ANP-2989(NP), *Brunswick Unit 1 Thermal-Hydraulic Design Report for ATRIUM™ 10XM Fuel Assemblies*, Revision 0

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Brunswick Unit 1 Thermal-Hydraulic Design Report for ATRIUM™ 10XM Fuel Assemblies

May 2011

AREVA NP Inc.

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial issue.

Contents

Tables

Figures

Nomenclature

AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BRK1-18	Brunswick Unit 1 Cycle 18
BRK1-19	Brunswick Unit 1 Cycle 19
BWR	boiling water reactor
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
IFG	Improved FUELGUARD™
LOCA	loss-of-coolant accident
LTP	lower tie plate
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
MFG	Modified FUELGUARD
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 Brunswick Unit 1 thermal-hydraulic analyses are presented to demonstrate that AREVA NP ATRIUM™ 10XM* fuel is hydraulically compatible with the previously loaded ATRIUM-10 fuel with improved FUELGUARD™ lower tie plates (designated in this report as IFG) and ATRIUM-10 fuel with modified FUELGUARD lower tie plates (designated in this report as MFG). This report also provides the hydraulic characterization of the ATRIUM 10XM and the coresident ATRIUM-10 IFG and ATRIUM-10 MFG designs for Brunswick Unit 1.

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).

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* ATRIUM and FUELGUARD are trademarks of AREVA NP.

2.0 Summary and Conclusions

ATRIUM 10XM fuel assemblies have been determined to be hydraulically compatible with the coresident ATRIUM-10 IFG and ATRIUM-10 MFG fuel designs in the Brunswick Unit 1 reactor 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 Table 3.4 to Table 3.8.

The ATRIUM 10XM fuel design is geometrically different from the coresident ATRIUM-10 IFG and ATRIUM-10 MFG designs, but the designs are hydraulically compatible. [

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Core bypass flow (defined as leakage flow through the lower tie plate (LTP) flow holes, channel seal, core support plate, and LTP-fuel support interface) is not adversely affected by the introduction of the ATRIUM 10XM fuel design. Analyses at rated conditions show core bypass flow varying between [ ] of rated flow for transition core configurations ranging from the BRK1-18 core loading with ATRIUM-10 IFG, ATRIUM-10 MFG, and GE14 fuel to a full ATRIUM 10XM core, respectively.

Analyses demonstrate the thermal-hydraulic design and compatibility criteria discussed in Section 3.0 are satisfied for the Brunswick Unit 1 transition cores consisting of ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG 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 10XM fuel design are 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 AREVA 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 evaluation 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 approved power and flow operating region. The stability performance of new fuel designs will be equivalent to, or better than, existing (approved) AREVA 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 reports.
- **Control rod drop accident (CRDA) analysis.** The deposited enthalpy must be less than 280 cal/gm for fuel coolability. 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 was addressed in Reference 3.

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

3.1 *Hydraulic Characterization*

Basic geometric parameters for the ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG fuel designs are summarized in Table 3.2. Component loss coefficients for the ATRIUM 10XM 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 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG are based on tests performed at AREVA's Portable Hydraulic Test Facility. [

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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 AREVA thermal-hydraulic methodology for BWRs. The methodology and constitutive relationships used by AREVA for the calculation of pressure drop in BWR fuel assemblies are presented in

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Reference 4 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 5. The NRC reviewed the information provided in Reference 6 regarding inclusion of water rod models in XCOBRA and accepted the inclusion in Reference 7.

Hydraulic compatibility, as it relates to the relative performance of the ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG fuel designs, has been evaluated. Detailed analyses were performed for the Brunswick Unit 1 Cycle 18 and full core ATRIUM 10XM configurations. Analyses for mixed cores with ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG fuel were also performed to demonstrate that the thermal-hydraulic design criteria are satisfied for transition core configurations.

The hydraulic compatibility analysis is based on [

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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 60% power/45% 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 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG results at rated power are within the range considered compatible, as expected. Similar agreement occurs at lower power levels.

As shown in Table 3.5, [

] Table 3.6 shows that [

] Differences in assembly flow between the ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG 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 transition from the BRK1-18 core loading to a full core of ATRIUM 10XM, the ATRIUM 10XM 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 AREVA'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 AREVA to determine the margin to thermal limits for BWRs.

CPR values for ATRIUM 10XM are calculated with the ACE/ATRIUM 10XM critical power correlation (Reference 8) while the CPR values for the ATRIUM-10 IFG and ATRIUM-10 MFG fuel are calculated with the SPCB critical power correlation (Reference 9). Assembly design features are incorporated in the CPR calculation through the K-factor term in the ACE correlation and the F-eff term for the SPCB correlation. The K-factors and F-effs 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 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG assemblies with radial peaking factors (RPFs) between [ ] The Reference 10 discussion identifies the concern that the K-factors used in the ACE correlation for the analysis of ATRIUM

10XM [

] This finding does not impact the validity of the results of this evaluation. [

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Table 3.5 and Table 3.6 show CPR results of the ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG fuels. 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 10XM fuel will not cause thermal margin problems for the coresident fuel designs.

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. Rod bow has been measured during post-irradiation examinations of BWR fuel fabricated by AREVA.

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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 changes from [ ] of rated core flow during the transition from the BRK1-18 core loading to a full ATRIUM 10XM core (middle-peaked power shape). In summary, adequate bypass flow will be available with the introduction of the ATRIUM 10XM fuel design 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 (NRC-approved) AREVA 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 11). The study shows that the ATRIUM 10XM fuel design has decay ratios equivalent to or better than other approved AREVA 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 10XM 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 10XM demonstrated to be compatible with ATRIUM-10 IFG and ATRIUM-10 MFG fuel. [ ]
3.3	Thermal margin performance	Fuel design shall be within the limits of applicability of an approved CHF correlation.	ACE/ATRIUM 10XM critical power correlation is applied to the ATRIUM 10XM fuel. SPCB critical power correlation is applied to the ATRIUM-10 IFG and ATRIUM-10 MFG 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.
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 10XM 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) AREVA fuel designs.	ATRIUM 10XM channel and core decay ratios have been demonstrated to be equivalent to or better than other approved AREVA 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.	Approved Appendix K LOCA model. Plant- and fuel-specific analysis with cycle-specific verifications.
	CRDA analysis	< 280 cal/gm for coolability.	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.	Criterion met per Reference 3.

Table 3.2 Comparative Description for Brunswick Unit 1 ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG Fuel Types

Fuel Parameter	ATRIUM 10XM	ATRIUM-10 IFG	ATRIUM-10 MFG
Number of fuel rods			
Full-length fuel rods	79	83	83
PLFRs	12	8	8
Fuel clad OD, in	0.4047	0.3957	0.3957
Number of spacers	9	8	8
Active fuel length, ft			
Full-length fuel rods	12.500	12.454	12.454
PLFRs	6.25	7.5	7.5
Hydraulic resistance characteristics	Table 3.3	Table 3.3	Table 3.3
Number of water rods	1	1	1
Water rod OD, in	1.378*	1.378*	1.378*

* Square water channel outer width.

Table 3.3 Hydraulic Characterization Comparison for Brunswick Unit 1 ATRIUM 10XM, ATRIUM-10 IFG, and ATRIUM-10 MFG Fuel

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Table 3.4 Brunswick Unit 1
Thermal-Hydraulic Design Conditions

Reactor Conditions	100%P / 100%F	60%P / 45%F
Core power level, MWt	2923.0	1753.8
Core exit pressure, psia	1058.3	993.1
Core inlet enthalpy, Btu/lbm	528.3	505.2
Total core coolant flow, Mlbm/hr	77.0	34.7
Axial power shape	Middle-peaked (Figure 3.1)	Middle-peaked (Figure 3.1)

	Number of Assemblies Central Region	 Peripheral Region
BRK1-18 Core Loading		
[		]
[		]
[		]
First Transition Core Loading		
[		]
[		]
[		]
Second Transition Core Loading		
[		]
[		]

Table 3.5 Brunswick Unit 1
First Transition Core Thermal-Hydraulic Results at
Rated Conditions (100%P / 100%F)

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Table 3.6 Brunswick Unit 1
First Transition Core Thermal-Hydraulic Results at
Off-Rated Conditions (60%P / 45%F)

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Table 3.7 Brunswick Unit 1 Thermal-Hydraulic Results at Rated Conditions (100%P / 100%F) for Transition to ATRIUM 10XM Fuel

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Table 3.8 Brunswick Unit 1 Thermal-Hydraulic Results at Off-Rated Conditions (60%P / 45%F) for Transition to ATRIUM 10XM Fuel

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Figure 3.1 Axial Power Shapes

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

[

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**Figure 3.3 First Transition Core:
Hydraulic Demand Curves
60%P / 45%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. ANP-2948P Revision 0, *Mechanical Design Report for Brunswick ATRIUM 10XM Fuel Assemblies*, AREVA NP, October 2010.

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

5. 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.

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

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

8. ANP-10298PA Revision 0, *ACE/ATRIUM 10XM Critical Power Correlation*, AREVA NP, March 2010.

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

10. Condition Report 2011-2274, [ ] AREVA NP, March 2011.

11. 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.

AREVA Affidavit Regarding Withholding
ANP-3061(P), *Brunswick Unit 1 Cycle 19 Reload Safety Analysis*, Revision 0,
from Public Disclosure

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

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

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

3. I am familiar with the AREVA NP information contained in the report ANP-3061(P) Revision 0, entitled, "Brunswick Unit 1 Cycle 19 Reload Safety Analysis," dated December 2011 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP 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 AREVA NP 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

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(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 AREVA NP.

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8. AREVA NP 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 20th day of December, 2011.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/10/12

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