Commission, U.S.
OD	outside diameter
PCT	peak cladding temperature
PD	pump discharge
PS	pump suction
RCIC	reactor core isolation cooling
RDIV	recirculation discharge isolation valve
RHR	residual heat removal
RWCU	reactor water cleanup

SF-BATT	single failure of battery (DC) power
SF-HPCI	single failure of the HPCI system
SF-LPCI	single failure of an LPCI injection valve
SLO	single-loop operation
TLO	two-loop operation

1.0 Introduction

The results of a loss-of-coolant accident (LOCA) break spectrum and emergency core cooling system (LOCA-ECCS) analyses for Susquehanna Units 1 and 2 are documented in this report. The purpose of the break spectrum analysis is to identify the break characteristics that result in the highest calculated peak cladding temperature (PCT) **[ ]** during a postulated LOCA. The results provide the maximum average planar linear heat generation rate (MAPLHGR) limit for ATRIUM™ 11 fuel as a function of exposure for normal (two-loop) operation.

Variation in the following LOCA parameters is examined:

- Break location
- Break type (double-ended guillotine (DEG) or split)
- Break size
- Limiting ECCS single failure
- Axial power shape (top- or mid-peaked)
- Initial statepoint
- Fuel rod type

The analyses documented in this report are performed with LOCA Evaluation Models developed by Framatome*, and approved for reactor licensing analyses by the U.S. Nuclear Regulatory Commission (NRC). The models and computer codes used by Framatome for LOCA analyses are collectively referred to as the AURORA-B LOCA Evaluation Model (References 1 – 3). The calculations described in this report are performed in conformance with 10 CFR 50 Appendix K requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46.

Key model characteristics included in the report analyses are shown below. Other initial conditions used in the analyses are described in Section 4.0.

- Operation in the MELLLA domain of Figure 1.1 is supported. **[**

  ]
- **[**

  ]

---

* Framatome Inc. formerly known as AREVA Inc.

- The core is composed entirely of ATRIUM 11 fuel.
- A 2.0% increase in initial core power to address the maximum uncertainty in monitoring reactor power, as per NRC requirements, is included.
- [ ] were assumed to be at the MAPLHGR limit shown in Figure 2.1.
- [ ]

The limiting break characteristics from the break spectrum study are used in analyses to determine the MAPLHGR limit and [ ] versus exposure. Even though the limiting break will not change with exposure, the value of PCT calculated for any given set of break characteristics is dependent on exposure and the corresponding MAPLHGR and [ ].

Single-loop operation (SLO) results are discussed in Section 7.0. Long term coolability is addressed in Section 8.0.

Figure 1.1 Susquehanna Power / Flow Map

2.0 Summary of Results

The LOCA break spectrum and exposure analysis results presented in this report are applicable to Susquehanna Units 1 and 2. A more detailed discussion of results is provided in Sections 6.0 – 7.0.

The PCT and metal-water reaction (MWR) results, from the ATRIUM 11 fuel exposure-dependent analysis presented in Section 9.0, are presented below.

Parameter	ATRIUM 11*
Peak cladding temperature (°F)	1784 [ ]
Local cladding oxidation (max %)	4.64 [ ]
Total hydrogen generated (% of total hydrogen possible)	0.30

The MAPLHGR limit was determined by applying the AURORA-B LOCA Evaluation Model for the analysis of the limiting LOCA event. The exposure-dependent MAPLHGR limit for ATRIUM 11 fuel is shown in Figure 2.1. Exposure dependent results with the [ ] are presented in Section 9.0. The results of these calculations confirm that the LOCA acceptance criteria in the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below these limits.

The LOCA analysis results (i.e., the limiting break characteristics and exposure analysis) presented in this report are applicable for a full core of ATRIUM 11 fuel as well as transition cores containing ATRIUM 11 fuel. [ ]

* [ ]

[ ]

The SLO LOCA analyses support operation with an ATRIUM 11 multiplier of 0.80 applied to the normal two-loop operation MAPLHGR limit. [ ]

The long-term coolability evaluation confirms that the ECCS capacity is sufficient to maintain adequate cooling in an ATRIUM 11 core for an extended period after a LOCA.

All analyses also support the [ ]

The analysis supports operation in the MELLLA domain of the Susquehanna power/flow map shown in Figure 1.1.

[ ]

Figure 2.1 MAPLHGR Limit for ATRIUM 11 Fuel

[ ]

Figure 2.2 [ ] for ATRIUM 11 Fuel

3.0 LOCA Description

3.1 *Accident Description*

The LOCA is described in the Code of Federal Regulations 10 CFR 50.46 as a hypothetical accident that results in a loss of reactor coolant from breaks in reactor coolant pressure boundary piping up to and including a break equivalent in size to a double-ended rupture of the largest pipe in the reactor coolant system. There is not a specifically identified cause that results in the pipe break. However, for the purpose of identifying a design basis accident, the pipe break is postulated to occur inside the primary containment before the first isolation valve.

For a boiling water reactor (BWR), a LOCA may occur over a wide spectrum of break locations and sizes. Responses to the break vary significantly over the break spectrum. The largest possible break is a double-ended rupture of a recirculation pipe; however, this is not necessarily the most severe challenge to the ECCS. A double-ended rupture of a main steam line causes the most rapid primary system depressurization, but because of other phenomena, steam line breaks are seldom limiting with respect to the event acceptance criteria (10 CFR 50.46). Because of these complexities, an analysis covering the full range of break sizes and locations is performed to identify the limiting break characteristics.

Regardless of the initiating break characteristics, the event response is conveniently separated into three phases: the blowdown phase, the refill phase, and the reflood phase. The relative duration of each phase is strongly dependent upon the break size and location. **[**

]

During the blowdown phase of a LOCA, there is a net loss of coolant inventory, an increase in fuel cladding temperature due to core flow degradation, and for the larger breaks, the core becomes fully or partially uncovered. There is a rapid decrease in pressure during the blowdown phase. During the early phase of the depressurization, the exiting coolant provides core cooling. Consistent with the discussion presented in Reference 1, **[**

]

In the refill phase of a LOCA, the ECCS is functioning and there is a net increase of coolant inventory. During this phase the core sprays provide core cooling and, along with low-pressure and high-pressure coolant injection (LPCI and HPCI), supply liquid to refill the lower portion of the reactor vessel. In general, the core heat transfer to the coolant is less than the fuel decay heat rate and the fuel cladding temperature continues to increase during the refill phase.

In the reflood phase, the coolant inventory has increased to the point where the mixture level re-enters the core region. During the core reflood phase, cooling is provided above the mixture level by entrained reflood liquid and below the mixture level by pool boiling. Sufficient coolant eventually reaches the core hot node and the fuel cladding temperature decreases. **[**

]

3.2 *Acceptance Criteria*

A LOCA is a potentially limiting event that may place constraints on fuel design, local power peaking, and in some cases, acceptable core power level. During a LOCA, the normal transfer of heat from the fuel to the coolant is disrupted. As the liquid inventory in the reactor decreases, the decay heat and stored energy of the fuel cause a heatup of the undercooled fuel assembly. In order to limit the amount of heat that can contribute to the heatup of the fuel assembly during a LOCA, an operating limit on the MAPLHGR is applied to each fuel assembly in the core.

The Code of Federal Regulations prescribes specific acceptance criteria (10 CFR 50.46) for a LOCA event as well as specific requirements and acceptable features for Evaluation Models (10 CFR 50 Appendix K). The conformance of the AURORA-B LOCA Evaluation Models to Appendix K is described in Reference 1. The ECCS must be designed such that the plant response to a LOCA meets the following acceptance criteria specified in 10 CFR 50.46:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- The calculated local oxidation of the cladding shall nowhere exceed 0.17 times the local cladding thickness.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, except the cladding surrounding the plenum volume, were to react.

- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are commonly referred to as the PCT criterion, the local oxidation criterion, the hydrogen generation criterion, the coolable geometry criterion, and the long-term cooling criterion. A MAPLHGR limit is established for each fuel type to ensure that these criteria are met.

LOCA results are provided in Section 6.0 to identify the LOCA events which produce the highest PCT **[ ]** LOCA analysis results demonstrating that the PCT, local oxidation, and hydrogen generation (core wide oxidation) criteria are met are provided in Section 9.0. Compliance with these three criteria ensures that a coolable geometry is maintained. Long-term coolability criterion is discussed in Section 8.0.

4.0 LOCA Analysis Description

The Evaluation Model used for the break spectrum analysis is the AURORA-B LOCA analysis methodology described in Reference 1. The AURORA-B LOCA methodology employs two major computer codes to evaluate the system and fuel response during all phases of a LOCA. These are the S-RELAP5 and RODEX4 computer codes. A **[** **]** of the LOCA to determine the PCT and maximum local clad oxidation for **[** **]**

A complete analysis starts with the specification of fuel parameters using RODEX4 (Reference 3). RODEX4 is used to determine the **[** **]** The initial stored energy used in S-RELAP5 is **[** . **]**

4.1 *Break Spectrum Analysis*

S-RELAP5 is used to calculate the thermal-hydraulic response during all phases of the LOCA using a **[** **]** The reactor vessel nodalization is shown in Figure 4.1 and the core nodalization is shown in Figure 4.2 consistent with those in the topical report submitted to the NRC (Reference 1). The reactor core is modeled with heat generation rates determined from reactor kinetics equations with reactivity feedback and decay heat as required by Appendix K of 10 CFR 50. The clad swelling and rupture models from NUREG-0630 (Reference 2) have been incorporated into S-RELAP5.

The S-RELAP5 model is executed over a range of break locations, break sizes, break types, initial statepoints, axial shapes and assumed single-failures to determine the break that yields the highest PCT **[** **]**

4.2 *Exposure Analysis*

The [ ] from beginning-of-life to end-of-life [ ] increments to determine an exposure-dependent MAPLHGR limit and [ ] Figures of merit including PCT, local cladding oxidation, and core-wide metal-water reaction are evaluated over the range of exposures to confirm the acceptability of the LOCA analysis with respect to 10 CFR 50.46 criteria. [ ]

4.3 *Plant Parameters*

The LOCA analysis is performed using the plant parameters provided by the utility. Table 4.1 provides a summary of reactor initial conditions used in the break spectrum analysis. Table 4.2 lists selected reactor system parameters.

The LOCA analysis is performed for a full core of ATRIUM 11 fuel. Some of the key fuel parameters used in the analysis are summarized in Table 4.3.

4.4 *ECCS Parameters*

Table 4.4 – Table 4.7 provide the important ECCS characteristics assumed in the analysis. The ECCS is modeled as time-dependent junctions connected to the appropriate reactor locations: LPCS injects into the upper plenum, HPCI injects into the upper downcomer, and LPCI injects into the recirculation lines.

The flow through each ECCS valve is determined based on system pressure and valve position. Flow versus pressure for a fully open valve is obtained by linearly interpolating the pump capacity data provided in Table 4.4 – Table 4.6. No credit for ECCS flow is assumed until the ECCS injection valves are fully open and the ECCS pumps reach rated speed.

The ADS valves are modeled as a junction connecting the reactor steam line to the suppression pool. The flow through the ADS valves is calculated based on pressure and valve flow characteristics. The valve flow characteristics are determined such that the calculated flow is equal to the rated capacity at the reference pressure shown in Table 4.7. All six ADS valves are assumed operable and the potential single failure of one ADS valve is analyzed.

In the Framatome LOCA analysis model, ECCS initiation is assumed to occur when the water level drops to the applicable level setpoint. No credit is assumed for the start of LPCS or LPCI due to high drywell pressure. **[**

]

Table 4.1 Initial Conditions

Reactor power (% of rated)	102	102	[ ]
[			]
Reactor power (MWt)	4031	4031	[ ]
[			]
[			]
Steam flow rate (Mlb/hr)	17.0	17.0	10.9
Steam dome pressure (psia)	1054.7	1054.6	998.8
Core inlet enthalpy (Btu/lb)	524.4	521.9	501.5
[			]
[			]
[			]
Rod average power distributions	Figure 4.3	Figure 4.4	Figure 4.5

* [ ]

Table 4.2 Reactor System Parameters

Parameter	Value
Vessel ID (in)	251
Number of fuel assemblies	764
Recirculation suction pipe area (ft^2)	3.503
1.0 DEG suction break area (ft^2)	7.006
Recirculation discharge pipe area (ft^2)	3.503
1.0 DEG discharge break area (ft^2)	7.006

Table 4.3 ATRIUM 11 Fuel Assembly Parameters

Table 4.4 High-Pressure Coolant Injection Parameters

Parameter	Value
Coolant temperature (maximum) (°F)	100
Initiating Signals and Setpoints	
Water level (in)*	457.5
High drywell pressure (psig)	Not used
Time Delays	
Startup time (sec)	1.0
Delay to startup (sec)	34.0
Delivered Coolant Flow Rate Versus Pressure	
Differential Pressure (psid)	Flow Rate (gpm)
0	0
128	0
165	4500
1210	4500

* Relative to vessel zero.

Table 4.5 Low-Pressure Coolant Injection Parameters

Parameter	Value
Reactor pressure permissive for opening valves – analytical (psig)	380
Coolant temperature (maximum) (°F)	120
Initiating Signals and Setpoints	
Water level (in)*	366.5
High drywell pressure (psig)	Not used
Time Delays	
Diesel generator startup time (sec)	25.1
Diesel generator power at pump (sec)	4.0
LPCI pump at rated speed (sec)	7.5
Start opening injection valves (sec)	9.0
LPCI injection valve stroke time (sec)	24.0

Delivered Coolant Flow Rate Versus Pressure †

Differential Pressure (psid)	2 Pumps/Loop Flow Rate (gpm)	Differential Pressure (psid)	1 Pump/Loop Flow Rate (gpm)
0	19,307	0	11,347
79	16,193	53	10,161
152	12,422	177	6,420
230	6,283	237	2,971
257	3,033	266	466
272	0	270	0

* Relative to vessel zero.

† Selected values.

Table 4.6 Low-Pressure Core Spray Parameters

Parameter	Value
Reactor pressure permissive for opening valves - analytical (psig)	380
Coolant temperature (maximum) (°F)	120
Initiating Signals and Setpoints	
Water level (in)*	366.5
High drywell pressure (psig)	Not used
Time Delays	
Diesel generator start time (sec)	25.1
Diesel generator power at pump (sec)	11.5
LPCS pump at rated speed (sec)	3.5
Start opening injection valves (sec)	9.0
LPCS injection valve stroke time (sec)	19.0
Delivered Coolant Flow Rate Versus Pressure†	
Differential Pressure (psid)	Flow Rate per Pump (gpm)
0	6,785
181	4,115
218	3,215
278	1,045
298	0

* Relative to vessel zero.

† Selected values.

Table 4.7 Automatic Depressurization System Parameters

Parameter	Value
Number of valves installed	6
Number of valves available*	6
Minimum flow capacity of available valves (Mlbm/hr at psig)	4.8 at 1125
Initiating Signals and Setpoints	
Water level (in)†	366.5
High drywell pressure (psig)	Not used
Time Delays	
ADS timer (delay time from initiating signal to time valves are open (sec)	120

* SF-ADS is explicitly modeled such that all 6 ADS valves are available for the other single failure scenarios.

† Relative to vessel zero.

[ ]

Figure 4.1 S-RELAP5 Vessel Model

Figure 4.2 S-RELAP5 Core Model

[

]

Figure 4.3 Rod Average Power Distributions for 102%P and [ ] Mid- and Top-Peaked

[ ]

Figure 4.4 Rod Average Power Distributions for 102%P and [ ] Mid- and Top-Peaked

[

]

Figure 4.5 Rod Average Power Distributions
for [ ]
Mid- and Top-Peaked

5.0 Break Spectrum Analysis Description

The objective of the LOCA break spectrum analyses is to ensure that the operating conditions, break location, break type, break size, and ECCS single failure which produce the maximum PCT [ ] are identified. The LOCA response scenario varies considerably over the spectrum of break locations. Potential break locations have been separated into two groups: recirculation line breaks and non-recirculation line breaks. The basis for the break locations and potentially limiting single failures analyzed in this report is described in the following sections.

5.1 *Limiting Single Failure*

Regulatory requirements specify that the LOCA analysis consider availability of offsite power supplies and that only safety grade systems and components are available. In addition, regulatory requirements also specify that the most limiting single failure of ECCS equipment must be assumed in the LOCA analysis. The term "most limiting" refers to the ECCS equipment failure that produces the greatest challenge to event acceptance criteria. The limiting single failure can be a common power supply, an injection valve, a system pump, or system initiation logic. The most limiting single failure may vary with break size and location. The potential limiting single failures identified in the FSAR (Reference 4) are shown below:

- Backup battery power (SF-BATT)
- Opposite unit false LOCA signal (SF-LOCA)
- Low-pressure coolant injection valve (SF-LPCI)
- Diesel generator (SF-DGEN)
- High-pressure coolant injection system (SF-HPCI)
- Automatic depressurization system valve (SF-ADS)

The single failures and the available ECCS for each failure assumed in these analyses are summarized in Table 5.1. Other potential failures are not specifically considered because they result in as much or more ECCS capacity.

The scope of calculations needed to evaluate the single failures listed in Table 5.1 is reduced by comparing the ECCS systems available for each single failure scenario.

- No SF-DGEN calculations are needed. The SF-LOCA and SF-DGEN scenarios each model ADS, HPCI, 1 LPCS and at least 1 LPCI pump. For PS breaks, SF-DGEN adds 2 additional LPCI pumps while SF-LOCA adds only 1 additional LPCI pump. Therefore, the ECCS resources for SF-LOCA equal or conservatively bound those of SF-DGEN.
- No SF-HPCI calculations are needed. The SF-BATT and SF-HPCI scenarios each model ADS and a failure of the HPCI system. For all recirculation break locations, SF-HPCI has 1 additional LPCS pump and 1 additional LPCI pump. Therefore, the ECCS resources available for SF-BATT conservatively bound those of SF-HPCI.

Therefore, break spectrum calculations that evaluate the SF-ADS, SF-BATT, SF-LOCA, and SF-LPCI single failure scenarios will assure that the limiting failure is considered in the analysis.

5.2 *Recirculation Line Breaks*

The response during a recirculation line LOCA is dependent on break size. The rate of reactor vessel depressurization decreases as the break size decreases. The high-pressure ECCS and ADS will assist in reducing the reactor vessel pressure to the pressure where the LPCI and LPCS flows start. For large breaks, rated LPCS and LPCI flow is generally reached before or shortly after the time when the ADS valves open so the ADS system is not required to mitigate the LOCA. ADS operation is an important emergency system for small breaks where it assists in depressurizing the reactor system faster, and thereby reduces the time required to reach rated LPCS and LPCI flow.

The two largest flow resistances in the recirculation piping are the recirculation pump and the jet pump nozzle. For breaks in the discharge piping, there is a major flow resistance in both flow paths from the reactor vessel to the break. For breaks in the suction piping, both major flow resistances are in the flow path from the vessel to the pump side of the break. As a result, pump suction side breaks experience a more rapid blowdown, which tends to make the event more severe. For suction side breaks, the recirculation discharge isolation valve on the broken loop closes which allows the LPCI flow to fill the discharge piping and supply flow to the lower plenum and core. For discharge side breaks, the LPCI flow in the broken loop is assumed to exit the system through the break resulting in a decrease in available LPCI flow to the core, thereby increasing the severity of the event. Both suction and discharge recirculation pipe breaks are considered in the break spectrum analysis.

Two break types (geometries) are considered for the recirculation line break. The two types are the double-ended guillotine (DEG) break and the split break.

For a DEG break, the piping is assumed to be completely severed resulting in two independent flow paths to the containment. The DEG break is modeled by setting the break area (at both ends of the pipe) equal to the full pipe cross-sectional area and varying the discharge coefficient between 1.0 and 0.4. The range of discharge coefficients is used to cover uncertainty in the actual geometry at the break. [ ] The most limiting DEG break is determined by varying the discharge coefficient.

A split type break is assumed to be a longitudinal opening or hole in the piping that results in a single break flow path to the containment. Appendix K of 10 CFR 50 defines the cross-sectional area of the piping as the maximum split break area required for analysis.

Break types, break sizes, and single failures are analyzed for both suction and discharge recirculation line breaks.

Section 6.0 provides a description and results summary for breaks in the recirculation line.

5.3 *Non-Recirculation Line Breaks*

In addition to breaks in the recirculation line, breaks in other reactor coolant system piping must be considered in the LOCA break spectrum analysis. Although the recirculation line large breaks result in the largest coolant inventory loss, they do not necessarily result in the most severe challenge to event acceptance criteria. The double-ended rupture of a main steam line is expected to result in the fastest depressurization of the reactor vessel. Special consideration is required when the postulated break occurs in ECCS piping. Although ECCS piping breaks are small relative to a recirculation pipe DEG break, the potential to disable an ECCS system increases their severity.

The following sections address potential LOCAs due to breaks in non-recirculation line piping.

Non-recirculation line breaks outside containment are inherently less challenging to fuel limits than breaks inside containment. For breaks outside containment, isolation or check valve closure will terminate break flow prior to the loss of significant liquid inventory and the core will remain covered. If high-pressure coolant inventory makeup cannot be reestablished, ADS actuation may become necessary. [

] Although analyses of breaks outside containment may be required to address non-fuel related regulatory requirements, these breaks are not limiting relative to fuel acceptance criteria such as PCT.

5.3.1 Main Steam Line Breaks

A steam line break [

] The break results in high steam flow out of the broken line and into the containment. Prior to MSIV closure, a steam line break also results in high steam flow in the intact steam lines as they feed the break via the steam line manifold. A steam line break inside containment results in a rapid depressurization of the reactor vessel. Initially the break flow will be high quality steam; however, the rapid depressurization produces a water level swell that results in liquid discharge at the break. For steam line breaks, the largest break size is most limiting because it results in the most level swell and liquid loss out of the break.

[

]

5.3.2 Feedwater Line Breaks

[

]

5.3.3 HPCI Line Breaks

The HPCI injection line is connected to the feedwater line outside containment.

[

]

The HPCI steam supply line is connected to the main steam line inside containment.

[

]

5.3.4 LPCS Line Breaks

A break in the LPCS line is expected to have many characteristics similar to [

] However, some characteristics of the LPCS line break are unique and are not addressed in other LOCA analyses. Two important differences from other LOCA analyses are that the break flow will exit from the region inside the core shroud and the break will disable one LPCS system. The LPCS line break is assumed to occur just outside the reactor vessel. [

]

5.3.5 LPCI Line Breaks

The LPCI injection lines are connected to the larger recirculation discharge lines. [ ]

5.3.6 RCIC Line Breaks

The reactor core isolation cooling (RCIC) line discharges to the feedwater line, therefore a break in the RCIC discharge line is equivalent to a feedwater line break of the same size.

The steam supply to the RCIC turbine comes from the main steam line from the reactor vessel; therefore, a break in the RCIC turbine steam supply is equivalent to a main steam line break of the same size.

5.3.7 RWCU Line Breaks

The reactor water cleanup (RWCU) extraction line is connected to a recirculation suction line with an additional connection to the vessel bottom head. [ ]

The RWCU return line is connected to the feedwater line; [ ]

5.3.8 Shutdown Cooling Line Breaks

The shutdown cooling suction piping is connected to a recirculation suction line and the shutdown cooling return line is connected to a recirculation discharge line. [ ]

5.3.9 Instrument Line Breaks

[ ]

Table 5.1 Available ECCS for
Recirculation Line Break LOCAs

Assumed Failure	Recirculation Suction Break Systems Remaining*	Recirculation Discharge Break Systems Remaining
Battery (SF-BATT)	6 ADS, 1 LPCS†, 3 LPCI‡	6 ADS, 1 LPCS, 1 LPCI
Opposite unit false LOCA signal (SF-LOCA)	6 ADS, HPCI, 1 LPCS, 2 LPCI§	6 ADS, HPCI, 1 LPCS, 1 LPCI
LPCI injection valve (SF-LPCI)	6 ADS, HPCI, 2 LPCS, 2 LPCI‡	6 ADS, HPCI, 2 LPCS
Diesel generator (SF-DGEN)	6 ADS, HPCI, 1 LPCS, 3 LPCI‡	6 ADS, HPCI, 1 LPCS, 1 LPCI
HPCI (SF-HPCI)	6 ADS, 2 LPCS, 4 LPCI‡	6 ADS, 2 LPCS, 2 LPCI
ADS (SF-ADS)	5 ADS, HPCI, 2 LPCS, 4 LPCI‡	5 ADS, HPCI, 2 LPCS, 2 LPCI

* Systems remaining, as identified in this table for recirculation suction line breaks, are applicable to other non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed for recirculation suction breaks, less the ECCS in which the break is assumed.

† Each LPCS means operation of two core spray pumps in a system. It is assumed that both pumps in a system must operate to take credit for core spray cooling or inventory makeup in that loop.

‡ Two LPCI pumps inject into broken loop.

§ One LPCI pump injects into broken loop.

6.0 TLO Recirculation Line Break Spectrum Analyses

The largest diameter recirculation system pipes are the suction line between the reactor vessel and the recirculation pump and the discharge line between the recirculation pump and the riser manifold ring. LOCA analyses are performed for breaks in both of these locations with consideration for both DEG and split break geometries. The break sizes considered included DEG breaks with discharge coefficients from 1.0 to 0.4 and split breaks with areas ranging between the full pipe area and **[ ]** ft^2. As discussed in Section 5.0, the single failures considered in the recirculation line break analyses are SF-ADS, SF-BATT, SF-LOCA, and SF-LPCI.

[

]

6.1 *Break Spectrum Analysis Results*

The break spectrum analyses demonstrate that the recirculation line break case with the highest PCT **[ ]** is the 0.07 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 102% rated core power and **[ ]** These two cases are presented in Table 6.1.

Table 6.2 provides a summary of the **[ ]** from the recirculation line break calculations for each of the single failures, state points, and axial power shapes. The event times for the **[ ]** are presented in Table 6.3 and plots of key parameters from the LOCA analyses of this case are provided in Figures 6.1 – 6.15.

Table 6.1 Break Spectrum Results* for TLO Recirculation Line Breaks

Break spectrum case resulting **[ ]**	0.07 ft^2 pump discharge SF-BATT Top-peaked axial 102%P/**[ ]**
Break spectrum case resulting **[ ]**	0.07 ft^2 pump discharge SF-BATT Top-peaked axial 102%P/**[ ]**

* The cases identified in Table 6.1 from the TLO break spectrum analyses are further evaluated in Section 9.0 with exposure dependent analysis.

Table 6.2 Summary of Break Spectrum [ ] for TLO Recirculation Line Breaks

[ ]

Table 6.3 Event Times for the [ ] from the TLO Recirculation Line Break Spectrum Analysis

[ ]

Figure 6.1 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
Upper Plenum Pressure

Figure 6.2 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
Total Break Flow Rate

Figure 6.3 [ ] from the TLO Recirculation Line Break Spectrum Analysis Core Inlet Flow Rate

Figure 6.4 [ ] from the TLO Recirculation Line Break Spectrum Analysis ADS Flow

[ ]

Figure 6.5 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
HPCI Flow

[ ]

Figure 6.6 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
LPCS Flow

Figure 6.7 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
LPCI Flow

Figure 6.8 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
RDIV Flows

[ ]

Figure 6.9 [ ] from the TLO Recirculation Line Break Spectrum Analysis Relief Valve Flow

[ ]

Figure 6.10 [ ] from the TLO Recirculation Line Break Spectrum Analysis Downcomer LOCA Water Level

**Figure 6.11 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
Upper Plenum Liquid Level**

**Figure 6.12 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
Hot Channel Liquid Level**

[ ]

**Figure 6.13 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
Core Bypass Liquid Level**

[ ]

**Figure 6.14 [ ] from the
TLO Recirculation Line Break Spectrum Analysis
Lower Plenum Liquid Level**

[ ]

Figure 6.15 [ ] from the TLO Recirculation Line Break Spectrum Analysis Hot Channel Inlet Flow

7.0 Single-Loop Operation LOCA Analysis

During SLO, the pump in one recirculation loop is not operating. A break may occur in either loop, but results from a break in the inactive loop would be similar to those from a two-loop operation break. If a break occurs in the inactive loop during SLO, the intact active loop flow to the reactor vessel would continue during the recirculation pump coastdown period and would provide core cooling similar to that which would occur in breaks during TLO. The system response would be similar to that resulting from an equal-sized break during two-loop operation. A break in the active loop during SLO results in a more rapid loss of core flow and earlier degraded core conditions relative to those from a break in the inactive loop. Therefore, only breaks in the active recirculation loop are analyzed.

A break in the active recirculation loop during SLO will result in an earlier loss of core heat transfer relative to a similar break occurring during two-loop operation. This occurs because there will be an immediate loss of jet pump drive flow. Therefore, fuel rod surface temperatures will increase faster in an SLO LOCA relative to a TLO LOCA. Also, the early loss of core heat transfer will result in higher stored energy in the fuel rods at the start of the heatup. The increased severity of an SLO LOCA can be reduced by applying an SLO multiplier to the two-loop MAPLHGR limit.

7.1 *SLO Analysis Modeling Methodology*

[

]

7.2 *SLO Analysis Results*

[

]

The SLO analyses are performed with a 0.80 multiplier applied to the two-loop MAPLHGR limit resulting in an SLO MAPLHGR limit of [ ] kW/ft. [

] The analyses are performed at maximum stored energy fuel conditions. The limiting SLO LOCA 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 [

]

A comparison of the limiting SLO and the limiting two-loop results is provided in Table 7.1. The results in Table 7.1 show that the two-loop LOCA results bound the limiting SLO results when a 0.80 multiplier is applied to the two-loop MAPLHGR limit. [

]

**Table 7.1 Single- and Two-Loop Operation
PCT Summary**

Operation	Limiting Case	PCT (°F)
Single-loop	0.09 ft^2 pump discharge top-peaked SF-BATT	[ ]
Two-loop	0.07 ft^2 pump discharge top-peaked SF-BATT	[ ]

8.0 Long-Term Coolability

Long-term coolability addresses the issue of reflooding the core and maintaining a water level adequate to cool the core and remove decay heat for an extended time period following a LOCA. For non-recirculation line breaks, the core can be reflooded to the top of the active fuel and be adequately cooled indefinitely. For recirculation line breaks, the core will initially remain covered following reflood due to the static head provided by the water filling the jet pumps to a level of approximately two-thirds core height. Eventually, the heat flux in the core will not be adequate to maintain a two-phase water level over the entire length of the core. Beyond this time, the upper third of the core will remain wetted and adequately cooled by core spray. Maintaining water level at two-thirds core height with one core spray system operating is sufficient to maintain long-term coolability as demonstrated by the NSSS vendor (Reference 5). Since fuel temperatures during long-term cooling are low relative to the PCT and are not significantly affected by fuel design, this conclusion is applicable to ATRIUM 11 fuel. This LOCA analysis assesses conditions from the time of the initiation of the break to the time when long term cooling conditions can be established as demonstrated in Reference 5.

9.0 Exposure-Dependent LOCA Analysis Description and Results

Exposure-dependent LOCA results for ATRIUM 11 fuel are obtained by repeated analyses based on the cases identified in Table 6.1 from the break spectrum analysis **[** **]**

Table 9.1 shows the exposure-dependent LOCA analysis results for the ATRIUM 11 fuel. The S-RELAP5 model is applied to obtain these results as described in Section 4.2. The analysis is performed at **[** **]** which ensures appropriate limits are applied up to the monitored maximum assembly average and rod average exposure limits. The MAPLHGR input is consistent with the data in Figure 2.1. **[** **]** Exposure-dependent fuel rod data is provided from RODEX4 results **[** **]** The impact of thermal conductivity degradation is addressed with RODEX4.

The ATRIUM 11 limiting PCT is 1784°F at **[** **]** exposure for the 0.07 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 102% rated core power and **[** **]**. The maximum local MWR of 4.64% occurred at **[** **]** exposure, **[** **]** Analysis results show that the hot rod average MWR is 0.30%. Since all other rods in the core are at lower power, the core average metal water reaction (CMWR) will be significantly less than 0.30%.

Figure 9.1 shows the cladding temperature of the ATRIUM 11 PCT rod as a function of time for the limiting PCT result from the exposure-dependent LOCA analysis. The maximum temperature of 1784°F occurs at **[** **]**. These results demonstrate the acceptability of the ATRIUM 11 MAPLHGR limit shown in Figure 2.1.

Table 9.1 ATRIUM 11 Exposure-Dependent LOCA Analysis Results

[ ]

CMWR is < 0.30% at all exposures.*

* The rod average MWR for the hot rod is 0.30% which supports the conclusion that the CMWR is less than 0.30%.

[ ]

Figure 9.1 Limiting [ ] PCT
Exposure-Dependent LOCA Analysis

10.0 Conclusions

The AURORA-B LOCA Evaluation Model was applied to confirm the acceptability of the ATRIUM 11 MAPLHGR limit and **[ ]** for Susquehanna Units 1 and 2. The following conclusions were made from the analyses presented in this report.

- The limiting PCT is obtained from Section 9.0 based on a recirculation line break of 0.07 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 102% of rated core power and **[ ]**.
- The limiting break analysis identified above satisfies all the acceptance criteria specified in 10 CFR 50.46. The analysis is performed in accordance with 10 CFR 50 Appendix K requirements.
- The multiplier applied to the MAPLHGR limit for SLO is 0.80 for ATRIUM 11 fuel. **[ ]** This multiplier ensures that a LOCA from SLO is less limiting than a LOCA from two-loop operation.
- The acceptance criteria of the Code of Federal Regulations (10 CFR 50.46) are met for operation at or below the ATRIUM 11 MAPLHGR limit given in Figure 2.1 **[ ]**.
    - Peak PCT < 2200°F.
    - Local cladding oxidation thickness < 17%.
    - Total hydrogen generation < 1%.
    - Coolable geometry, satisfied by meeting peak PCT, local cladding oxidation, and total hydrogen generation criteria.
    - Core long-term cooling, satisfied by concluding core flooded to top of active fuel or core flooded to the jet pump suction elevation (Reference 1).
- The MAPLHGR limit and **[ ]** are applicable for ATRIUM 11 full cores as well as transition cores containing ATRIUM 11 fuel.

11.0 References

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

2. XN-NF-82-07(P)(A) Revision 1, *Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model*, Exxon Nuclear Company, November 1982.

3. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,* Framatome, February 2008.

4. *Susquehanna Steam Electric Station Units 1 and 2 Final Safety Analysis Report*, Revision 67.

5. NEDO-20566A*, General Electric Company Analytical Model for Loss of Coolant Analysis in Accordance with 10CFR50 Appendix K*, September 1986.

Appendix A Limitations from the Safety Evaluation for LTR ANP-10332PA

Compliance to the limitations and conditions from Section 5 of the safety evaluation in ANP-10332PA, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Loss of Coolant Accident Scenarios" (Reference 1) is discussed in the following table.

Appendix A *(Continued)*

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
1	The AURORA-B LOCA evaluation model shall be supported by an approved nodal core simulator and lattice physics methodology. Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model shall identify the nodal core simulator and lattice physics methods supporting the AURORA-B LOCA analysis and reference an NRC-approved TR confirming their acceptability for the intended application.	MICROBURN-B2 and the underlying cross section generation code, CASMO-4, are used for the nodal core simulator and lattice physics methodology from the following NRC-approved TR: EMF-2158(P)(A) Revision 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2,” Siemens Power Corporation, October 1999.
2	The full, stand-alone version of the RODEX4 code shall be used in accordance with an approved methodology to supply steady-state fuel thermal-mechanical inputs to the AURORA-B LOCA evaluation model.	The stand-alone version of RODEX4 is used to supply steady-state fuel thermal-mechanical input in accordance with the following NRC-approved methodology: BAW-10247PA Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” AREVA NP Inc., February 2008.
3	The AURORA-B LOCA evaluation model may not be used to perform analyses that result in any of its constituent components or supporting codes (i.e., S-RELAP5, RODEX4 kernel, RODEX4, core simulator and lattice physics methods) being operated outside approved limits documented in their respective TRs, SEs, code manuals, and plant-specific licensing applications.	The analyses are within the limits of the TRs, SEs, code manuals and plant-specific licensing applications.
4	TR ANP-10332P **[** **]**.	**[** **]** LOCA report.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
5	As discussed above in Section 2.1, the conclusions of this SE apply only to the use of the AURORA-B LOCA evaluation model for the purpose of demonstrating compliance with relevant regulatory requirements in effect at the time the NRC staff's technical review of ANP-10332P was completed (i.e., as of December 31, 2018).	The analyses only apply regulatory requirements in effect at the time the NRC staff's review was completed. They [ ].
6	This SE does not constitute [ ] of the evaluation model.	The evaluation model [ ].
7	[ ].	The [ ].
8	[ ].	The [ ] in the analyses.
9	Safety analyses performed with the AURORA-B LOCA evaluation model [ ].	[ ].

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
10	To ensure adequate conservatism in future plant-specific safety analyses, absent specific NRC staff approval for higher values, this SE limits [ ].	A [ ].
11	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model [ ].	BWR fuel rods are [ ].
12	The Appendix K lockout preventing the return to nucleate boiling [ ].	The analyses [ ].
13	[ ].	[ ].

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
14	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[** **]**.	Analyses **[** **]**.
15	**[** **]**.	The **[** **]**.
16	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model **[** **]**.	**[** **]**.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
17	To assure satisfaction of GDC 35 (or similar plant-specific design criterion), [ ].	A [ ].
18	Safety analyses performed with the AURORA-B LOCA evaluation model [ ].	[ ].
19	Safety analyses for [ ].	This application of AURORA-B LOCA [ ]. [ ]. Approximately [ ]

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
		[**]**.
20	Simulations supporting plant safety analyses **[** **]**.	Simulations **[** **]**.
21	As discussed in Section 3.3.5.7, Framatome used a **[** **]**	The **[** **]**.

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
	[ ].	
22	The NRC staff has not specifically reviewed any plant parameters in ANP-10332P or deemed them acceptable for use in plant safety analyses. Therefore, [ ].	The licensee [ ].
23	Safety analyses performed with the AURORA-B LOCA evaluation model shall include justification that [ ].	A [ ].

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
24	[ ].	[ ].
25	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model [ ]	The [ ].
26	Plant-specific licensing applications referencing the AURORA-B LOCA evaluation model [ ].	The [ ].

Limitation and Condition Number	Limitation and Condition Description	Disposition/Discussion
27	As discussed in Section 4.3 of this SE, new or modified Framatome **[** **]**.	The analyses **[** **]**.

Enclosure 15c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3784P, Susquehanna Units 1 and 2
LOCA Analysis for ATRIUM 11 Fuel

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3784P Revision 0, "Susquehanna Units 1 and 2 LOCA Analysis for ATRIUM 11 Fuel," dated June 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 7th

day of June, 2019.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

Enclosure 16b of PLA-7783

Framatome Topical Report
ANP-3771NP

Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology

(Non-Proprietary Version)

framatome

Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology

ANP-3771NP
Revision 0

May 2019

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	Definition
ASME	American Society of Mechanical Engineers
AST	alternate source term
BOC	beginning of cycle
BPWS	banked position withdrawal sequence
BWR	boiling water reactor
CFR	code of federal regulations
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
CWSR	cold work stress relief aka SRA
CZP	cold zero power
EFP	end of full power uprate
EOC	end of cycle
GDC	general design criteria
LBRF	licensing basis release fraction
LTR	licensing topical report
LWR	light water reactor
MOC	middle of cycle corresponding to peak hot excess
NRC	Nuclear Regulatory Commission, U.S.
PCMI	pellet clad mechanical interaction
RIA	reactivity insertion accident
RPS	reactor protection system
SE	safety evaluation
SRA	stress relief annealed
SRP	standard review plan
SSRF	steady state release fraction
TFGR	transient fission gas release
U. S. NRC	Nuclear Regulatory Commission, U. S.
ΔH	transient change in enthalpy
ΔH_p	prompt enthalpy rise
ΔH_tot	total enthalpy rise

1.0 INTRODUCTION

The Framatome AURORA-B CRDA methodology has been used to evaluate the Susquehanna ATRIUM 11 equilibrium fuel cycle (Reference 1). The methodology includes the use of a nodal three-dimensional kinetics solution with both thermal-hydraulic (T-H) and fuel temperature feedback. These models provide more precise localized neutronic and thermal conditions than previous methods to show compliance with regulatory criteria for the BWR CRDA event as presented in the U. S. NRC Standard Review Plan Section 15.4.9 (Reference 2) or that presented in Draft Regulatory Guide DG-1327 (Reference 3). The report summarizes the application of the AURORA-B CRDA methodology (Reference 4) on the Susquehanna ATRIUM 11 equilibrium cycle.

The control rod drop calculations were performed with the AURORA-B CRDA methodology. All startup sequences were evaluated and no fuel rod failures were identified through end of full power. Evaluations of the drops at the licensing basis end of cycle identified potential fuel rod failures in one startup sequence.

2.0 REGULATORY BASIS

The current regulatory basis for the acceptance criteria for the Susquehanna licensing is fuel failure at 170 cal/g and violent expulsion of fuel at 280 cal/g consistent with Reference 5 (SRP 15.4.9, Revision 2). It is anticipated that the final criteria will be similar to that presented in DG-1327 and will be applied in the near future. Therefore this demonstration evaluation using the methodology of Reference 4 is applied assuming the criteria of DG-1327. It is understood that DG-1327 is in the process of being revised for clarification. However, it is not believed that the actual failure criteria for SRA cladding will change.

3.0 INITIAL METHODOLOGY DEMONSTRATION

The initial application of the AURORA-B CRDA methodology involves sensitivity studies and determination of an evaluation boundary. The determination of the evaluation boundary provided in Appendix A is a demonstration of the process discussed in Reference 4 for Susquehanna with ATRIUM 11 fuel.

3.1 *Initial Conditions*

Sensitivity studies are performed **[**

]

3.2 *Group Pull Sequence*

All allowed pull orders are evaluated such that each control rod group, with the exception of groups 5 and 6, is pulled as the second group as indicated in Table 3.1. The third and fourth groups are assumed to be banked. It is assumed that the first and second groups selected for withdrawal are completely withdrawn prior to pulling control rods in the third group. For clarification since both the first and second groups must be out before the third group, both pull sequences A1234 and A2134 have the same starting control rod pattern for the third group. Therefore the sequences A1234 and A2143 also cover sequences A2134 and A1243.

Table 3.1 Group Pull Sequences

	Analyzed Groups for both A and B sequences
1st and 2nd groups	(1,2), (2,1), (3,4), (4,3)
3rd and 4th groups	(3,4), (4,3), (1,2), (2,1)

3.3 *Inoperable Control Rod Positions*

A maximum of 8 inoperable control rods are allowed for this plant with-up to three inoperable per group. To maximize the worth of the drops in the second and third groups, three inoperable control rods are assigned to both the first and second group in each sequence. The assignment of inoperable control rods adhered to the separation criteria on group bases.

Given the uniform core configuration for the equilibrium cycle and that the prior analyses for the sample plant in Reference 4 was limited by drops based on inoperable rod configurations, only drops with inoperable control rod configurations were evaluated. (Note that the sample plant used in Reference 4 was a Susquehanna core.) The selected inoperable control rods for drops in the second group are identified in Figure 3.1. Three inoperable control rods are defined from the first group withdrawn. This results in eight different inoperable control rod configurations for the second group. For the drops in the third group, there are six inoperable control rods in the first and second group. The inoperable control rod configurations for the third group drops are given in Figure 3.2. For each set of inoperable rods, all control rods in the next group are dropped to evaluate the impact of the inoperable rods. Therefore the position of the inoperable control rods was evaluated based on dropping all rods.

[

]

Figure 3.1 Inoperable Control Rods for 2nd Group Pulls

Figure 3.2 Inoperable Control Rods for 3rd Group Pulls

3.4 *Time in Cycle*

[

]

3.5 *Group Critical Position*

The first step is to evaluate the end of group or bank position k-effective values to determine where criticality is anticipated to occur for the given control rod withdrawal sequence. The near critical range, determined per Section 7.4.1 of Reference 4, is given in Table 3.2. The calculated k-effective values at the end of groups 1 through 4 for the A and B sequence withdrawals are given in Table 3.3. **[**

]

Table 3.2 Near Critical Range

[

]

Table 3.3 Group Out Eigenvalues

3.6 *Determination of Static Control Rod Worth*

Based on the results of the group worth, static control rod worths were then determined. From the static rod drops, **[**

] The selected second group rods for transient evaluation are given in Table 3.4 and the third group rods are given in Table 3.5.

Table 3.4 Selected Rods with Inoperable Control Rods 2nd Group

[

]

Table 3.5 Selected Rods with Inoperable Control Rods 3rd and 4th Group

[

]

3.7 *Transient Evaluation*

The evaluation of each rod drop is performed with the AURORA-B system. The initial pre-rod drop state point is established with the MICROBURN-B2 core simulator. The initial conditions used for the transient calculation are identified in Table 3.6.

Table 3.6 Initial Conditions

[ ]

The channel grouping with a **[ ]** is used for this analysis. (Figure 3.3 illustrates the assemblies evaluated for the drop of control rod 22-55.) Once the channel grouping is defined, the power history information is processed to obtain the fuel rod characteristics for use in the RODEX-4 fuel rod mechanical models.

The maximum prompt enthalpy increase for the peak fuel rod and the maximum total enthalpy reported include the application of the uncertainty multiplier of **[ ]** on the enthalpy increase. The prompt enthalpy increase along with total enthalpy for the second group drops is given in Table 3.7. Likewise Table 3.8 contains the prompt enthalpy increase and total enthalpy for third group control rods (the third group bounded the fourth group.) Although there are high worth banked drops, the actual nodal enthalpy increase is small for the BOC banked drops compared to drops later in cycle with a top peaked power shape.

Figure 3.3 Map of Assemblies Evaluated for Drop of Control Rod 22-55

Table 3.7 Maximum Prompt Enthalpy Rise and Total Enthalpy 2nd Group

[ ]

Table 3.8 Maximum Prompt Enthalpy Rise and Total Enthalpy 3rd Group

4.0 EVALUATION AGAINST CLADDING FAILURE CRITERIA

4.1 *High Temperature Cladding Failure*

[

]

Table 4.1 Assemblies with Fuel Rod High Temperature Failures

[

]

Figure 4.1 Total Enthalpy versus High Temperature Cladding Failure Threshold All Drops

Figure 4.2 High Temperature Nominal and High Burnup for Drop EOC_R10

4.2 *PCMI Cladding Failure*

The ATRIUM 11 fuel is clad with stress relief annealed (SRA) Zircaloy-2 cladding. (Framatome uses the term Cold Work Stress Relieved CWSR to refer to SRA material.) Therefore, the SRA low temperature failure threshold is applied.

To establish the minimum failure threshold, the maximum fuel rod nodal hydrogen at end of cycle was tabulated for each assembly using the hydrogen model of Reference 8. **[**

]

[ ]

Figure 4.3 Minimum Failure Threshold Based on EOC Hydrogen Content

Since 150 Δcal/g is the maximum of the failure threshold curve, [ ] The rod drops are evaluated with assumed inoperable control rods. [ ]

[ ] There were no failures before or at end of full power. The assemblies with fuel rod failures for drops at EOC are given in Table 4.2.

Table 4.2 Assemblies with Fuel Rod PCMI Failures

[ ]

[ ]

Figure 4.4 PCMI Cladding Failure Results

4.3 *Molten Fuel Cladding Failure Threshold*

[ ]

5.0 RADIOLOGICAL CONSEQUENCES

The dose consequences for the CRDA determined for Susquehanna are summarized in the SSES UFSAR. The licensing basis dose evaluation based on ATRIUM 10 fuel determined that 2000 fuel rods could fail for SSES. Since the actual number of ATRIUM 11 allowed fuel rod failures has not been determined at this time it is assumed that the allowed number of failures will be similar to that of the ATRIUM 10. Therefore, demonstrating that there is significantly less than 2000 fuel rod failures will confirm that the radiological consequences are bounded by those given in the SSES UFSAR. Although two control rod drops indicated failures in Table 4.2, only the rod drop B_R010_EOC is evaluated for dose consequences due to the higher enthalpy. The high burnup drop is not evaluated for this demonstration in that the results would be very similar to the nominal burnup case.

Evaluation of dose consequences for fuel rod failures

Since fuel rod failures had been identified, revised release fractions or total release fraction (TOTR) are determined using the Licensing Basis Release Fractions (LBRF) from RG 1.183 as the steady state release fractions (SSRF) with the transient fission gas release (TFGR) as described in DG-1327. A ratio of the new TOTR to the LBRF used in the original licensing basis is then generated following the method provided in ANP-10333PA.

The transient release terms, from Reference DG-1327, expressed as a fraction are:

Peak Pellet BU < 50 GWd/MTU:

$$TFGR = \frac{[(0.26 * \Delta H - 13]}{100} \geq 0$$

Peak Pellet BU ≥ 50 GWd/MTU:

$$TFGR = \frac{[(0.26 * \Delta H - 5]}{100} \geq 0$$

The total fuel rod release fraction TOTR is dependent on the nuclide group and the enthalpy dependent TFGR average over the 25 nodes of fuel for a full length fuel rod. (If the failure were in a shorter fuel rod, the number of axial nodes would be decreased accordingly.)

Burnup < 50:

$$TOTR = SSRF + \frac{\sum_k \frac{[(0.26 * \Delta H) - 13]}{25}}{100} * GMUL$$

Burnup ≥ 50:

$$TOTR = SSRF + \frac{\sum_k \frac{[(0.26 * \Delta H) - 5]}{25} *}{100} * GMUL$$

Where,

- ΔH fuel enthalpy increase (cal/g)
- SSRF is the steady state release fraction
- Three multipliers (GMUL) are established in DG-1327 to be applied to the above TFGR term:

Group	GMUL	Applied to
Stable long lived isotopes (e.g., Kr-85)	1.0	Kr-85
Cs-134 and Cs-137	1.414	Alkali Metals
Short-lived radioactive isotopes (i.e., I, Xe and Kr noble gases except Kr-85)	0.333	Iodines, nobles, halogens

As noted above, the LBRF are utilized for Susquehanna as the SSRF for the respective groups.

For this analysis of the ATRIUM 11 fuel, a maximum of 15 fuel rods for any control rod drop case exceed one or more failure criteria. Based on the enthalpy increase, the enthalpy dependent release terms were determined for the fuel rods. The transient fission gas release fractions are provided in Table 5.1 based on nodal values of the peak fuel rod enthalpy increase. (The nodal increase in the peak fuel rod enthalpy is assumed in the determination of the transient gas release for all fuel rods that failed in a given assembly. Since the 25 node average was similar between Assembly 25A002 and 25A003, the more limiting value of 0.049 was used for both assemblies.) The total release fraction and the ratio to the licensing bases release fraction are provided in Table 5.2. The actual number of fuel rod failures is provided in Table 5.3. **[**

] This is significantly below 2000 control rods; therefore this event remains within the current evaluated dose consequences for Susquehanna.

Table 5.1 Transient Fission Gas Release Fractions

[ ]

[ ]

Table 5.2 Total Fission Gas Release Fractions

[ ]

Table 5.3 Fuel Rod Failures

[ ]

[ ]
[ ]

6.0 SYSTEM PRESSURE AND CPR

The impact of the CRDA on system pressure was addressed in Reference 4 and does not cause stresses to exceed Emergency Condition (Service Level C), as defined in Section III of the ASME Boiler and Pressure Vessel code. This generic evaluation on the impact of CRDA on system pressure remains applicable for Susquehanna.

The CPR response was evaluated in Section 7.7 of Reference 4 and resulted in a conclusion that the CRDA in the power range is **[**

]

7.0 CORE COOLABILTY

Two criteria are identified in Reference 3 for allowable limits with respect to core coolability.

- Peak radial average enthalpy <230cal/g
- The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions

[

]

Table 7.1 Peak Radial Average Fuel Enthalpy

[

]

8.0 LIMITATIONS AND CONDITIONS

The SER for the Reference methodology included a number of limitations and conditions. Some of the conditions are from the base AURORA-B AOO methodology (Reference 6) and additions specific to the CRDA are included. The numbering of the limitations and conditions below is consistent with that found in the AURORA-B CRDA SER.

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

This condition is met for application of the AURORA-B CRDA methodology to Susquehanna.

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

This condition is met for Susquehanna since it is a BWR/4.

20. The implementation of any new methodology within the AURORA-B EM (i.e., replacement of an existing CCD) is not acceptable unless the AURORA-B EM with the new methodology incorporated into it has received NRC review and approval. An existing NRC-approved methodology cannot be implemented within the AURORA-B EM without NRC review of the updated EM. *(This is a revised version of Condition 20 of the SE for the base AURORA-B TR, rewritten to be specific to the CRDA application. It remains applicable to CRDA analyses for BWRs/2-6.)*

The evaluation model will be implemented for Susquehanna as described in the base AURORA-B and AURORA-B CRDA Topical Reports. No CCD as described in the TR are replaced and therefore the intent of this condition is met.

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

approval. *(This is Condition 21 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

[

]

22. As discussed in Section 3.3.1.5 and Section 4.0 of Reference 6 (the SE for the base AURORA-B TR), the SPCB and ACE CPR correlations for the ATRIUM-10 and ATRIUM-10XM fuels, respectively, are approved for use with the AURORA-B EM. Other CPR correlations (existing and new) that would be used with the AURORA-B EM must be reviewed and approved by the NRC or must be developed with an NRC-approved approach such as that described in EMF-2245(P)(A), Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel". Furthermore, if transient thermal-hydraulic simulations are performed in the process of applying AREVA CPR correlations to co-resident fuel, these calculations should use the AURORA-B methodology. *(This is Condition 22 of the SE for the base AURORA-B TR. It remains applicable to at-power CRDA analyses for BWRs/2-6.)*

This condition is met within ANP-10333PA for at power evaluations. The ACE ATRIUM 11 Correlation has been reviewed and approved by the NRC (Reference 7).

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

NRC review and approval prior to their implementation in AURORA-B. *(This is Condition 23 of the SE for the base AURORA-B TR. It remains applicable to CRDA analyses for BWRs/2-6.)*

This condition is met as ATRIUM 11 does exhibit the structural similarities described in the restriction.

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

This condition is met through the use of the Framatome software development procedures.

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

No confirmation is required for this condition.

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

This condition is met through the use of the Framatome software development procedures which include 10CFR50.59 licensing considerations.

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

[ ]

Therefore, this condition is met.

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

The licensee has confirmed that this condition is met for the control rods at Susquehanna.

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

[ ]
Susquehanna with ATRIUM 11 fuel for determining the total enthalpy.

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

The inoperable control blade patterns were evaluated for all rods [ ] For the Susquehanna core, localizing the inoperable rods to one area of the core increases the rod worth of the dropped rod in another part of the core.

31. The evaluation boundary curve used to determine candidate control rods for further evaluation based on their static rod worths must be verified to bound the following local characteristics of the fuel being evaluated: design pin peaking factors, fuel assembly design, location in or adjacent to the outermost ring of control rods, and average burnup for the 16 fuel assemblies surrounding the rod of interest.

This condition is addressed in Appendix A for this demonstration analysis.

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

The highest control rod worth did result in a total enthalpy which exceeded the minimum high temperature failure threshold. Therefore, additional control rods were evaluated to address this condition (see Section 4.1).

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

The ATRIUM 11 product line requires an evaluation of the gap sensitivity study. The sensitivity studies were performed with a bounding value for the uncertainty range of **[** **]**. The resulting increase in peak total enthalpy **[** **]**

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

The uncertainty of **[** **]** percent is used in this evaluation.

9.0 REFERENCES

1. ANP-3727P, Revision 0, Susquehanna ATRIUM 11 Equilibrium Cycle Fuel Cycle Design, Framatome, October 2018.

2. NUREG-0800, Section 15.4.9, Revision 3, "SPECTRUM OF ROD DROP ACCIDENTS (BWR)." *Standard Review Plan: LWR Edition,* US NRC: Washington, DC. March 2007.

3. DG-1327, Pressurized Water Reactor Control Rod Ejection and Boiling Water reactor Control Rod Drop Accidents, November 2016 (NRC ADAMS ML16124A200).

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

5. NUREG-0800, Section 15.4.9, Revision 2, "SPECTRUM OF ROD DROP ACCIDENTS (BWR)." *Standard Review Plan: LWR Edition,* US NRC: Washington, DC. July 1981.

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

7. ANP-10335P-A, Revision 0, ACE/ATRIUM 11 Critical Power Correlation Topical Report, Framatome, May 2018.

8. BAW-10247PA Revision 0 Supplement 1P-A Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Qualification of RODEX4 for Recrystallized Zircaloy-2 Cladding, AREVA, April 2017.

9. ANP-3753P Revision 0, Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel Report, Framatome Inc., May 2019.

Appendix A Evaluation Threshold determination

Limitation and Condition 31 states:

31. The evaluation boundary curve used to determine candidate control rods for further evaluation based on their static rod worths must be verified to bound the following local characteristics of the fuel being evaluated: design pin peaking factors, fuel assembly design, location in or adjacent to the outermost ring of control rods, and average burnup for the 16 fuel assemblies surrounding the rod of interest.

The process to generate an evaluation threshold is demonstrated based upon the process described in the response to RAI-5 in ANP-10333Q1P (included in Reference 4). The peak fuel rod enthalpy rise was elevated using a multiplication factor of **[ ]** to double the uncertainty. The elevated enthalpy rise values were then tabulated against the static control rod worth.

[ ]

Figure A.1 Establishing Evaluation Boundary

[ ]

Figure A.2 Evaluation boundary for ATRIUM 11 Core

Using the evaluation boundary on this cycle, for interior assemblies, **[ ]**

For peripheral control rods **[ ]**

The local characteristics of fuel used to establish the evaluation boundary with respect to design fuel rod peaking factors, fuel assembly design, core location, and the average burnup of the 16 assemblies around the dropped rod have been are provided in Table A.1. This table is for use with future core licensing to confirm the applicability of the evaluation boundary curve.

Table A.1 Example Evaluation Boundary Characteristics

[ ]

[ ]

Enclosure 16c of PLA-7783

Framatome Affidavit

Affidavit for ANP-3771P, Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3771P Revision 0, "Susquehanna ATRIUM 11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology," dated May 2019 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 31st

day of May, 2019.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020

HAILEY M SIEKAWITCH
Commission Expires
NOTARY PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON