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*Brunswick Unit 1 Thermal-Hydraulic Design Report for ATRIUM™-10 Fuel Assemblies*,
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ANP-2646(NP)
Revision 0

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

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## Nature of Changes

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

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## 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 Brunswick Unit 1 thermal-hydraulic analyses are presented to demonstrate that AREVA NP* ATRIUM™-10† fuel is hydraulically compatible with coresident GE14 fuel. This report also provides the hydraulic characterization of the ATRIUM-10 and coresident GE14 fuel 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).

* AREVA NP Inc. is an AREVA and Siemens company.
† ATRIUM is a trademark of AREVA NP.

## 2.0 Summary and Conclusions

ATRIUM-10 fuel assemblies have been determined to be hydraulically compatible with GE14 fuel coresident in the 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 Tables 3.4 to 3.8.

The ATRIUM-10 fuel design is geometrically different from the coresident GE14 design, but hydraulically the two designs are compatible. [

]

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-10 fuel design. Analyses at rated conditions show core bypass flow varying between [ ] of rated flow for transition core configurations ranging from a full GE14 fuel core to a full ATRIUM-10 core, respectively.

Analyses demonstrate the thermal-hydraulic design and compatibility criteria discussed in Section 3.0 are satisfied for the Brunswick Unit 1 transition core consisting of ATRIUM-10 and GE14 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-10 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 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 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 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 ATRIUM-10 and GE14 fuel designs are summarized in Table 3.2. Component loss coefficients for the ATRIUM-10 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, and UTP losses for ATRIUM-10 are based on flow tests. The local losses for the Brunswick ATRIUM-10 modified high-pressure drop FUELGUARD™* LTP are based on pressure drop tests performed at AREVA's Portable Hydraulic Test Facility. **[**

**]** The local component (LTP, spacer, and UTP) loss coefficients for the GE14 fuel are based on flow test results.

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.

* ULTRAFLOW and FUELGUARD are trademarks of AREVA NP.

## 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 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-10 and GE14 fuel designs, has been evaluated. Detailed analyses were performed for full core GE14 and full core ATRIUM-10 configurations. Analyses for mixed ATRIUM-10 and GE14 cores were also performed to demonstrate that the thermal-hydraulic design criteria are satisfied for transition core configurations.

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 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 ATRIUM-10 and GE14 fuel rated power results are within the range considered compatible, as expected

based on previous transitions involving GE14 fuel. 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-10 and GE14 fuel designs as a function of assembly power level are shown in Figure 3.2 and Figure 3.3.

[

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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 GE14 to ATRIUM-10, the ATRIUM-10 design is considered hydraulically compatible with the GE14 design 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-10 and GE14 fuel are calculated with the SPCB critical power correlation (Reference 7). The NRC-approved methodology to demonstrate the acceptability of using the SPCB correlation for computing GE14 fuel CPR is presented in Reference 8. Assembly design features are incorporated in the CPR calculation through the F-eff term. The F-eff is 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-10 and GE14 assemblies with radial peaking factors (RPFs) between [

] Table 3.5 and Table 3.6 show CPR results of the ATRIUM-10 and GE14 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-10 fuel will not cause thermal margin problems for the coresident GE14 fuel.

## 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.

[

]

## 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 a full GE14 core to a full ATRIUM-10 core (middle-peaked power shape). [

] In summary, adequate bypass flow will be available with the introduction of the ATRIUM-10 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 9). The study

shows that the ATRIUM-10 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-10 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.<br><br>ATRIUM-10 demonstrated to be compatible with GE14.<br><br>[<br><br>] |
| 3.3 | Thermal margin performance | Fuel design shall be within the limits of applicability of an approved CHF correlation. | SPCB is applied to both the ATRIUM-10 and GE14 fuel. |
| | | < 0.1% of rods in boiling transition. | Verified on cycle-specific basis for Chapter 15 analyses. |
| | Fuel centerline temperature | No centerline melting. | Refer to the mechanical design report. |
| 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.<br><br>Analysis results demonstrate that adequate bypass flow is provided. |

**Table 3.1 Design Evaluation of Thermal and Hydraulic Criteria for the ATRIUM-10 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-10 channel and core decay ratios have been demonstrated to be equivalent to or better than other approved AREVA fuel designs.<br><br>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.<br><br>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. | Refer to the mechanical design report. |

**Table 3.2 Comparative Description of Brunswick Unit 1 ATRIUM-10 and GE14 Fuel**

| Fuel Parameter | ATRIUM-10 | GE14 |
|---|---|---|
| Number of fuel rods | | |
| Full-length fuel rods | 83 | 78 |
| PLFRs | 8 | 14 |
| Fuel clad OD, in | 0.3957 | 0.404 |
| Number of spacers | 8 | 8 |
| Active fuel length, ft | | |
| Full-length fuel rods | 12.454 | 12.500 |
| PLFRs | 7.5 | 7.0 |
| Hydraulic resistance characteristics | Table 3.3 | Table 3.3 |
| Number of water rods | 1 | 2 |
| Water rod OD, in | 1.378* | 0.980 |

* Square water channel outer width.

**Table 3.3 Hydraulic Characterization Comparison Between Brunswick Unit 1 ATRIUM-10 and GE14 Fuel Assemblies**

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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 | 1054.5 | 986.0 |
| Core inlet enthalpy, Btu/lbm | 528.3 | 504.3 |
| Total core coolant flow, Mlbm/hr | 77 | 34.65 |
| Axial power shape | Middle-peaked (Figure 3.1) | Middle-peaked (Figure 3.1) |

| | Number of Assemblies | |
|---|---|---|
| | Central Region | Peripheral Region |
| 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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[

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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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[

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

[

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[

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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. 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. EMF-2209(P)(A) Revision 2, *SPCB Critical Power Correlation*, Framatome ANP, September 2003.

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

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.