Enclosure 1 Contains 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

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

10 CFR 50.90

SUSQUEHANNA STEAM ELECTRIC STATION
SUBMITTAL OF REVISED TOPICAL REPORT TO SUPPORT LICENSE AMENDMENT REQUESTING APPLICATION OF ADVANCED FRAMATOME METHODOLOGIES
PLA-7847

Docket No. 50-387 and 50-388

References:

1) Susquehanna letter to NRC, "Proposed Amendment to Licenses NPF-14 and NPF-22: Application of Advanced Framatome Methodologies and TSTF-535 (PLA-7783)," dated July 15, 2019 (ADAMS Accession No. ML19196A270)

2) Susquehanna letter to NRC, "Thirty-Day Response to Request for Additional Information Regarding Proposed License Amendment Requesting Application of Advanced Framatome Methodologies (PLA-7841)," dated February 6, 2020 (ADAMS Accession No. ML20037A098)

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), submitted, in Reference 1, 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 amendment would revise 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, revise the low pressure safety limit in TS 2.1.1.1 and TS 2.1.1.2, and remove the neutronic methods penalties on Oscillation Power Range Monitor amplitude setpoint and the pin power distribution uncertainty and bundle power correlation coefficient.

The purpose of this letter is to provide a revision to a Topical Report provided in Reference 1. Enclosure 8a to Reference 1 provided Framatome Topical Report ANP-3753P, Revision 0, "Applicability of Framatome BWR [Boiling Water Reactor] Methods to Susquehanna with

ATRIUM 11 Fuel." Enclosure 8b to Reference 1 provided the non-proprietary version of Enclosure 8a. Subsequent to submittal of Reference 1, Framatome revised ANP-3753P, specifically with respect to the time step used in the evaluation of Anticipated Operational Occurrences (AOOs) utilizing the AURORA-B AOO methodology. This resulted in creation of a new Section 6.6, and two new acronyms being added to the nomenclature list on page v. No other changes were made in Revision 2 to ANP-3753P.

Enclosure 1 to this letter provides ANP-3753P, Revision 2. Information provided in Enclosure 1 is considered proprietary to Framatome. The proprietary information has been denoted therein by brackets. As owners of the proprietary information, Framatome has executed an affidavit for the document which identifies the information as proprietary, is customarily held in confidence, and should be withheld from public disclosure in accordance with 10 CFR 2.390. Enclosure 2 provides a non-proprietary version of Enclosure 1. The Framatome affidavit is included as Enclosure 3. The documents provided in Enclosures 1 and 2 supersede the documents provided in Enclosures 8a and 8b to Reference 1 in their entirety.

Susquehanna has reviewed the information supporting a finding of No Significant Hazards Consideration and the Environmental Consideration provided to the NRC in Reference 1 and determined the information provided herein does not impact the original conclusions in Reference 1. Further, the revisions to ANP-3753P do not impact Susquehanna's response to the NRC's Request for Additional Information in Reference 2.

There are no new or revised regulatory commitments contained in this submittal.

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: 3/5/20

K. Cimorelli

Enclosures:

1. Framatome Topical Report ANP-3753P, Revision 2, “Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel” **[Proprietary Information – Withhold from Public Disclosure in accordance with 10 CFR 2.390]**
2. Framatome Topical Report ANP-3753NP, Revision 2, “Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel” (Non-Proprietary Version)
3. Framatome Affidavit for ANP-3753P, Revision 2, “Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel”

Copy: NRC Region I
Ms. L. Micewski, NRC Sr. Resident Inspector
Ms. S. Goetz, NRC Project Manager
Mr. M. Shields, PA DEP/BRP (w/out Enclosure 1)

Enclosure 2 of PLA-7847

Framatome Topical Report
ANP-3753NP, Revision 2

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

(Non-Proprietary Version)

framatome

Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel

ANP-3753NP
Revision 2

March 2020

Nature of Changes

(The following changes are indicated with change bars instead of highlighting throughout the document.)

Item	Section(s) or Page(s)	Description and Justification*
1	Nomenclature	Included additional acronyms
2	Section 6.6	Added Section 6.6

* The changes reflected below were retained from Revision 1 of this report in order to identify the changes with change bars instead of yellow highlighting.

Contents

List of Tables

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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
FWCF	feedwater controller failure
HPCI	High Pressure Coolant Injection
IHPCIS	Inadvertant High Pressure Coolant Injection System
KATHY	Karlstein thermal hydraulic test facility
LHGR	linear heat generation rate
LOFH	loss of feedwater heating
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

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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

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

† [

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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).

[

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[

]. 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 **[**

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[

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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 **[**

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6.6 *AURORA-B AOO Time Step Size*

Section 6.8.2 of ANP-10300P-A, Reference 1, provides a discussion of a time step size sensitivity study using the AURORA-B AOO methodology. The conclusion of this section states:

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

This conclusion was based off of a set of sensitivity studies which **[**

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[

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7.0 STABILITY

Stability analyses 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

[ ]

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* 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 [ ]

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7.4.3 [ ]

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7.4.4 [ ]

[

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* [

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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.

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Table 8-1 [

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* 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.

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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 3 of PLA-7847

Framatome Affidavit for ANP-3753P, Revision 2, "Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel"

AFFIDAVIT

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 2, "Applicability of Framatome BWR Methods to Susquehanna with ATRIUM 11 Fuel," dated March 2020 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(a), 6(b), 6(c), 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.

Alan Meginnis

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

SUBSCRIBED before me this 5th day of March, 2020.

Katherine Kerr
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/12/2022

KATHERINE KERR
MY COMM. EXP. 09/12/2022
NOTARY PUBLIC
COMM. NO. 203028
STATE OF WASHINGTON