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

April 8, 2011

Mr. Michael J. Annacone, Vice President
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING ADDITION OF ANALYTICAL METHODOLOGY TOPICAL REPORT TO TECHNICAL SPECIFICATION 5.6.5 (TAC NOS. ME3858 AND ME3859)

Dear Mr. Annacone:

The U.S. Nuclear Regulatory Commission (Commission, NRC) has issued the enclosed Amendment No. 256 to Renewed Facility Operating License No. DPR-71 and Amendment No. 284 to Renewed Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant (BSEP), Units 1 and 2, respectively. The amendments are in response to your application dated April 29, 2010, as supplemented by letters dated June 9, July 22, July 29, September 29, October 12, November 9, November 18, and December 16, 2010; March 16, and April 6, 2011. The amendments revise the BSEP, Units 1 and 2 Technical Specification (TS) 5.6.5.b by adding AREVA's topical report, BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, April 2008, to the list of analytical methods that have been reviewed and approved by the NRC for determining core operating limits. The amendments changed the BSEP, Units 1 and 2 TSs to support transition to ATRIUM 10XM fuel and associated core design methodologies.

The NRC staff reviewed the licensee's application for amendment and the licensee's submitted supplements. In addition, the NRC staff reviewed the licensee's calculations and documents supporting the proposed amendments during an audit at the AREVA's offices located in Bethesda, Maryland. The NRC staff's review of the licensee's submittals identified one issue with the licensee's analytical methods that resulted in incorporation of one regulatory commitment as described in Section 4 of the enclosed safety evaluation (SE).

The NRC staff has determined that the enclosed SE contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390, "Public inspections, exemptions, request for withholding." Accordingly, the NRC staff has prepared a redacted, nonproprietary version. However, we will delay placing the nonproprietary SE in the public document room for a period of 10 working days from the date of this letter to provide you with the opportunity to comment on any proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10 working days, the nonproprietary SE will be made publicly available. Copies of the proprietary and nonproprietary versions of the SE are enclosed.

Document transmitted herewith contains sensitive unclassified information. When separated from Enclosure 4, this document is decontrolled.

A notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Farideh E. Saba

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 256 to
   License No. DPR-71
2. Amendment No. 284 to
   License No. DPR-62
3. Safety Evaluation (Nonproprietary Information)
4. Safety Evaluation (Proprietary Information)

cc w/enclosures 1, 2, 3, and 4: Addressee
cc w/enclosures 1, 2, and 3: Distribution via ListServ

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 256
Renewed License No. DPR-71

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), April 29, 2010, as supplemented by letters dated June 9, July 22, July 29, September 29, October 12, November 9, November 18, and December 16, 2010; March 16, and April 6, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 256 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to startup from the 2012 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachments:
Changes to the Operating License, and
Technical Specifications

Date of Issuance: April 8, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 256

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following page of Renewed Operating License DPR-71 with the attached revise page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
4	4

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

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5.0-22	5.0-22

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 256 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Renewed Facility Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 203.

(a) Effective June 30, 1982, the surveillance requirements listed below need not be completed until July 15, 1982. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2 shall apply.

Specification 4.3.3.1, Table 4.3.3-1, Items 5.a and 5.b

(b) Effective July 1, 1982, through July 8, 1982, Action statement "a" of Technical Specification 3.8.1.1 shall read as follows:

ACTION:

a. With either one offsite circuit or one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.A. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within two hours and at least once per 12 hours thereafter; restore at least two offsite circuits and four diesel generators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(3) Deleted by Amendment No. 206.

D. The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Physical Security Plan, Revision 2," and "Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006, and "Guard Training and Qualification Plan, Revision 0," submitted by letter dated September 30, 2004.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

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.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 284
Renewed License No. DPR-62

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Carolina Power & Light Company (the licensee), April 29, 2010, as supplemented by letters dated June 9, July 22, July 29, September 29, October 12, November 9, November 18, and December 16, 2010; March 16, and April 6, 2011, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

> The Technical Specifications contained in Appendix A, as revised through Amendment No. 284 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to startup from the 2011 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachments:
Changes to the Operating License, and
Technical Specifications

Date of Issuance: April 8, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 284

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following page of Renewed Operating License DPR-62 with the attached revise page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
3	3

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
5.0-22	5.0-22

as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

(5) Pursuant to the Act and 10 CFR Parts 30 and 70 to posses, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and H. B. Robinson Steam Electric Plant, Unit No. 2

(6) Carolina Power & Light Company shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the Safety Evaluation Report dated November 22, 1977, as supplemented April 1979, June 11, 1980, December 30, 1986, December 6, 1989, July 28, 1993, and February 10, 1994 respectively, subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 284 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

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.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 256 AND 284

TO RENEWED FACILITY OPERATING LICENSES NOS. DPR-71 AND DPR-62

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By application dated April 29, 2010 (Reference 1), as supplemented by letters dated June 9, July 22, July 29, September 29, October 12, November 9, November 18, and December 16, 2010; March 16, and April 6, 2011, (References 2 through 11 and Reference 41) Carolina Power & Light Company (CP&L, the licensee) requested license amendments to revise Technical Specifications (TSs) for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The proposed license amendments will revise BSEP TS 5.6.5.b by adding AREVA's topical report, BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, April 2008 (Reference 12), to the list of analytical methods that have been reviewed and approved by the Nuclear Regulatory Commission (NRC) for determining core operating limits. The proposed amendments would change the BSEP TSs to support transition to ATRIUM 10XM fuel and associated core design methodologies.

BSEP, Units 1 and 2 are General Electric (GE) boiling water reactors (BWRs) BWR/4 design. Both BSEP units are currently operating at 120 percent of originally licensed thermal power at extended power uprate and maximum extended load line limit analysis conditions. Each unit's reactor core is composed of 560 fuel assemblies. In each cycle, approximately 40 percent of the irradiated fuel assemblies are replaced with new fuel. On March 27, 2008, the NRC issued Amendments Nos. 246 and 274 for transition from GE fuel to ATRIUM-10 fuel for BSEP, Units 1 and 2, respectively (Reference 13). The first transition from GE fuel to ATRIUM-10 fuel occurred in spring 2008 and spring 2009, when the licensee loaded 248 and 238 fresh ATRIUM 10 fuel assemblies in BSEP, Units 1 and 2, respectively.

The licensee plans to implement this amendment, loading the ATRIUM 10XM fuel design in the BSEP, Unit 2 during the spring 2011 refueling outage, beginning with Cycle 20. The BSEP, Unit 1 amendments supporting transitioning to ATRIUM 10XM will be implemented during the spring 2012 refueling outage, beginning with Cycle 19.

The supplements dated June 9, July 29, September 29, October 12, November 9, November 18, and December 16, 2010; March 16, and April 6, 2011, provided additional information that clarified the application, did not expand the scope of the original *Federal Register* notice, did not change the NRC staff's original proposed no significant hazards consideration determination as

published in the *Federal Register* on August 10, 2010 (75 FR 48373), and did not expand the scope of the original license amendment request (LAR).

2.0 REGULATORY EVALUATION

2.1 Background

The NRC approved topical report BAW-10247PA, Revision 0 (Reference 12), in April 2008. The methodology in this topical report uses a fuel performance code, RODEX4, for best-estimate thermal-mechanical evaluation for fuel rods of BWRs (Reference 14). The RODEX4 fuel performance code is used to determine reactor core linear heat generation rate limits monitored as specified by BSEP, Unit 1 and 2 TS 3.2.3. The proposed amendments will add the topical report, BAW-10247PA to the list of analytical methods specified in BSEP, Units 1 and 2 TS 5.6.5.b.

The licensee also requested another amendment (Reference 15), in support of BSEP, Units 1 and 2 cores' transition to ATRIUM 10XM fuel, that would allow the addition of the AREVA's topical report, "ANP-10298PA, ACE/ATRIUM 10XM Critical Power Correlation," Revision 0, March 2010 (Reference 16) to the list of analytical methods specified in BSEP, Units 1 and 2 TS 5.6.5.b for determining core operating limits. Topical report, ANP-10298PA describes a new correlation, ACE/ATRIUM 10XM Critical Power Correlation, developed by AREVA to predict the critical power for BWRs. The ACE/ATRIUM 10XM correlation will be used to ensure that reactors using ATRIUM 10XM fuel remain within required safety limits during steady-state operation and anticipated operational occurrences (AOOs). The staff's determination on this amendment request will be documented in a separate safety evaluation (SE).

2.2 Regulatory Requirements and Guidance Documents

The NRC staff reviewed the LAR to evaluate the applicability of the BAW-10247PA methodology to the BSEP, Units 1 and 2 TSs, confirm that the use of this methodology is within the NRC-approved ranges of its applicability, and verify that the results of the analyses are in compliance with the requirements of the following General Design Criteria (GDC) specified in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50:

- GDC-10, "Reactor design," requiring the reactor design (reactor core, reactor coolant system (RCS), control and protection systems) to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including AOOs.

- GDC-12, "Suppression of reactor power oscillations," requiring that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed.

- GDC-15, "Reactor coolant system design," requiring the RCS and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

- GDC-20, "Protection system functions," requiring the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded

as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC-25, "Protection system requirements for reactivity control malfunctions," requiring the protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

- GDC-26, "Reactivity control system redundancy and capability," requiring two independent reactivity control systems of different design principles be provided, one of which is capable of holding the reactor subcritical under cold conditions.

- GDC-27, "Combined reactivity control system capability," requiring the reactivity control systems to be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes under postulated accident conditions.

- GDC-28, "Reactivity limits," requiring the reactivity control systems to 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.

- GDC-35, "Emergency core cooling," requiring a system to provide abundant emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

The Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP, NUREG-0800), Section 4.2, "Fuel System design", provides regulatory guidance for the review of fuel rod cladding materials and the fuel system. In addition, the SRP provides guidance for compliance with the applicable GDC in Appendix A to 10 CFR Part 50. According to SRP Section 4.2, the fuel system safety review provides assurance that:

- The fuel system is not damaged as a result of normal operation and AOOs,

- Fuel system damage is never so severe as to prevent control rod insertion when it is required,

- The number of fuel rod failures is not underestimated for postulated accidents, and

- Coolability is always maintained.

3.0 TECHNICAL EVALUATION

In general, methodologies or computer codes used to support licensing basis analyses are documented in topical reports which are reviewed by the NRC staff on a generic basis. The NRC staff in its safety evaluation for the approved topical report defines the basis for acceptance

in conjunction with any limitations and conditions on use of the topical report, as appropriate. A generic topical report describing a methodology or computer code does not provide the full justification for each plant-specific application. In situations where a plant-specific LAR references a topical report that has not been previously applied, the licensee submits a plant-specific analysis to demonstrate applicability of the topical report.

The BSEP, Units 1 and 2 cores contain 560 fuel assemblies that consist of both GE14 and ATRIUM-10 assemblies. The licensee plans to load the ATRIUM 10XM fuel design in BSEP, Units 1 and 2 in spring of 2012 and 2011, respectively. The proposed Cycle 20 core for BSEP, Unit 2 will consist of 226 fresh ATRIUM 10XM assemblies, 238 irradiated ATRIUM-10 assemblies, and 96 irradiated GE14 assemblies. The use of the ATRIUM 10XM fuel design requires the addition of AREVA topical report, BAW-10247PA (Reference 12) to the list of analytical methods specified in BSEP, Units 1 and 2 TS 5.6.5.b for determining core operating limits.

The BAW-10247PA methodology for the realistic evaluation of the thermal-mechanical performance of the fuel rods is developed in two major parts. The first is the best-estimate fuel performance code RODEX4 (Reference 14). The RODEX4 code models the thermal-mechanical behavior of the fuel rods during normal operation and AOOs. The RODEX4 code is structured into a modular architecture, in which the mechanical models are improved comparing to the previous RODEX codes. Also in RODEX4, high burnup models are implemented, and validation to an extensive fuel performance database has been performed. The second component of the BAW-10247PA methodology is the application of the RODEX4 code to determine the behavior of rods in a BWR core during normal operation and AOOs and to quantify the design margins relative to the generic design criteria in a statistical manner (Reference 12).

The RODEX4 fuel performance code simulates the thermal and mechanical response of a fuel rod in a reactor core as a function of exposure and local power and flow conditions during reactor operations. The code is calibrated to the observable pellet, clad and rod behaviors, such as, central pellet temperature, clad circumferential and axial deformation, clad oxidation, rod void volume, and fission gas release fraction.

The licensee, in its application dated April 29, 2010 (Reference 1), requested changes to the TSs to support the addition of the topical report, BAW-10247PA to the list of analytical methods specified in BSEP, Units 1 and 2 TS 5.6.5.b. By References 2 through 8, the licensee submitted information to demonstrate compliance with the NRC staff limitations and conditions imposed for application of the ACE/ATRIUM and RODEX methodologies, and to demonstrate the applicability of the AREVA codes and methods for BSEP, Unit 2 at extended power uprate conditions. As a result of addition of an alternate method to the approved topical report for ACEATRIUM 10XM methodology (Reference 16), the licensee submitted a letter dated April 6, 2011 (Reference 41), which included the licensee's submittal of AREVA's operability assessment for BSEP, Unit 2, Cycle 20 (Condition Report (CR) 2011-2274), the licensee's responses to the NRC staff's RAIs regarding this CR, and a license condition for the ACE/ATRIUM 10XM requested amendments (Reference 15)

The NRC staff has reviewed the LAR (Reference 1) in conjunction with the supplemental letters (References 2 through 8), the responses to the staff's requests for additional information (RAIs) (References 9, 10, and 11), and the licensee's submittal dated April 6, 2011 (Reference 41) to (1) evaluate the acceptability of the BSEP transition to ATRIUM-10XM fuel, (2) evaluate the use of the associated AREVA methodologies for licensing applications and (3) confirm adequate

technical basis for the proposed TS changes. In addition, the staff conducted a regulatory audit at the AREVA office in Bethesda, MD on November 3, 4 and 5, 2010, and reviewed the BSEP-specific safety analyses, calculation notebooks and associated fuel transition methodologies.

3.1 ATRIUM 10XM Fuel Design

The AREVA topical report, ANF-89-98(P)(A), Revision 1, Supplement 1 (Reference 17) is one of the NRC-approved methodologies that describes a process and criteria that allow Siemens Power Corporation (now AREVA NP) to apply to changes or improvements in existing BWR fuel designs without an explicit NRC review.

3.1.1 Fuel Thermal-Mechanical Design

The ATRIUM 10XM fuel design, as described in the submittal of ANP-2899(P), Revision 0 (Reference 18) and BSEP 10-0118, Enclosure 1 (Reference 6), shares many of the same features of ATRIUM-9 and ATRIUM-10 fuel designs that were used in BWR plants. The ATRIUM 10XM fuel bundle has the same basic geometry as the currently approved ATRIUM-10 fuel bundle design. The geometry consists of a 10x10 fuel lattice with a square internal water channel that displaces a 3x3 array of fuel rods. The ATRIUM 10XM incorporates additional key design features relative to ATRIUM-10 fuel:

[[

]][1]

The fuel pellet can be either UO_2 or UO_2-Gd_2O_3. The fuel rods are made with zircaloy-2 cladding. The fuel bundle is encased in a channel box of identical material and dimensions as the ATRIUM-10 fuel bundle design.

AREVA uses the approved generic fuel rod design methodology (Reference 17) and fuel performance code RODEX4 (Reference 12) to evaluate the thermal and mechanical performance of the ATRIUM 10XM fuel design. The RODEX4 code was approved to a peak rod average burnup of 62 gigawatt days per metric ton of uranium (GWd/MTU).

[1] The information in [[ ]] contained proprietary information and as such has been redacted in this nonproprietary version of SE.

3.1.1.1 Oxidation, Crud Buildup, and Hydriding

Section 4.2 of the SRP defines an allowable total cladding strain limit of 1.0 percent (total meaning elastic plus plastic, plus creep). As such, fuel vendors are required to (1) define the total strain capability of the fuel rod design/cladding alloy combination, (2) provide evidence supporting this strain capability, and (3) demonstrate that this design criterion is not exceeded during AOOs. Acceptable evidence of a cladding alloy's strain capability consists of mechanical testing under prototypical loading on irradiated cladding specimens. While irradiation damage under normal operation promotes an increase in yield strength (and lower ductility), the formation of zirconium hydrides within the cladding (resulting from the absorption of hydrogen during cladding corrosion) limits the strain capability of the fuel rod cladding. Therefore, the NRC requires that fuel vendors specify a design limit on cladding hydrogen content corresponding to the specified cladding strain limit and supporting database.

In response to a follow up RAI regarding a design limit on cladding hydrogen content for the ATRIUM 10XM cold worked stress relief (CWSR) zircaloy-2 cladding, the licensee in a letter dated March 16, 2011 (Reference 11) proposed a limit of **[[ ]]** weight part per million (wppm) hydrogen. In Section 2.2.2 of its response, the licensee states that the approved methodology assumes that the dominant phenomenon impacting the cladding ductility during in-reactor operation is irradiation hardening, which overshadows the effect of hydrogen when it is precipitated as hydrides. The NRC staff does not agree that precipitated hydrides have a lower impact on cladding than the fluence damage. In contrast, the NRC staff believes that the formation of hydrides, which is well documented, is critical to the cladding ductility and failure strain.

In support of the **[[ ]]** wppm hydrogen limit, the licensee, in its response, cites several publically available technical papers containing results from various mechanical test programs including a report from Pacific Northwest National Laboratory (PNNL), PNNL-17700 (Reference 19). The licensee credits an elastic strain capability greater than **[[ ]]**. At higher hydrogen content, overall strain at failure is diminished. Alloy composition, heat treatment, fluence, hydrogen content, hydride distribution and orientation, and testing temperature and protocols all impact measured uniform plastic strain that determines cladding failure strain. It is unclear whether any of the data was developed for the AREVA commercial grade irradiated zircaloy-2 CWSR cladding. Therefore, the basis for the proposed **[[ ]]** hydrogen limit is questionable and is not acceptable for ATRIUM 10XM fuel in BSEP, Units 1 and 2.

In Section 2.2.2.3 of its March 16, 2011, response, the licensee states that the 95/95 upper bound on hydrogen content at the 62 GWd/MTU burnup is **[[ ]]** wppm. Based on AREVA's empirical database and a survey of available zircaloy-2 hydrogen measurements, the NRC staff agrees that the end-of-life hydrogen uptake at the licensed burnup limit is unlikely to exceed **[[ ]]** wppm. At **[[ ]]** wppm, mechanical testing on irradiated cladding (i.e., measured uniform strain), including the cited PNNL report and testing on CWSR zirconium alloys supports the 1.0 percent total cladding strain limit. Therefore, the NRC staff finds the use of the 1.0 percent total strain limit up to the explicit 62 GWd/MTU burnup with the **[[ ]]** wppm hydrogen content upper bound for ATRIUM 10XM fuel in BSEP, Units 1 and 2 cores is acceptable.

In addition to explicitly accounting for the effects of cladding oxidation and crud, the NRC requires that fuel vendors establish a design limitation on cladding oxidation. This upper bound

on cladding oxidation defines (1) the limit of oxidation included in the design analyses, and (2) the limit of oxidation under which cladding oxide spallation and hydride blisters have not been observed. Uniform mechanical properties along the axial and circumferential directions of the fuel rod cladding are a foundation of currently approved fuel performance models. Localized cladding defects (e.g., spallation and hydride blisters) may significantly impact fuel rod stress and strain calculations and ultimately the ability to accurately predict cladding failure.

In response to an RAI regarding a design limit on cladding oxidation for the ATRIUM 10XM CWSR zircaloy-2 cladding (Reference 11), the licensee proposed an upper limit of **[[ ]]** micrometer peak oxide. The licensee in a regulatory commitment in Enclosure 5 of its response dated March 16, 2011, as stated in Section 4 of this SE, committed to confirm for each reload cycle that the predicted fuel cladding peak oxide thickness will remain below this upper limit based on the RODEX4 corrosion model.

In support of the cladding oxide limit, the licensee cites several publically available technical papers containing results from various hot-cell examinations. Further, AREVA's database of pool-side visual examination does not indicate oxide spallation up to the proposed oxide limit.

In Section 2.2.1.3 of its response, the licensee states that based on theoretical considerations, as confirmed by measurements, AREVA has established that the corrosion performance of CWSR and recrystalized annealed zircaloy-2 material is the same. Based upon comparisons to a larger database of zircaloy-2 oxide measurements, the NRC staff does not fully support this conclusion. However, this disagreement does not impact this review since RODEX4 models are currently limited to CWSR zircaloy-2 alloy.

The licensee in Section 2.2.1.3 of its response, states that the proposed oxide thickness limit is larger than a nodal average oxide thickness, because it is a span-maximum versus a node average value, and because its basis is consistent with a lift-off measurement that includes the combined thickness of oxide and tenacious crud. The licensee also notes that the peak span oxide database used in the development of the RODEX4 oxidation model explicitly captures the higher oxide thickness of nodules.

Based upon the information presented in response to the NRC staff RAI, the staff finds the licensee's proposed cladding oxide thickness limit with the associated regulatory commitment in Section 4 of this SE is acceptable for ATRIUM 10XM fuel design.

3.1.1.2 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX4 code and methodology. The rod internal pressures are allowed to exceed the reactor coolant system pressure provided that (1) the fuel-to-clad gap does not reopen due to the cladding creep outward, and (2) unfavorable hydride reorientation during cooldown does not occur. The re-opening gap would result in increasing the consequences of transient and accident conditions. Hydride reorientation could degrade the cladding strength. AREVA has established an approved rod internal pressure limit that exceeds the system pressure.

The maximum rod pressure is calculated under steady-state conditions including slow transients. The ATRIUM 10XM fuel design features shorter PLFRs than the ATRIUM-10 fuel design. The NRC staff questioned whether these PLFRs would be able to meet the rod internal pressure limit. AREVA provided the rod pressure calculations for full length fuel rods and PLFRs. The results showed all rods met the allowable limit.

The NRC staff performed rod internal pressure calculations using the NRC audit code FRAPCON-3. The FRAPCON-3 code is a best-estimate code with nominal input values and no modeling uncertainties. The results showed that the rod pressures could exceed the AREVA limit. In responding to the staff question, AREVA stated the discrepancy could be from the different power histories used in FRAPCON-3 and RODEX4 (Reference 5). The NRC staff derived power history according to thermal-mechanical operating limits (TMOL), which is a conservative approach. However, AREVA adopted a somewhat realistic approach using fuel rod power histories from a pool of almost [[ ]]. The power histories for all of the rods over all of the time are examined and [[

]]. AREVA contended that no single rod could follow such a bounding path of TMOL for its entire lifetime. This approach has been approved in the safety evaluation of the RODEX4 code (Reference 4). The NRC staff reviewed the response and found that the licensee's response was acceptable.

Based on the approved methodology in RODEX4, the NRC staff concludes that the rod pressure analysis is acceptable for the ATRIUM 10XM fuel design.

3.1.1.3 Overheating of Fuel Pellets

Licensees must ensure that core temperatures are maintained such that fuel failure from overheating of the fuel pellets does not occur. The fuel centerline temperature must remain below the UO_2 or UO_2-Gd_2O_3 melting temperature during normal operations and AOOs. AREVA established a linear heat generation rate (LHGR) limit to protect against fuel centerline melting during normal operations and AOOs.

For LHGR consequences, AREVA determined that [[ ]] are the most serious events and bound other slow transients (Reference 10). These two events are considered slow AOOs and are initiated at or near rated power conditions. Slow AOO transients are those occurring over a period of time in minutes. AREVA indicated that the analyses required assumptions of no initial intervention by the operator to generate maximum consequences. [[

]]

[[

]] events would result in increasing LHGR for maximum consequences. The analyses showed that fuel centerline temperature, cladding strain, and rod internal pressure are all within the allowable limits for equilibrium and cycle-specific cores. The NRC staff reviewed the results and considers that the analyses are conservative and acceptable.

Based on the conservative analyses, the NRC staff concludes that the analysis for overheating of fuel pellets is acceptable for the ATRIUM 10XM fuel design.

3.1.1.4 UO_2-Gd_2O_3 Rods

In responding to the staff question concerning the UO_2-Gd_2O_3 rod (Reference 10), AREVA stated that the UO_2-Gd_2O_3 material properties are explicitly included in the RODEX2-2A code (Reference 20). The RODEX2-2A code is used in conjunction with other two codes, RELAX and HUXY, for the ECCS performance evaluation.

The ECCS analysis demonstrated that all the acceptance criteria were met, including for UO_2-Gd_2O_3 rods. Based on the approved ECCS model, the NRC staff concludes that the UO_2-Gd_2O_3 rod treatment is acceptable for the ATRIUM 10XM fuel design.

3.1.1.5 Fuel Thermal Conductivity

The RODEX4 thermal conductivity model is a function of temperature, burnup, gadolinia, and plutonium content similar to the thermal conductivity model in the NRC audit code, FRAPCON-3. The RODEX4 model is based on the thermal conductivity data at intermediate to high temperatures of unirradiated UO_2 using measurements of thermal diffusivity and heat capacity by a specialized laser-flash method. Comparison of these two models for unirradiated UO_2 shows that the RODEX4 model predicts slightly higher thermal conductivity than the FRAPCON-3 model with increasing temperature. Based on the consistent results with FRAPCON-3, the NRC staff considers the RODEX4 model acceptable for ATRIUM 10XM fuel design.

The RODEX4 thermal conductivity model contains a degradation function for the treatment of urania-gadolinia (UO_2-Gd_2O_3) fuel. This degradation function is proportional to the weight fraction of Gd_2O_3 contained in burnable absorber rods. The RODEX4 model was compared to a corrected model for gadolinia addition in FRAPCON-3. The RODEX4 degradation for gadolinia is applied in addition to burnup degradation applied to UO_2 fuel. At high temperature, the FRAPCON-3 thermal conductivity model under predicts these data while the RODEX4 model provides a best-estimate prediction of these data. Based on the best-estimate predictions, the NRC staff finds that the RODEX4 gadolinia modification to fuel thermal conductivity is acceptable. Based on the consistent results from FRAPCON-3, the NRC staff concludes that the RODEX4 thermal conductivity model is acceptable for UO_2 and UO_2-Gd_2O_3 fuel pellets of ATRIUM 10XM fuel.

3.1.1.6 Cladding Thermal Conductivity

The cladding thermal conductivity in RODEX4 is the same as in both MATPRO and FRAPCON-3. MATPRO is an NRC-developed material handbook for fuel design. The FRAPCON-3 model is based on the MATPRO model. The NRC staff finds the use of RODEX4 for cladding thermal conductivity for the ATRIUM 10XM fuel design acceptable.

3.1.1.7 Gap Heat Transfer

RODEX4 treats the pellet-gap heat transfer with a model that consists of three modes, conduction through the interface gas, convection through the interface gas, and radiation heat transfer from the fuel surface to the cladding inner surface. The equations for these heat transfer modes are standard equations. The uncertainty in the gap heat transfer is dominated by the uncertainty in the effective gap size. The effective gap size is the sum of the mechanical gap, the effective surface roughness, and the extrapolation distance. FRAPCON-3 has similar

terms in its treatment with the exception that the extrapolation distance is considered as a temperature jump rather than a distance.

Because of the similar treatment in RODEX4 and FRAPCON-3, the NRC staff concludes that the gap conductance model is acceptable in the ATRIUM 10XM fuel design process.

3.1.1.8 Fuel Thermal Expansion

The fuel thermal expansion model in RODEX4 is nearly identical to that in FRAPCON-3 except in the high temperature range beyond 2000 degrees Celsius (°C) (3632 degrees Fahrenheit (°F)). The method used in RODEX4 is identical to the method in FRAPCON-3 to determine the expansion of the pellet due to thermal expansion. The impact of the greater thermal expansion on most RODEX4 temperature calculations should not be great because at higher temperature the gap should be nearly closed. Higher fuel thermal expansion should result in more conservative clad strain analyses. The RODEX4 code does not model the large increase in fuel expansion due to fuel melting. Because the code does not model properties above the fuel melting temperature, the RODEX4 code will be limited to applications with fuel temperatures less than the melting temperature.

Based on the similar features in the codes, the NRC staff concludes that the RODEX4 fuel thermal expansion model is acceptable in the ATRIUM 10XM fuel design process.

3.1.1.9 Fission Gas Release Model

The fission gas release (FGR) model in RODEX4 assumes spherical grains and uses a two-stage diffusion model for low and high temperatures. The FGR model has several empirical tuning parameters including two activation energies and preexponential diffusion coefficients, the burnup dependent coefficients, number of gas atoms for grain boundary saturation, and fractional area coverage. Different tuning parameters are applied during rapid power changes in order to adequately fit fission gas release data from ramp tests so the code satisfactorily predicts gas release during power ramps.

Examination of the RODEX4 predictions suggests that the code may have under predicted at high release. AREVA provided predicted-minus-measured versus burnup plots along with mean and standard deviations. AREVA also provided a histogram to demonstrate whether the data was skewed towards underprediction. AREVA stated that some of the FGR data were known to have high experimental uncertainties and possible biases such that these were eliminated in their optimized database. A small under predictive bias is still observed in the optimized database. However, a closer examination demonstrates that these under predictions are from power-ramped rods and not from rods with steady-state operation. AREVA stated that some of the FGR data also had high uncertainties, but demonstrated that the upper 95/95 confidence predictions bounded all the power-ramped data. AREVA also provided predictions of a typical BWR 4 fuel rod design with a nominal grain size and typical power ramps. FRAPCON-3 predictions were performed against a similar database. The results show that the RODEX4 predictions are consistent with the FRAPCON-3 results.

Due to limitations within the FGR model, the analytical fuel pellet grain size shall not exceed 20 microns 3-D when the as-manufactured fuel pellet grain size could exceed 20 microns 3-D. Based on the comparison calculations done with FRAPCON-3, the NRC staff concludes that the fission gas release model in RODEX4 is acceptable for steady-state and transient analyses for the ATRIUM 10XM fuel.

3.1.1.10 Mechanical Model

The modeling of the fuel rod mechanical behavior in RODEX4 assumes that the pellet is non-rigid such that the fuel and cladding are allowed to strain when there is hard contact between the fuel rod and cladding. The fuel strains are calculated from fission product (gaseous and solid) swelling, densification, thermal expansion, fuel cracking, and fuel creep models. When there is hard contact between the fuel and cladding, the two are locked together, i.e., there is no axial slippage.

The cladding mechanical model assumes a plane strain, i.e., cladding deformation in the radial azimuthal directions are independent of the axial direction, i.e., shear stress and strain are assumed to be zero. The code utilizes anisotropic properties for the cladding. Based upon its review of the cladding models including creep, the NRC staff concludes that the cladding models are acceptable for RODEX4.

Based on the data and code comparisons, the NRC staff concludes that the mechanical model in RODEX4 is acceptable for the ATRIUM 10XM fuel mechanical design.

3.1.1.11 RODEX4 Limitations and Conditions

The NRC staff safety evaluation for the AREVA topical report identified five limitations and conditions on the use of the BAW-10247PA methodology. Compliance with the following conditions and limitations are ensured when referencing the RODEX4 code as described in BAW-10247PA, Revision 0 (Reference 4). The licensee has committed to comply with these limitations and conditions (Reference 1).

1) Due to limitations within the FGR model, the analytical fuel pellet grain size shall not exceed 20 microns 3-D when the as-manufactured fuel pellet grain size could exceed 20 microns 3-D.

   Since the fuel pellet grain size in the RODEX4 code does not exceed 20 microns 3-D, the NRC staff considers that this condition is satisfied.

2) RODEX4 shall not be used to model fuel above incipient fuel melting temperatures.

   Since RODEX4 does not predict fuel temperature above incipient fuel melting temperatures, the NRC staff considers that this condition is satisfied.

3) The hydrogen pickup model within RODEX4 is not approved for use.

   As mentioned in Section 3.1.1.1, RODEX4 does not use the unapproved hydrogen pickup model for predicting hydrogen uptake; the NRC staff considers that this condition is satisfied.

4) Due to the empirical nature of the RODEX4 calibration and validation process, the specific values of the equation constants and tuning parameters derived in the topical report, BAW-10247PA, Revision 0 (as updated by RAI responses) become inherently part of the approved models. Thus, these values may not be updated without necessitating further NRC review.

During the regulatory audit, the NRC staff confirmed that no unreviewed update in the RODEX4 code occurred. Thus the NRC staff considers that this condition is satisfied.

5) RODEX4 has no crud deposition model. Due to the potential impact of crud formation on heat transfer, fuel temperature, and related calculations, RODEX4 calculations must account for a design basis crud thickness. The level of deposited crud on the fuel rod surface should be based upon an upper bound of expected crud and may be based on plant-specific history. Specific analyses would be required if an abnormal crud or corrosion layer (beyond the design basis) is observed at any given plant. For the purpose of this evaluation, an abnormal crud/corrosion layer is defined by a formation that increases the calculated fuel average temperature by more than 25 °C beyond the design basis calculation.

As mentioned in Section 3.1.1.1, the licensee will perform a specific analysis to address the higher crud level in the event of abnormal crud is observed for a plant. Thus the NRC staff considers that this condition is satisfied.

In addition, the NRC staff safety evaluation for topical report BAW-10247PA concludes that RODEX4 is approved for modeling BWR fuel rods with the following conditions:

a. Peak rod average burnup limit of 62 GWd/MTU.
b. Solid UO_2 fuel pellet with a maximum gadolinia content of 10.0 weight percent.
c. CWSR zircaloy-2 fuel clad material.

Based on the audit, the NRC staff confirms that the ATRIUM 10XM fuel design has complied with these conditions. Therefore, the NRC staff concludes that the ATRIUM 10XM fuel design satisfies all the limitations and conditions specified in the staff safety evaluation.

3.1.1.13 Conclusion

The NRC staff has reviewed the AREVA submittal of the ATRIUM 10XM fuel design as described in ANP-2899P Revision 0, and the licensee submittal of the ATRIUM 10XM BSEP specific fuel design performance based on the BAW-10247PA, Revision 0 methodology, as described in BSEP 10-0118, Enclosure 1 (Reference 6). Based on the NRC staff's evaluation, the staff concludes that BAW-10247PA, Revision 0 is acceptable for referencing in licensing applications for BWRs to the extent specified and under the limitations and conditions delineated in the NRC staff safety evaluation of BAW-10247PA, Revision 0. The ATRIUM 10XM fuel design is approved to the peak rod average burnup of 62 GWd/MTU.

3.2 AREVA Methodologies and Computer Codes

As indicated in the LAR (Reference 1), the licensee performed licensing analyses using a variety of AREVA methodologies and computer codes as described below. The NRC staff evaluated the applicability of these codes and methods specifically to BSEP, Units 1 and 2.

3.2.1 AREVA Methodologies

Critical Power Correlation Methodologies

The safety limit minimum critical power ratio (SLMCPR) for all fuel types in the BSEP, Unit 2 Cycle 20 core is determined using the methodology described in topical report, ANF-524(P)(A)

(Reference 21). This topical report provides the basis for the methodology for determining the operating safety limit for minimum critical power ratio (MCPR) that ensures that 99.9 percent of the fuel rods are protected from boiling transition (BT) during normal operation and AOOs. The methodology consists of a series of Monte Carlo calculations in which the variables affecting the probability of BT are determined for each Monte Carlo trial. The analysis was performed with a power distribution that conservatively represents expected reactor operating states that could both exist at the MCPR operating limit and produce a MCPR equal to the SLMCPR during an AOO.

The BSEP, Unit 2 Cycle 20 SLMCPR analysis used the ACE/ATRIUM 10XM critical power correlation additive constants and additive constant uncertainty for the ATRIUM 10XM fuel described in Reference 16. The ACE/ATRIUM 10XM correlation is used to accurately predict assembly critical power for the ATRIUM 10XM fuel design. The correlation provides an accurate prediction of the limiting rod. The impact of local spacer effects and assembly geometry on critical power is accounted for by two different sets of parameters. The first is a set of constants, one constant for each rod in the assembly, called additive constants, listed in Table 5-2 of Reference 16, and the second a set of parameters that provides for modeling of design-specific axial effects including spacers within the critical power correlation. For comparison of correlation predictions to experimental data, an experimental critical power ratio (ECPR) is defined as the ratio of calculated critical power to the measured critical power. The ECPR distribution associated with ACE/ATRIUM 10XM correlation is adequately represented with a normal distribution. The range of applicability of the ACE/ATRIUM 10 XM correlation is listed in Table 2-1 of Reference 16 and is reproduced below.

Table 3.1: Range of Applicability of ACE/ATRIUM-10 and ACE/ATRIUM 10XM Correlations

[[		
		]]

The ANF-524(P)(A) methodology is modified slightly for use with the ACE correlation form due to the [[ ]]. The modifications concern the treatment of channel bow variation along the length of the fuel channel. [[

]] The key difference between the SPCB (Reference 22) and both ACE correlations that must be accounted for in the safety limit methodology is [[

]]. The impact of the modifications is that they [[

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[[

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The NRC staff identified two limitations and conditions on the use of the ACE/ATRIUM 10XM correlation. Limitation and Condition 1 states: "Since ACE/ATRIUM-1OXM was developed from test assemblies designed to simulate ACE/ATRIUM-1OXM fuel, the methodology may only be used to perform evaluations for fuel of that type without further justification."

The licensee will apply the ACE/ATRIUM 10XM critical power correlation to ATRIUM 10XM fuel in BSEP, Unit 2 for Cycle 20 and in BSEP, Unit 1 for Cycle 19. BSEP, Unit 2 Cycle 20 is expected to consist of 226 fresh ATRIUM 10XM fuel assemblies, 238 once-burned ATRIUM-10 fuel assemblies, and 96 twice-burned GE14 fuel assemblies. The licensee will continue to apply the SPCB correlation to the GE14 fuel design.

The BSEP, Unit 1 core will be loaded with ATRIUM 10XM fuel assemblies for Cycle 19 and will contain the third reload of AREVA fuel; therefore, Unit 1 is not expected to contain any GE14 fuel.

Limitation and Condition 2 states: "ACE/ATRIUM-10XM should not be used outside its range of applicability defined by the range of the test data from which it was developed and the additional justifications provided by AREVA in this submittal. This range is listed in Table 2-1 of Reference 1." Reference 1 is ANP-10298P.

The restrictions on range of applicability for mass flow rate, pressure, and inlet subcooling are also implemented in AREVA engineering computer codes, which include the BSEP POWERPLEX-III core monitoring system. The restriction on design local peaking is also implemented in AREVA automation tools.

3.2.2 Reactor Analysis Methodologies and Computer Codes

The NRC approved the use of ATRIUM-10 fuel and core design methodologies to determine BSEP core operating limits with the issuance of License Amendments 246 and 274 for BSEP, Units 1 and 2, respectively (Reference 13).

XCOBRA/XCOBRA-T (XN-NF-84-105-P-A): XCOBRA predicts the steady-state thermal-hydraulic performance of BWR cores at various operating conditions and power distributions. It is used to evaluate pressure drops, channel and bypass flow distributions, and minimum critical power ratios (MCPRs), as well as the hydraulic compatibility of fuel designs. XCOBRA-T predicts the transient thermal-hydraulic performance of BWR cores during postulated system transients and is used to evaluate the change in critical power ratio (ΔCPR) for the limiting fuel bundles in the core. As documented in XN-NF-84-105(P)(A) (Reference 23), the XCOBRA-T code has been approved by the NRC for use in BWR licensing applications. The use of the steady-state XCOBRA code has been accepted by the NRC staff (Reference 24) based on approval of XCOBRA-T and the similarity of the thermal-hydraulic models between the codes. The BSEP licensing analysis (Reference 2) shows that the core thermal-hydraulic conditions during steady-state and transient conditions are within the NRC-approved range of the code. The staff concludes that the application of XCOBRA and XCOBRA-T for the BSEP core thermal-hydraulic calculations is acceptable.

COTRANSA2 ANF-913(P)(A): COTRANSA2 is a BWR system transient simulation code, which includes an axial one-dimensional neutronics model that captures the effects of axial power shifts associated with the system transients (Reference 25). It is used to evaluate key reactor system parameters during core-wide BWR transient events. These parameters, such as power, flow, pressure, and temperature, are provided as boundary conditions to the hot channel analyses in XCOBRA-T and XCOBRA codes for determining critical power ratios for limiting transients. As documented in ANF-913(P)(A), the code has been generically approved by the NRC to analyze system responses to fast transients in BWRs (Reference 25).

The NRC approval of COTRANSA2 is subject to the limitations set forth in the safety evaluations for the methodologies described and approved for XCOBRA-T (Reference 23). COTRANSA2 is approved to perform the system analysis of the following fast AOO and anticipated transient without scram (ATWS) events: (1) load rejection without bypass, (2) turbine trip without bypass, (3) feedwater controller failure maximum demand, (4) pressure regulator downscale failure, (5) ATWS main steam isolation valve closure and pressure regulator failure open, and (6) the American Society of Mechanical Engineers (ASME) overpressurization analysis. The NRC staff,

upon reviewing the results from the transient analyses, concludes that the licensee's use of COTRANSA2 to perform analysis of fast transient events for the BSEP is acceptable.

SLMCPR Methodology (ANF-524-P-A): The SLMCPR is imposed to protect at least 99.9 percent of the fuel rods in the core from boiling transition during steady-state and transient conditions (Reference 21). The NRC approved the topical report, ANF-524(P)(A), which identifies the fuel- and nonfuel-related uncertainties and the statistical process used to determine an MCPR safety limit. Compliance to each restriction in the NRC staff's SE approving topical report ANF-524(P)(A) is demonstrated in Reference 26. As discussed in further detail in Section 3.3 of this SE, the staff finds that the appropriate values of the MICROBURN-B2 uncertainties are used for the BSEP SLMCPR. As discussed in further detail in Section 3.3 of this SE, the staff finds that the appropriate values of the MICROBURN-B2 uncertainties are used for the BSEP SLMCPR. Therefore, the staff finds the licensee's use of the SLMCPR methodology as documented in topical report ANF-524(P)(A) in support of the BSEP license amendment application acceptable.

CASMO-4/MICROBURN-B2 (EMF-2158(P)(A)): The two principal computer programs for BWR nuclear design and analysis used by AREVA are CASMO-4 and MICROBURN-B2. The CASMO-4 code is a two-dimensional multi-group transport theory code used to calculate the lattice physics constants of BWR fuel assemblies. The MICROBURN-B2 code is a two group nodal code used for the three-dimensional simulation of the nuclear and thermal-hydraulic conditions in BWR cores. The MICROBURN-B2 code determines core-wide nodal neutron flux, fission power, and coolant density distributions; reactivity parameters; nodal exposure and nuclide density distributions; control rod patterns; channel inlet flow distributions; and fuel thermal performance parameters such as linear heat generation rate (LHGR), axial planar LHGR, and critical power ratio (CPR). These results are used to design fuel cycles, to assess safety margins, and to monitor operating reactor cores.

Section 5.2.3 of BAW-10247PA (Reference 12) describes the application of power distribution measurement uncertainties (i.e., radial and axial) by the BAW-10247PA methodology. The radial and axial power uncertainties are calculated from uncertainty components as described in EMF-2158(P)(A) (Reference 27). Three of the uncertainty components used to calculate these power distribution uncertainties are determined using traversing incore probe (TIP) measurements. These uncertainty components are: (1) The deviation between the CASMO 4/MICROBURN-B2 (C4/MB2) calculated TIP response and the measured TIP response on a radial ($\delta T'_{ij}$), nodal ($\delta T'_{ijk}$) and planar ($\delta T'_{planar}$) bases, (2) TIP measurement uncertainty on a radial (δT^{m}_{ij}), nodal (δT^{m}_{ijk}) and planar (δT^{m}_{planar}) bases, and (3) Synthesis uncertainty on a radial (δS_{ij}), and nodal (δS_{ijk}) basis. The licensee has shown that all three uncertainty components identified above are bounded by the values reported in sections 9.4 and 9.5 of EMF-2158(P)(A), and the net calculated TIP distribution uncertainty components (δT_{ijk}, δT_{ij} and δT_{planar}) are also bounded by the values reported in section 9.4 of EMF-2158(P)(A). Details of the calculation process and results of the power distribution uncertainties are discussed in Section 3.3.5 of this safety evaluation.

The licensee has shown compliance with the NRC-approved methodology for power distribution measurement uncertainties and, therefore, the staff has determined that the licensee's use of the EMF-2158(P)(A) methodology is acceptable.

Stability Methodology (NEDO-32465-A): BSEP has implemented the Boiling Water Reactor Owners Group long-term solution, Option III as their licensing basis stability protection

methodology (Reference 28). To support Option III, the licensee uses the RAMONA5-FA system analysis code to generate the delta over initial CPR versus oscillation magnitude relationship, which provides the CPR performance during reactor instability (Reference 29). RAMONA5-FA is a coupled neutronic thermal-hydraulic three-dimensional transient model for the purpose of determining the relative change in ΔCPR and the hot channel oscillation magnitude on a plant-specific basis.

Backup stability protection (BSP) analyses are performed in anticipation of the long-term Option III solution becoming unavailable and oscillation power range monitoring system being declared inoperable. The BSP is a prevention approach where certain areas on the power-to-flow map, when instability is likely based on decay ratio calculations, are excluded from operation. The calculations to support BSP are performed using the NRC-approved STAIF code (References 30 and 31). Compliance to each restriction in the NRC staff's SE approving use of the STAIF code is demonstrated in Reference 26. Therefore, the staff finds that the licensee's use of the STAIF code in support of BSEP licensing application is acceptable.

EXEM BWR-2000 Loss-of-Coolant Accident Methodology (EMF-2361-P-A): The AREVA methodology for showing compliance with 10 CFR Part 50, Appendix K, is referred to as the EXEM BWR-2000 evaluation model. This model was reviewed and approved by the NRC staff in Reference 20. The EXEM BWR-2000 methodology employs three primary codes. The reactor system and hot channel response is evaluated with RELAX (Reference 20); fuel assembly heatup during the loss-of-coolant accident (LOCA) is analyzed with HUXY (Reference 32), which incorporates approved cladding swelling and rupture models (Reference 33); and stored energy and fuel characteristics are determined with RODEX2 (Reference 34). Compliance to each restriction in the staff's SE approving use of the STAIF code is demonstrated in Reference 26. Therefore, the staff finds that the licensee's use of the EXEM BWR-2000 and associated code systems in support of the BSEP licensing application is acceptable.

Methods and Codes Summary

The licensee has evaluated the compliance with the restrictions specified in each of the staff's safety evaluations approving the AREVA topical reports in the LAR (Reference 1). Accordingly, the staff concludes that the licensee adequately demonstrated conformance to the SE conditions. The licensee performed plant-specific analyses of the limiting licensing basis events with the AREVA codes and methods to show that the use of those codes and methods is acceptable. The analyses show that the results meet the applicable criteria (Sections 3.2 and 3.3 of the SE). Therefore, the NRC staff concludes that the application of the NRC-approved AREVA codes and methods to the BSEP for licensing analysis is acceptable.

3.3 Transition Core Approach

Each of the BSEP, Units has 560 assemblies. The BSEP, Unit 2 Cycle 20 core will contain 226 fresh ATRIUM 10XM assembles, 238 once-burned ATRIUM-10 assemblies and 96 twice-burned GE14 fuel assemblies. The NRC staff reviewed the licensee's evaluation of the mixed-core configuration of the BSEP, Unit 2 Cycle 20 core to predict the thermal-hydraulic performance, hydraulic compatibility, thermal margin performance, critical power performance, and the impact on core design and licensing analysis (Reference 4).

3.3.1 Hydraulic Compatibility

The licensee reported the results of thermal-hydraulic analyses in accordance with NRC-approved AREVA thermal-hydraulic methodology (References 17, 35, and 36). The methodology and constitutive relationships used in the licensee's calculation of pressure drop in BWR fuel assemblies are presented in Reference 36 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. Hydraulic compatibility, as it relates to the relative performance of the ATRIUM 10XM, ATRIUM-10, and GE14 fuel designs, has been evaluated. Analysis for mixed cores with ATRIUM 10XM, ATRIUM-10, and GE14 fuel assemblies were performed to demonstrate that the thermal-hydraulic design criteria are satisfied for transition core configurations.

The calculations were performed with explicit modeling of ATRIUM 10XM, ATRIUM-10, and GE14 assemblies for several power-to-flow conditions, rated and off-rated, and for bottom-, middle-, and top-peaked axial power distributions. Four core configurations were analyzed to address the relative assembly comparisons and core hydraulic compatibility evaluations. The four core configurations are; (1) ATRIUM-10 coresident with GE14 for Cycle 19, (2) first transition core for Cycle 20 fuel loading of ATRIUM 10XM, ATRIUM-10, and GE14, (3) second transition loading with ATRIUM 10XM and ATRIUM-10, and (4) full core of ATRIUM 10XM. Results of the calculations listed in Reference 4 and those presented during the regulatory audit indicate that core average results and the difference between ATRIUM 10XM, ATRIUM-10, and GE14 results at rated power are within the range considered compatible. Similar agreement exists at off-rated power levels. **[[**

]] The NRC staff concludes that the licensee's hydraulic compatibility analysis provides reasonable assurance that the introduction of ATRIUM 10XM fuel into the BSEP units will not significantly impact the core flow distribution.

3.3.2 Thermal Margin Performance

Thermal margin analyses were performed in accordance with the thermal hydraulic methodology based on AREVA's XCOBRA code. The calculation of CPR, which is a measure of thermal margin performance, is established by means of an empirical correlation based on results of BT test programs. CPR values for ATRIUM 10XM are calculated with the NRC-approved ACE/ATRIUM 10XM critical power correlation (Reference 16) while the CPR values for the ATRIUM-10 and GE14 fuel are calculated using the NRC approved SPCB critical power correlation (Reference 22). Fuel assembly design features are incorporated in the CPR calculation through the K-factor in the ACE correlation and through the F-effective term for the SPCB correlation. The K-factors and F-effective terms are based on local power peaking factors that are functions of assembly void fraction and exposure. Analysis results (Reference 4) indicate that the introduction of ATRIUM 10XM in the BSEP units will not cause thermal margin problems for the coresident fuel designs.

The NRC staff concludes that there is no adverse impact on thermal margin performance due to the mixed core configuration at BSEP units.

3.3.3 Safety Limit Minimum Critical Power Ratio (SLMCPR) Analysis

The SLMCPR is defined as the minimum value of the critical power ratio which ensures that less

than 0.1 percent of the fuel rods in the core are expected to experience boiling transition during normal operation or an AOO. The SLMCPR is determined using the methodology in Reference 21 using a power distribution that conservatively represents expected reactor operating states that exist at the MCPR operating limit and produce an MCPR equal to the SLMCPR during an AOO. The BSEP, Unit 2 Cycle 20 SLMCPR analysis uses ACE/ATRIUM 10XM critical power correlation additive constants and additive constant uncertainty for ATRIUM 10XM fuel described in Reference 16. For the ATRIUM 10 fuel SLMCPR analysis, the SPCB critical power correlation additive constants and related uncertainty are used as per Reference 22. The SLMCPR analysis explicitly includes channel bow. The channel bow local peaking uncertainty is a function of the nominal and bowed local peaking factors and the standard deviation of the channel bow.

The SLMCPR analysis results supporting a two-loop operation SLMCPR is 1.11 with 0.092 percent of the rods in BT (Reference 8). For single-loop operation, the SLMCPR value is 1.13 with 0.076 percent of the rods in BT.

3.3.4 Core Design and Licensing Analysis

Fuel cycle design and fuel management calculations for the Cycle 20 operation of BSEP, Unit 2 are performed in accordance with the NRC-approved methodology, EMF-2158(P)(A) (Reference 27). The CASMO-4 lattice depletion code is used to generate nuclear data including cross sections and local power peaking factors. The MICROBURN-B2 three-dimensional core simulator code utilizes the pin power reconstruction model to determine the thermal margins. The ACE correlation is used for the ATRIUM 10XM fuel assemblies while the coresident ATRIUM-10 and GE14 fuel assemblies are monitored with the SPCB correlation. The core neutronic design includes control blade depletion, explicit neutronic treatment of the spacer grids, explicit modeling of PLFR plenums, and explicit modeling of the water rod flow.

Control rod patterns are developed to be consistent with conservative margin to thermal limits. The fuel cycle design demonstrates adequate hot excess reactivity and cold shutdown margin throughout the cycle. Fuel assembly thermal-mechanical limits for ATRIUM 10XM, ATRIUM-10 and coresident fuel are verified and monitored for each mixed core designed by AREVA. The thermal mechanical limits established by the vendor of the coresident fuel are applied for that fuel in mixed (transition) cores. AREVA performed design and licensing analyses to demonstrate that the core design meets the limits during steady-state and AOO conditions. The NRC staff finds the approach acceptable.

3.3.5 Radial and Axial Power Distribution Measurement Uncertainties

As stated on Page 9-1 of topical report, EMF-2158(P)(A) (Reference 27), the AREVA methodology for measuring the power distribution in a BWR reactor and the procedure by which the uncertainty associated with the measurement of a BWR power distribution would be determined, was originally described in XN-NF-80-19(P)(A), Volume 1, Supplements 3 and 4 (Reference 37). Section 5 of Reference 37, and Section 9 of Reference 27, together provide a very detailed description of the analyses and calculations performed to determine the TIP uncertainty components. The NRC staff requested more details on the TIP uncertainty components listed in the two tables on Page 4 of Enclosure 1 of Reference 1. The licensee provided details of these TIP uncertainty measurements in Reference 9 and during the regulatory audit conducted in November 2010.

Figures 1, 2, and 3 on Pages 11 through 13 of Enclosure 1 of Reference 1 present 177 database points. Each database point is calculated using a TIP flux map consisting of measurements obtained from 21 axial levels at 31 radial core locations. Except for the size of the data population, the detailed equations provided by References 27 and 37 and listed in Table 17.1 of Reference 9 are the same for both the database points and the final uncertainty components. Each database point is based on 651 local TIP readings (i.e., 21 times 31) obtained at a core operating state characterized by its core power, core void fraction, and core power-to-flow ratio, whereas the TIP uncertainty components are based on 115,227 local TIP readings. The 177 TIP flux maps were obtained from BSEP, Units 1 and 2 core operating states from March 2000 through February 2010, from cores consisting entirely of 9x9 GE13 fuel, mixed cores of GE13 and 10x10 GE14 fuel, GE14 fuel alone, and mixed cores of GE14 and 10x10 ATRIUM-10 fuel.

The BSEP C4/MB2 benchmark completed by AREVA to incorporate explicit water rod, PLFR plenum, and spacer model options was not available at the time the TIP statistics presented in the CP&L letter, BSEP 10-0057 (Reference 1) were calculated; however, none of these changes materially affect C4/MB2 calculated TIP distributions. The licensee has since recalculated TIP statistics based on the latest AREVA benchmark. The database values presented in the CP&L letter, BSEP 10-0057 that are dependent on C4/MB2 calculated TIP response are plotted against the values calculated based on the latest AREVA benchmark in Figure 17.1 of Reference 9. The results demonstrate the explicit water rod, PLFR plenum, and spacer model options have no impact on the TIP statistics. The TIP uncertainty component values and trends based on the latest C4/MB2 benchmark (i.e., incorporating explicit water rod, PLFR plenum, and spacer model options) are provided as Table 17.2 and Figures 17.2, 17.3 and 17.4 of Reference 9.

The D-Lattice (BSEP) uncertainty component values identified in Sections 9.4 and 9.5 of EMF-2158(P)(A) bound the BSEP-specific uncertainty component values shown in Table 17.2 and reproduced below in Table 3.3. This result demonstrates that the uncertainties applied by the BAW-10247PA and ANF 524(P)(A) methodologies and determined in accordance with the EMF-2158(P)(A) methodology are applicable to BSEP, Units 1 and 2.

The NRC staff concludes that the licensee's evaluation of the TIP database for previous cycles including both BSEP units has demonstrated that uncertainties documented in EMF-2158(P)(A) for D-Lattice plants (BSEP, Units 1 and 2) remain conservative and none of the features of the ATRIUM 10XM design will have any impact on the accuracy of the methodology to predict TIP response.

Table 3.3: Updated TIP Component Uncertainties

Component	Reference 1 Value (%)	Latest Benchmark Value (%)	EMF-2158 Value (D-Lattice (%)
Nodal TIP Measurement Uncertainty, δT^{m}_{ijk}	1.90	1.90	[[ ]]
Radial Tip Measurement Uncertainty, δT^{m}_{ij}	1.25	1.25	[[ ]]
Planar TIP Measurement Uncertainty, δT^{m}_{planar}	1.97	1.97	[[ ]]
Nodal Deviation between measured TIP Readings and Calculated by MB2, δT_{ijk}	4.47	4.44	[[ ]]
Radial Deviation between Measured TIP Readings and Calculated by MB2, δT_{ij}	2.07	2.07	[[ ]]
Planar Deviation between Measured TIP Readings and Calculated by MB2, δT_{planar}	2.58	2.58	[[ ]]
Nodal Synthesis Procedure Uncertainty, δS_{ij}	0.22	0.21	[[ ]]
Radial Synthesis Procedure Uncertainty, δS_{ijk}	1.79	1.68	[[ ]]
Net Nodal Calculated TIP Distribution Uncertainty, δT_{ijk}	Calculation method provided in Reference 1 instead of proprietary value	[[ ]]	[[ ]]
Net Radial Calculated TIP Distribution Uncertainty, δT_{ij}		[[ ]]	[[ ]]
Net Planar Calculated TIP Distribution Uncertainty, δT_{planar}		[[ ]]	[[ ]]

3.3.6 Transition Core Summary

In the AREVA thermal-hydraulic methodology, each fuel type is explicitly modeled. Therefore, the impacts of the differences in mechanical design on geometry and loss coefficients are explicitly accounted for. The critical power performance of each fuel type is also explicitly modeled using the applicable critical power correlation for each fuel design. Limits are established for each fuel type and operation within these limits is verified by the core monitoring system during plant operation. Therefore, the NRC staff concludes that the licensee's treatment of transition cores is acceptable.

3.4 Plant-Specific Reload Safety Analyses

Reload licensing analyses in support of the BSEP, Units 1 and 2 fuel transition are performed using NRC-approved generic methodologies for boiling water reactors. The reload licensing analyses are performed for the potentially limiting events and other events are identified as disposition events. The results of the analyses are used to establish the BSEP TSs core

operating limit report limits and ensure that that the design and licensing criteria are met (Reference 8).

A summary of disposition of events is listed in Tables 2.1 and 2.2 of Reference 8. The objective of the disposition of events is to identify the limiting events which must be analyzed to support operation at the BSEP with the introduction of ATRIUM 10XM fuel. Events and analyses identified as potentially limiting are either evaluated generically for the introduction of ATRIUM 10XM fuel or on a cycle-specific basis.

The sections below list the limiting events and a short description of the analyses and results.

3.4.1 Anticipated Operational Occurrences

Section 3.2 of this safety evaluation lists the major codes used in the thermal limits analyses, neutronics methodology, and critical power calculations. The limiting exposure for rated power pressurization transients is typically at end-of-full power when the control rods are fully withdrawn. To provide additional margin to the operating limits earlier in the cycle, analyses were also performed to establish operating limits at a near end-of-cycle (NEOC) exposure of 16,700 megawatt days per metric ton of uranium. Analyses were also performed to support extended cycle operation with final feedwater temperature reduction (FFTR) and power coastdown. The sections below provide brief descriptions of a few select AOO analyses performed for the BSEP units.

Load Rejection No Bypass (LRNB): The load rejection causes a fast closure of the turbine control valves. The resulting compression wave in the steam lines into the vessel creates a rapid pressurization. Pressurization causes a decrease in voids and causes a rapid increase in power. The turbine control valve closure causes a reactor scam. LRNB analyses are performed for a range of power/flow conditions to support generation of the thermal limits. Results are used to generate the NEOC and end-of-the cycle licensing basis (EOCLB) operating limits for both technical specifications scram speed (TSSS) and nominal scram speed (NSS) insertion times.

Turbine Trip No Bypass (TTNB): The turbine trip causes a closure of the turbine stop 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. The closure of the turbine stop valves also causes a reactor scram. The excursion of the core power due to the void collapse is terminated primarily by the reactor scram and re-voiding of the core. The LRNB analyses for previous cycles have shown that the consequences of the TTNB event are bound by those of the LRNB event. The licensee's TTNB analysis for Cycle 20 has shown that the LRNB event remains bounding.

Feedwater Controller Failure (FWCF): The increase in feedwater flow due to a failure of the feedwater control system 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. The water level continues to rise and eventually reaches the high water level trip point. The high water level trip causes the turbine stop valves to close in order to prevent damage to the turbine from excessive liquid inventory in the steam lines. The valve closures create 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. FWCF analyses are performed for a range of power/flow conditions to generate the thermal limits. Reference 8

lists the FWCF analyses results that are used to generate the NEOC and EOCLB operating limits for both TSSS and NSS insertion times.

Loss of Feedwater Heating (LFWH): The LFWH event analysis supports an assumed 100 °F decrease in the feedwater temperature. The result is an increase in core inlet subcooling, which reduces voids, thereby increasing core power and shifting the axial power distribution toward the bottom of the core. The axial power shift and increase in core power causes the voids to build up in the bottom of the core, acting as negative feedback to the increased subcooling effect. The negative feedback moderates the core power increase. The increase in core thermal power event does not result in a corresponding increase in steam flow because some of the added power is used to overcome the increase in inlet subcooling. The increase in steam flow is accommodated by the pressure control system via the turbine control valves or the turbine bypass valves, so no pressurization occurs. The licensee performed a cycle-specific analysis according to Reference 38 methodology to determine the change in MCPR for an LFWH event. The NRC staff finds the results acceptable.

Control Rod Withdrawal Error (CRWE): The CRWE transient is an inadvertent reactor operator initiated withdrawal of a control rod. This withdrawal increases local power and core thermal power, lowering the core MCPR. The CRWE is typically terminated by control rod blocks initiated by the rod block monitor (RBM). The CRWE event was analyzed assuming no xenon and allowing credible instrumentation out-of-service in the rod block monitor (RBM) system. The CRWE analysis has demonstrated, in addition to support of the standard filtered RBM setpoint reductions, 1 percent strain and centerline melt criteria are met for both ATRIUM 10XM and ATRIUM-10.

Equipment Out-of-Service Scenarios (EOOS):

The following EOOS scenarios are supported for the BSEP, Unit 2 Cycle 20 operation:

Feedwater heater out-of-service (FHOOS): This scenario assumes an FFTR of 110.3 °F at rated power and steam flow. An FFTR causes an increase in core inlet subcooling that can change the axial power shape and core void fraction. The steam flow for a given power level decreases since more power is required to increase the enthalpy of the coolant to saturated conditions. The FWCF is analyzed to ensure that appropriate FHOOS operating limits are established.

Other EOOS scenarios analyzed are:

- Turbine bypass valves out-of-service (TBVOOS) - Analysis of the FWCF are performed to establish the TBVOOS operating limits.

- Combined FHOOS and TBVOOS - Operating limits for this combination are established using the FWCF analysis results.

- One safety/relief valve out-of-service (One SRVOOS) – The EOOS operating limits support operation with one SRVOOS.

- One main steam isolation valve out-of-service (One MSIVOOS) – Operation with one MSIVOOS is supported for operation up to 70 percent rated power operation. Operation with one MSIVOOS has no impact on the other non-pressurization events evaluated to establish power-dependent operating limits. Therefore, the power-dependent operating

limits applicable to base case operation with all MSIVs in service remain applicable for operation with one MSIVOOS for power levels less than or equal to 70 percent of rated. The slow flow runup analyses were performed to support operation with one MSIVOOS.

- Single-loop operation

3.4.2 Core Hydrodynamic Stability

BSEP is currently operating under the requirements of the reactor stability long-term Option III solution approved by the NRC staff in GE licensing topical report, NEDO-32465-A (Reference 28). The stability based operating limit MCPR (OLMCPR) is provided for two conditions as a function of oscillation power range monitor (OPRM) amplitude setpoint as listed in Table 4.3 of Reference 8. The two conditions evaluated are for a postulated oscillation at 45 percent core flow steady-state operation and following a two-recirculation pump trip from the limiting full power operation state point. The Cycle 20 power and flow dependent limits provide adequate protection against violation of SLMCPR for postulated reactor instability as long as the operating limit is greater than or equal to the specific value for the selected OPRM setpoint.

AREVA performed calculations for the relative change in ΔCPR as a function of the calculated hot channel oscillation magnitude (HCOM). These calculations were performed using the RAMONA5-FA code in accordance with Reference 29 methodology. RAMONA5-FA is a coupled neutronic-thermal-hydraulic three-dimensional transient model for determining the relationship between the relative change in ΔCPR and the HCOM on a plant-specific basis. The stability based OLMCPRs were calculated using the most limiting of the calculated change in relative ΔCPR for a given oscillation magnitude or the generic value provided in Reference 28.

In cases where the OPRM system is declared inoperable for BSEP, Unit 2, BSP is provided. BSP curves are evaluated using STAIF (Reference 39) to determine endpoints that meet decay ratio criteria for BSP base minimal Region I (scram region), and BSP base minimal Region II (control entry region). Analyses are performed to support operation with both nominal and reduced feedwater temperature conditions (both FFTR and FHOOS). The BSP endpoints are provided in Table 4.4 of Reference 8.

Based on the information provided by the licensee and discussed above and based on the information presented during the audit in November 2010, the NRC staff finds that the stability analysis and evaluation performed in support of the LAR provides reasonable assurance that the proposed transition in fuel and methods will not adversely impact BSEP ability to satisfy GDC 10 and 12.

3.4.3 Emergency Core Cooling System (ECCS) Performance

The ECCS is designed to mitigate postulated LOCAs caused by ruptures in the primary system coolant piping. The ECCS performance under all LOCA conditions and the evaluation model must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

For a BWR, a LOCA may occur over a wide spectrum of break locations and sizes. Because of significant variations in responses over a break spectrum, 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 LOCA event is separated in to three phases; the blowdown phase, the refill phase, and the reflood phase. 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 the core may become fully or partially uncovered depending on the break size. During the refill phase of a LOCA, the ECCS is functioning and there is a net increase of coolant inventory due to the activation of the core sprays that provide core cooling. The low pressure and high pressure coolant injection systems supply coolant to refill the lower portion of the reactor vessel. During the reflood phase, when the coolant inventory has increased, the cooling is provided above the mixture level by entrained reflood liquid. The ECCS must be designed such that the plant response to a LOCA meets the acceptance criteria specified in 10 CFR 50.46(b).

The evaluation model used for the BSEP LOCA analysis is the NRC-approved EXEM BWR-2000 LOCA analysis methodology described in Reference 20. The EXEM BWR-2000 methodology employs three major computer codes, RELAX, HUXY, and RODEX2, to evaluate the system and fuel response during all phases of a LOCA. RELAX (Reference 20) is used to calculate the system and hot channel response during the blowdown, refill and reflood phases of the LOCA. The HUXY code (Reference 40) code is used to perform heatup calculations for the entire LOCA, and calculates the peak clad temperature (PCT) and local clad oxidation at the axial plane of interest. RODEX2 (Reference 34) is used to determine fuel parameters (such as stored energy) for input to the other LOCA codes. A complete analysis for a given break size starts with the specification of fuel parameters using RODEX2. RODEX2 is then used to determine the initial stored energy for both the blowdown analysis (RELAX hot channel) and the heatup analysis (HUXY). This is accomplished by ensuring that the initial stored energy in RELAX and HUXY is the same or higher than that calculated by RODEX2 for the power, exposure, and fuel design being considered.

The LOCA break spectrum analysis is performed for a full core of ATRUM 10XM fuel. Table 3.2 provides a summary of reactor initial conditions used in the break spectrum analysis.

Table 3.2: Initial Conditions for Break Spectrum Analysis and Heatup Analysis

Parameter	[[ ]]	[[ ]]
Reactor power (% of rated)	102	102
[[ ]]	[[ ]]	[[ ]]
Reactor power (MW(th))	2981.5	2981.5
[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]
Steam flow rate (10^6 lb/hr)	13.1	13.1
Steam dome pressure (psia)	1048.9	1048.7
Core inlet enthalpy (Btu/lb)	527.7	522.4
ATRIUM 10XM hot assembly MAPLHGR (kw/ft)	13.1	13.1
[[ ]]	[[ ]]	[[ ]]
Rod average power distributions	Mid- and Top-Peaked, Figure 4.6*	Mid- and Top-Peaked, Figure 4.7*

*Reference 5

The 2 percent uncertainty increase in rated power is required by 10 CFR Part 50, Appendix K. The analyses are performed at [[

]]

The licensee stated that the break spectrum analyses are applicable to [[

]] The break characteristics identified in the LOCA break spectrum analysis are used in the subsequent fuel type specific LOCA heat up analysis (Enclosure 2 (ANP-2943(P)) to Reference 5) to determine maximum average planar linear heat generation rate (MAPLHGR) limits for the appropriate fuel type. The NRC staff finds that it is reasonable to conclude that the BSEP LOCA break spectrum analysis for full core ATRIUM 10XM fuel can reasonably be applied to the transition cycles.

In conformance with regulatory requirements, the LOCA analyses are performed assuming that all off-site power supplies are lost instantaneously and that only safety grade systems and components are available. In addition, per regulatory requirements the most limiting single failure of ECCS equipment is assumed in the LOCA analysis. The term "most limiting" refers to the ECCS equipment failure that produces the greatest challenge to event acceptance criteria (10 CFR 50.46(b)). The potential limiting single failures identified in the BSEP Updated Final Safety Analysis Report are: DC power (SF-BATT); DC power, diesel generator, low-pressure coolant injection valve (SF-LPCI); and high-pressure coolant injection system. The licensee reviewed the accident scenarios and demonstrated that the SF-BATT and SF-LPCI injection valve failures are limiting failures, as the other single failures result in as much or more ECCS capacity.

The licensee has performed a complete spectrum analysis of break sizes that include double ended guillotine (DEG) with discharge coefficients from 1.0 to 0.4, split breaks with areas between full pipe area and 0.05 ft^2 and break locations (recirculation and non-recirculation pipes). As discussed above, the single failures considered in the recirculation line break analyses are SF-BATT and SF-LPCI.

The results of the LOCA break spectrum analysis show that the limiting recirculation line break is the 0.8 DEG break in the pump discharge piping with an SF-LPCI single failure and top-peaked axial power shape when operating at 102 percent rated core power and [[

]]. Detailed results are provided in Enclosure 1 (ANP-2941(P)) to Reference 5.

For a single loop operation (SLO), a multiplier less than one is applied to the MAPLHGR limits to ensure that the SLO LOCA results are bounded by the two-loop operation LOCA results. In the SLO analysis, the decrease in the MAPLHGR limit is achieved by applying this factor to the radial peaking factor. Local power distributions for the BSEP ATRIUM 10XM neutronic designs are used in the heatup analysis (Enclosure 2 (ANP-2943(P)) to Reference 5). The initial

conditions used for the LOCA heatup analysis are listed in the second column of Table 3.2, with exception that the rod average power distribution is a top-peaked axial as shown in Figure 4.5 of ANP-2943(P).

The EXEM BWR-2000 evaluation model is applied to confirm the acceptability of the ATRIUM 10XM MAPLHGR limit for BSEP, Units 1 and 2. The analysis results are listed below.

The acceptance criteria of 10 CFR 50.46 are met for operation at or below the ATRIUM 10XM MAPLHGR limit specified in Figure 2.1 of ANP-2943(P).

- PCT 1871 °F < 2200 °F.
- Local cladding oxidation thickness 0.99 percent < 17 percent
- Total hydrogen production 0.46 percent <1 percent
- Coolable geometry, satisfied by meeting all of the criteria
- Long term core cooling satisfied by concluding core flooded to top of active fuel or core flooded to the jet pump suction elevation.

The MAPLHGR limit is applicable for ATRIUM 10XM full cores as well as transition cores containing ATRIUM 10XM fuel

The licensee performed BSEP-specific LOCA analysis based on an NRC-approved methodology. The initial conditions, break spectrum, and power profiles selected for LOCA analysis are consistent with the NRC-approved TR, which covers sufficiently limiting scenarios to reach a maximum PCT. The NRC staff finds that the 10 CFR 50.46 acceptance criteria are met and the ECCS performance is acceptable. Based on above, the NRC staff finds the LOCA analyses performed in support of the LAR acceptable.

3.4.4 Anticipated Transient Without Scram (ATWS) Events

An ATWS is defined as an AOO followed by the failure of the scram function of the protection system required by GDC-20. The NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME service level C limit of 1500 pounds per square-inch gauge (psig); (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 °F; (3) the peak suppression pool temperature is less than the design limit (220 °F for BSEP); and (4) the peak containment pressure is less than the containment design pressure (62 psig for BSEP). Since AREVA does not have a generically approved long-term ATWS containment evaluation methodology, the NRC staff reviewed the licensee's long term evaluation for ATRIUM 10XM introduction.

The ATWS overpressurization analyses were performed at 100 percent power at 99 and 104.5 percent flows. 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 ATWS overpressurization analyses are presented in Table 7.3 and Figures 7.5 through 7.8 of Reference 8 show the response of various reactor plant parameters during the limiting PRFO event, the event which results in the maximum vessel pressure. The maximum lower plenum pressure and the maximum dome pressure are both below the ASME limit of 1500 psig. The peak pressure results are adjusted to address NRC's concerns associated with the void-quality correlation and doppler effects. The effects of exposure-dependent thermal conductivity degradation were included in the analysis. The results demonstrate that the ATWS maximum vessel pressure limit of 1500 psig is not exceeded.

Relative to the 10 CFR 50.46 acceptance criteria (i.e., PCT and cladding oxidation), the consequences of an ATWS event are bound by those of the limiting LOCA event. Based on fuel performance analyses conducted for BWR ATWS events, the NRC staff finds that there is reasonable assurance that 10 CFR 50.46 criteria will not be challenged during ATWS, and therefore finds the licensee's conclusion acceptable.

In addition to the short-term vessel overpressure and PCT analysis, the long-term suppression pool performance must be evaluated for acceptability during ATWS. Fuel design differences may impact the power and pressure excursion experienced during the ATWS event. This, in turn, impacts the amount of steam discharged to the suppression pool and containment. The licensee stated that [[

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[[

]] Therefore, it is concluded that the introduction of ATRIUM 10XM fuel [[

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The NRC staff concludes that the licensee has demonstrated that the required systems are installed at BSEP and that they will continue to meet the requirements of 10 CFR 50.62. In addition, the NRC staff reviewed the information submitted by the licensee related to ATWS and concludes that the licensee adequately accounted for the effects of the proposed fuel and methodology transition on ATWS. Therefore, the NRC staff finds the proposed LAR acceptable with respect to ATWS.

3.5 Summary and Conclusion

The NRC staff has reviewed the LAR (Reference 1), in conjunction with the supplemental information (References 2 through 8) and the responses to the staff's RAIs (References 9, 10 and 11), and the licensee's submittal dated April 6, 2011(Reference 41) to evaluate the

acceptability of the BSEP transition to ATRIUM 10XM fuel with AREVA safety analysis and core design methodologies. Based on its review, the NRC staff has determined that the licensee provided adequate technical basis to support the proposed TSs changes. Specifically, the NRC staff finds the licensee has demonstrated that (1) BSEP complies with the staff limitations and conditions imposed for application of the topical reports, (2) AREVA codes and methods are applicable for BSEP (3) the BSEP-specific safety analysis results based on the AREVA methodology meet the applicable licensing criteria, and (4) the proposed TSs changes are acceptable.

4.0 REGULATORY COMMITMENT

The licensee in its letter dated March 16, 2011, responded to the NRC staff's RAI with regard to the localized cladding defects (e.g., spallation, hydride blisters) that could impact fuel rod stress and strain calculations and ultimately the ability to accurately predict cladding failure, and committed to the following regulatory commitment.

Commitment	Due Date
When using AREVA topical report, BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, April 2008 to determine core operating limits, the fuel cladding peak oxide thickness calculated by the RODEX4 corrosion model will be limited to less than the proprietary value defined in Section 2.2 of AREVA report, ANP-2992P, Revision 0, "AREVA Response to Additional RAI on the Brunswick RODEX4 LAR."	Upon implementation of the Unit 1 and Unit 2 license amendments authorizing the incorporation of AREVA topical report, BAW-10247PA into TS 5.6.5.b.

The proprietary limit for the fuel cladding peak oxide thickness calculated by the RODEX4 corrosion model that is defined in Section 2.2 of the AREVA report, ANP-2992P, Enclosure 3 to Reference 11, is **[[ ]]**, as described in Section 3.1.1.1 of this SE. The NRC staff reviewed this commitment and determined that it does not need to be a regulatory requirement.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (75FR 48373; August 10, 2010). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter BSEP 10-0057 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Request for License Amendments – Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, Core Operating Limits Report (COLR)," April 29, 2010.

2. Letter BSEP 10-0071 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Brunswick Unit 2 Cycle 20 Fuel Cycle Design Report ANP-2936(P)," June 9, 2010.

3. Letter BSEP 10-0083 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Additional Information Supporting Requests for License Amendments – Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," July 22, 2010.

4. Letter BSEP 10-0093 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Brunswick Unit 2 Cycle 20 Thermal-Hydraulic Design Report," July 29, 2010.

5. Letter BSEP 10-0112 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Brunswick Unit 2 Cycle 20 Loss of Coolant Accident Analysis Reports, (a) ANP-2941(P) Revision 0 "Brunswick Units 1 and 2 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel," (b) ANP-2943(P) Revision 0, "Brunswick Units 1 and 2 LOCA-ECCS Analysis MAPLHGR Limit for ATRIUM 10XM Duel," September 29, 2010.

6. Letter BSEP 10-0118 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Additional Information Supporting License Amendment Requests – ATRIUM 10XM Fuel Rod Thermal and Mechanical Evaluation for Brunswick Unit 2 Cycle 20," (AREVA Report ANP-2950(P)), October 12, 2010.

7. Letter BSEP 10-0126 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Additional Information Supporting License Amendment Requests – Mechanical Design Report for Brunswick ATRIUM 10XM Fuel Assemblies (AREVA Report ANP-2948(P), Revision 0), November 9, 2010.

8. Letter BSEP 10-0126 from Michael J. Annacone to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Additional Information Supporting License Amendment Requests – Brunswick Unit 2 Cycle 20 Reload safety Analysis," (AREVA Report ANP-2956(P), Revision 0), November 9, 2010.

9. Letter BSEP 10-0133 from William Jefferson, Jr. to NRC, "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Response to Additional Information Report Supporting License Amendment Requests for Analytical Methodology Topical Reports to Technical Specification 5.6.5 (NRC TAC Nos. ME3856, ME3857, ME3858, and ME3859), November 18, 2010.

10. Letter BSEP 10-0141 from William Jefferson, Jr. to NRC "Brunswick Steam Electric Plant, Units 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Response to Additional Information Report Supporting License Amendment Requests for Analytical Methodology Topical Reports to Technical Specification 5.6.5 (NRC TAC Nos. ME3856, ME3857, ME3858, and ME3859), December 16, 2010.

11. Letter BSEP 11-0031 from Michael Annacone to NRC, "Brunswick Steam Electric Plant, Units1 and 2, Response to Request for Additional Information Regarding License Amendment Request for Additional Methodology Topical Reports to Technical Specification 5.6.5," March 16, 2011.

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Principal Contributors: Mathew Panicker
Shih-Liang Wu

Date: April 8, 2011

A notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 256 to License No. DPR-71
2. Amendment No. 284 to License No. DPR-62
3. Safety Evaluation (Nonproprietary Information)
4. Safety Evaluation (Proprietary Information)

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