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

March 1, 2013

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

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING ADDITION OF ANALYTICAL METHODOLOGY TOPICAL REPORTS TO TECHNICAL SPECIFICATION 5.6.5 AND REVISION TO MINIMUM CRITICAL POWER RATIO SAFETY LIMIT (TAC NOS. ME8135 AND ME8136)

Dear Mr. Annacone:

The Commission has issued the enclosed Amendment No. 262 to Renewed Facility Operating License No. DPR-71 and Amendment No. 290 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2. The amendments are in response to your application dated March 6, 2012, as supplemented by letters dated August 29, 2012, September 21, 2012, November 29, 2012, and January 22, 2013.

The amendments, (1) revise Technical Specification (TS) 5.6.5.b by replacing AREVA Topical Report ANF-524(P)(A), *ANF Critical Power Methodology for Boiling Water Reactors* with AREVA Topical Report ANP- I 0307PA. Revision 0, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,* June 2011, in the list of analytical methods that have been reviewed and approved by the U.S. Nuclear Regulatory Commission for determining core operating limits, (2) revise TS 2.1.1, "Reactor Core SLs [Safety Limits]," by incorporating revised Safety Limit Minimum Critical Power Ratio (SLMCPR) values, and (3) revise the license condition in Appendix B, "Additional Conditions," of the operating licenses regarding an alternate method for evaluating SLMCPR values.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* Notice.

Sincerely,

Araceli T. Billoch Colón for

Christopher Gratton, 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. 262 to DPR-71
2. Amendment No. 290 to DPR-62
3. Safety Evaluation

cc w/enclosures: 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. 262
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), dated March 6, 2012, as supplemented by letters dated August 29, 2012, September 21, 2012, November 29, 2012, and January 22, 2013, 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 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 262 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. The license is also amended by changes to the Additional Conditions contained in Appendix B, as indicated in the attachment to this license amendment; and Section 3 of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   3. Additional Conditions

   The Additional Conditions contained in Appendix B, as revised through Amendment No. 262 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions

4. This license amendment is effective as of the date of its issuance and shall be implemented prior to the startup from the 2014 Unit 1 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Jessie F. Quichocho, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: March 1, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 262

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace Pages 4 and 8 of Renewed Operating License DPR-71 with the attached Pages 4 and 8.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
2.0-1	2.0-1
5.0-22	5.0-22

Replace the following page of the Appendix B, "Additional Conditions," 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
App. B-2	App. B-2

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 262 , 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.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 262, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 1 - Technical Specifications - Appendices A and B

Date of Issuance: June 26, 2006

2.0 SAFETY LIMITS (SLS)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

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

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

(continued)

Amendment Number	Additional Conditions	Implementation Date
262	Safety Limit Minimum Critical Power Ratio (SLMCPR), setpoint, and core operating limit values determined using the ANP-10298PA, ACE/ATRIUM 10XM Critical Power Correlation (i.e., TS 5.6.5.b.21), shall be evaluated to verify the values determined using the NRC-approved method remain applicable and the core operating limits include margin sufficient to bound the effects of the K-factor calculation issue described in AREVA Operability Assessment CR 2011-2274, Revision 1. SLMCPR shall be evaluated with methods described in AREVA Document ANP-3086(P), Revision 0, *Brunswick Unit 1 and Unit 2 SLMCPR Operability Assessment Critical Power Correlation for ATRIUM 10XM Fuel – Improved K-factor Model*. Setpoint and core operating limit values shall be evaluated with methods described in AREVA Operability Assessment CR 2011-2274, Revision 1. The results of the evaluation shall be documented and submitted to the NRC, for review, at least 60 days prior to startup of each operating cycle.	Upon implementation of Amendment No. 262
	The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors* (i.e., TS 5.6.5.b.11) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined.	Upon implementation of Amendment No. 262

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. 290
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), dated March 6, 2012, as supplemented by letters dated August 29, 2012, September 21, 2012, November 29, 2012, and January 22, 2013, 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 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 290 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. The license is also amended by changes to the Additional Conditions contained in Appendix B, as indicated in the attachment to this license amendment; and Section 3 of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

   3. Additional Conditions

   The Additional Conditions contained in Appendix B, as revised through Amendment No. 290 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions

4. This license amendment is effective as of the date of its issuance and shall be implemented prior to the startup from the 2013 Unit 2 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Jessie Quichocho, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: March 1, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 290

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace Pages 3 and 8 of Renewed Operating License DPR-62 with the attached Pages 3 and 8.

Replace the following pages of the Appendix A, Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
2.0-1	2.0-1
5.0-21	5.0-21

Replace the following page of the Appendix B, "Additional Conditions," 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
App. B-2	App. B-2
- -	App. B-3

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. 290 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 2 - Technical Specifications - Appendices A and B

Date of Issuance: June 26, 2006

2.0 SAFETY LIMITS (SLS)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

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

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications.

(continued)

Amendment Number	Additional Conditions	Implementation Date
276	Upon implementation of Amendment No. 276 adopting TSTF–448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.3.3, in accordance with TS 5.5.13.c.(i), the assessment of CRE habitability as required by Specification 5.5.13.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.13.d, shall be considered met. Following implementation: (a) The first performance of SR 3.7.3.3, in accordance with Specification 5.5.13.c.(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from June 11, 2004, the date of the most recent successful tracer gas test. (b) The first performance of the periodic assessment of CRE habitability, Specification 5.5.13.c.(ii), shall be within the next 9 months. (c) The first performance of the periodic measurement of CRE pressure, Specification 5.5.13.d, shall be within 18 months, plus the 138 days allowed by SR 3.0.2, as measured from the date of the most recent successful pressure measurement test.	As described in paragraphs (a), (b), and (c) of this Additional Condition.
290	Safety Limit Minimum Critical Power Ratio (SLMCPR), setpoint, and core operating limit values determined using the ANP-10298PA, ACE/ATRIUM 10XM Critical Power Correlation (i.e., TS 5.6.5.b.21), shall be evaluated to verify the values determined using the NRC-approved method remain applicable and the core operating limits include margin sufficient to bound the effects of the K-factor calculation issue described in AREVA Operability Assessment CR 2011-2274, Revision 1. SLMCPR shall be evaluated with methods described in AREVA Document ANP-3086(P), Revision 0, *Brunswick Unit 1 and Unit 2 SLMCPR Operability Assessment Critical Power Correlation for ATRIUM 10XM Fuel – Improved K-factor Model.* Setpoint and core operating limit values shall be evaluated with methods described in AREVA Operability Assessment CR 2011-2274, Revision 1. The results of the evaluation shall be documented and submitted to the NRC, for review, at least 60 days prior to startup of each operating cycle.	Upon implementation of Amendment No. 290

Amendment Number	Additional Conditions	Implementation Date
290	The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors* (i.e., TS 5.6.5.b.11) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined.	Upon implementation of Amendment No. 290

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 262 AND 290

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 letter dated March 6, 2012, Progress Energy, the licensee for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2, submitted a license amendment request to revise the Technical Specifications (TSs) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12076A062). The request was supplemented by letters dated August 29, 2012 (ADAMS Accession No. ML12251A142), September 21, 2012 (ADAMS Accession No. ML12277A344), November 29, 2012 (ADAMS Accession No. ML12348A015), and January 22, 2013 (ADAMS Accession No. ML13031A010).

The proposed amendment updates the topical report used to determine the safety limit minimum critical power ratio (SLMCPR) from AREVA Topical Report ANF-524(P)(A), *ANF Critical Power Methodology for Boiling Water Reactors* (proprietary) with AREVA Topical Report ANP-10307PA, Revision 0, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors*, June 2011 (proprietary). This topical report updates the calculation to use SAFLIM3D from the previous code version of SAFLIM2. The new code version provides for a more accurate calculation of the SLMCPR. The licensee will update its TSs to incorporate the revised SLMCPR values and will revise the license condition in Appendix B, "Additional Conditions," regarding a modification to the generically approved method for evaluating the SLMCPR.

The January 22, 2013, supplemental letter also notified the NRC of the following:

> While performing analyses to support preparation of [supplemental] reports... AREVA identified application errors associated with SAFLIM3D analyses which were submitted as part of CP&L's [Carolina Power & Light Company's or the licensee's] March 6, 2012, license amendment request... These application errors do not affect the proposed Safety Limit Minimum Critical Power Ratio values, nor do the application errors affect the generic NRC-approved methodology described in the SAFLIM3D licensing topical report...

Since the errors affect neither the proposed SLMCPR values nor the generic U.S. Nuclear Regulatory Commission (NRC)-approved methodology, the NRC staff determined that the errors

are a minor issue, and their correction did not affect the conclusions presented in this review. The NRC staff notes, though, that the January 22, 2013, supplemental letter included updated copies of five AREVA reports that were enclosed in the original license amendment request letter.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 36(b) states, in part, that TSs will be derived from the analyses and evaluations included in the safety evaluation report, and amendments thereto, submitted pursuant to Section 50.34. Section 50.36(c)(5) states that TSs shall contain Administrative Controls "relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." In addition, Section 50.36(c)(1)(i)(A) requires the establishment of safety limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.

General Design Criterion (GDC) 10, "Reactor Design," of Appendix A, 10 CFR Part 50 states that the reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Section 3.1.2.2.1 of the BSEP Updated Final Safety Analysis Report (UFSAR) states the following, regarding compliance with GDC 10, "Analyses of abnormal operational transients, described in [UFSAR] Chapter 15, show that the minimum critical power ratio (MCPR) is always > 1.0."

The SLMCPR is one of the SAFDLs to which GDC 10 refers. Pertinent to the MCPR, the applicable criteria for meeting the requirements of GDC 10 are discussed in NUREG-0800, *Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*, Chapter 4.4, "Thermal and Hydraulic Design." The limiting value of critical power ratio (CPR) correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

The SLMCPR meets this criterion by taking the CPR at which boiling transition would be expected on a fuel rod (1.0), and adding to it a margin for correlation, fuel design, and plant operating uncertainties. An additional margin, added for the change in critical power ratio due to the limiting anticipated operational occurrence, is then added to the SLMCPR to determine the operating limit minimum critical power ratio (OLMCPR). Although the SLMCPR is a safety limit whose value is limited by the TS, the OLMCPR is a cycle-specific parameter operating limit that is determined on a cycle-by-cycle basis.

Finally, NRC Generic Letter (GL) 1988-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," applies, since the licensee will revise the Core Operating Limits Report (COLR) references list in the TS to reflect the implementation of the new AREVA SLMCPR methodology and its plant-specific modification. Although the SLMCPR will be calculated for each cycle, the specific value must remain below the TS limit.

3.0 TECHNICAL EVALUATION

3.1 Background

The BSEP, Units 1 and 2 cores consist primarily of ATRIUM-10 and ATRIUM 10XM fuel assemblies. At Unit 1, all co-resident, non-AREVA fuel was expected to be removed beginning with Cycle 19 in spring 2012. A similar shift to remove all co-resident, non-AREVA fuel at Unit 2 is expected to begin with Cycle 21 in spring 2013. In connection with the issuance of License Amendment Nos. 257 and 285 to the operating licenses of BSEP, Units 1 and 2 to support transition to ATRIUM 10XM fuel and associated core design methodologies, a license condition was included in Appendix B of the operating licenses for BSEP, Units 1 and 2 (ADAMS Accession No. ML111010234). This license condition was intended to ensure the operating limit for the application of the ACE correlation for ATRIUM 10XM (Reference 5) fuel and appropriately bound the effects of the K-factor calculation issue described in AREVA Operability Assessment Condition Report (CR) 2011-2274, Revision 1. This license condition was appropriate for application with ANF-524(P)(A).

The transition to ANP-10307PA will begin with BSEP Unit 2, Cycle 21, and there are several noteworthy differences between ANF-524(P)(A) and ANP-10307PA. In its letter dated September 21, 2012, the response to NRC Staff Questions 1, 6, and 7, the licensee summarized the differences between this methodology and prior methodologies. These are (1) the ANP-10307 SLMCPR calculation has a more detailed power distribution, and it can be used in the ACE critical power correlation calculations, (2) the new SLMCPR calculation benefits from an expanded coupling with MICROBURN-B2, which results in more precise modeling of the thermal-hydraulic conditions in the fuel assembly, and (3) the new method implements a more realistic channel bow methodology.

The licensee proposed three changes regarding its SLMCPR and the associated methodology. The first proposed change replaces the SLMCPR methodology listed in Technical Specification (TS) 5.6.5.b from ANF-524(P)(A) with AREVA Topical Report ANP-10307PA, Revision 0. ANP-10307PA, Revision 0, was approved by the NRC for use in boiling water reactors by letter dated June 2011 (ADAMS Accession No. ML11140A125). The second change proposed by the licensee will update the values contained in TS 2.1.1.2, "Safety Limit Minimum Critical Power Ratio," to values that reflect implementation of the new method. Finally, the licensee is updating the license condition associated with the SLMCPR calculation to reflect a methodology modification appropriate for the new SLMCPR method. The NRC staff review of each of these topics is discussed in the following sections of this safety evaluation. The staff also performed a more detailed review of the licensee's implementation of the revised channel bow model contained in ANP-10307PA, as discussed in Section 3.4.

3.2 Methodology Update

The licensee proposed to replace ANF-524(P)(A) with AREVA Topical Report ANP-10307PA, Revision 0, in the list of analytical methodologies in TS Section 5.6.5.b that may be used for determining core operating limits.

GL 88-16 provides guidance for implementing a COLR and removing cycle-specific parameter operating limits from TSs. The discussion in the enclosure to GL 88-16 states, in part:

> ...the specific values of [parameter operating] limits may be modified by licensees, without affecting nuclear safety, provided that these changes are determined using an NRC-approved methodology and consistent with all applicable limits of the plant safety analysis...

Based on the fact that the licensee proposes to replace ANF-524(P)(A) with ANP-10307PA, which is NRC-approved, the NRC staff concludes that the proposed revision to TS 5.6.5.b is acceptable. The proposed change is consistent with the guidance contained in GL 88-16.

The remaining elements of the NRC staff review, described in the following sections of this safety evaluation, address the overall applicability of ANP-10307PA for use at BSEP.

3.3 <u>Proposed Change to Safety Limit Minimum Critical Power Ratio</u>

The licensee has proposed to revise TS 2.1.1.2, which specifies the values for the SLMCPR. The current BSEP Unit 1 TS states:

> MCPR shall be ≥ [greater than or equal to] 1.11 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

and the current BSEP Unit 2 TS states:

> MCPR shall be ≥ 1.11 for two recirculation loop operation or ≥ 1.13 for single recirculation look operation.

The proposed amendments will revise the SLMCPR values in TS 2.1.1.2 for two loop operation and single loop operation. The SLMCPR value for two recirculation loop operation is being changed from ≥ 1.11 to ≥ 1.08, and for single recirculation loop operation, the value is being changed from ≥ 1.12 (Unit 1)/1.13 (Unit 2) to ≥ 1.11 (both units).

The licensee states on page 5 of Enclosure 1 to its March 6, 2012, letter, that the assumed limiting initial conditions used in the SLMCPR calculation will bound the as-operated core design. In a request for addition information (RAI), the NRC staff requested that the licensee provide data to demonstrate that the design basis power distribution assumed in the SLMCPR analysis is conservative, thereby explaining this statement.

In Enclosure 1 to its August 29, 2012, supplemental letter, the licensee provided information to explain the relationship between the SLMCPR and the OLMCPR. The SLMCPR is a convolution of uncertainties associated with core and fuel monitoring, design, and manufacturing uncertainties to ensure that there is adequate margin to a calculated CPR to account for these uncertainties, without exceeding the critical power. The OLMCPR is a quantification of the additional CPR margin required for anticipated operational occurrences (AOO), so that when the AOO occurs, the CPR does not fall below the SLMCPR.

The licensee stated that, in order for the core to reach the SLMCPR during an AOO, the AOO would need to be at the OLMCPR at the initiation of the transient. The licensee provided a plot of the cycle design maximum fraction of limiting CPR for several upcoming BSEP cycles to illustrate the steady-state operating margin that is expected to be maintained throughout these cycles.

The NRC staff reviewed the plots and determined that they illustrated a reasonable amount of margin, which serves as a demonstration that possible AOO initial conditions are analyzed with conservative initial conditions. The NRC staff concluded that this information suggests that the analyzed OLMCPR values are conservative, and this is an expression of available margin to the calculated SLMCPR value. Although the information provided in the RAI response does not exactly address the staff's concern, it does quantify available critical power margin during cycle operation, which the staff finds acceptable.

Note that the proposed TS change will apply the same SLMCPR values to both units. The NRC staff asked the licensee to explain the differences between Unit 1 and Unit 2 and how the licensee determined the TS value for each unit in both single- and two-loop operation. In its August 29, 2012, response letter, the licensee stated that the major differences between the system configurations are the core inlet region and the turbine bypass. The difference in the core inlet region is that Unit 1 has a larger orifice diameter than Unit 2 and the difference in the turbine bypass is that Unit 1 has four valves and Unit 2 has ten. The licensee also stated that both of these differences have a small effect on the SLMCPR calculation.

The second part of this staff request concerned the specific values chosen for the TS SLMCPR. Previously, during single-loop operation, Unit 1 and Unit 2 had SLMCPR limits that differed by 0.01. The staff requested this information to determine whether the licensee had applied the updated methodology consistently between both units. In the August 29, 2012, response letter, the licensee listed the ANP-10307PA results for both single- and two-loop operation, for several upcoming BSEP cycles. The requested TS limit for the single-loop SLMCPR is higher, and more conservative, than the calculated results obtained using the ANP-10307PA methodology. Both Unit 1 and Unit 2 request the same, more conservative value for the TS SLMCPR value. The licensee's response showed that the proposed TS SLMCPR value incorporates margin for small cycle-to-cycle operating variations. The chosen SLMCPR values are acceptable, because the licensee's response confirms that, at both units, the TS SLMCPR limit provides margin, and is therefore conservative relative, to cycle-specific values calculated for each unit, despite the minor differences between the units.

Based on the NRC staff review described above, the NRC staff concluded that the proposed TS values for SLMCPR are acceptable. The proposed values were determined in accordance with NRC-approved methodology, and the licensee provided information to demonstrate that the proposed TS values are conservative relative to the cycle-specific values obtained using the new method. The licensee also provided information that the calculated SLMCPR value is conservative relative to the as-operated core designs. Finally, the licensee also addressed the staff's concern regarding differences between the previous values, which were different between the two BSEP units, and are now the same.

3.4 Application of Revised Channel Bow Model

Fuel channel bowing has been shown historically to have an adverse effect on the SLMCPR prediction. This tendency has been attributed to several different causes, including control blade shadow corrosion and differences between fluence gradients on the fuel channel inner and outer surfaces. This issue is discussed in NRC Information Notice (IN) 89-69, "Loss of Thermal Margin Caused by Channel Box Bow," and IN 89-69, Supplement 1, "Shadow Corrosion Resulting in Fuel Channel Bowing." Several notifications in accordance with 10 CFR Part 21

have also been made to the NRC discussing these issues, and the various SLMCPR calculative methods include provisions to address channel bow issues.

Page 5 of Enclosure 1 to the licensee's March 6, 2012, submittal discusses the treatment employed in the SAFLIM-3D method to account for channel bow. The licensee stated that ANP-10307PA incorporates a realistic fuel channel bow model, with uncertainty that is based on AREVA fuel channel measurements. The submittal states that the BSEP fuel is channeled with AREVA Zircaloy-4 fuel channels, and that BSEP has not experienced indications of abnormal channel bow with the AREVA fuel channels.

The staff requested additional information to verify the applicability of the AREVA channel bow model to the BSEP SLMCPR calculations. In response, the licensee provided data, which indicated the extent of the AREVA channel bow database and compared the BSEP predicted channel bow values (Enclosure 1 to August 29, 2012, supplemental letter). The information provided by the licensee included calculated fluence gradients, which are inputs to the mechanistic channel bow model. The data showed that a very small portion of the calculated fluence gradients for the BSEP core designs fell outside the empirical database that is used to verify the channel bow model and associated uncertainties. Because the large majority of the predicted fuel channel fluence gradients are clearly supported by the empirical channel bow data base, the NRC staff determined that the channel as-approved ANP-10307PA bow treatment for SAFLIM-3D is acceptable for use at BSEP, provided that it can be shown that those channels whose fluence gradients exceed the bounds of the measured channel bow database are in non-limiting locations within the core.

The response to the staff's supplemental RAI explains that the empirical channel deflection model is applied well within its valid range; however, it does not show that the mechanistic channel bow model has been empirically verified for channel fast fluence gradients that exceed a certain value. The AREVA supplemental information concluded that data show that uncertainty decreases with increasing fast fluence gradient; however, the staff determined that the data were insufficient to support this conclusion. The scatter in the existing data set indicates that the relation between channel geometry and fluence-induced channel strain is not the only phenomenon affecting the channel bow behavior. Based on these considerations, the staff finds that further data are needed to justify the application of current uncertainties to a broader range of input values.

Since information provided by the licensee demonstrated that certain channel fast fluence gradients were predicted to exceed the value stated above, the NRC staff was unable to conclude that the channel bow model is being applied within its valid range under BSEP's uprated normal operation conditions.

In order to address the NRC staff-identified concern discussed above, the BSEP licensee proposed the following license conditions for Units 1 and 2, by letter dated January 22, 2013:

> The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, *AREVA MCPR Safety Limit Methodology for Boiling Water Reactors* (i.e., TS 5.6.5.b.11) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient bound of the channel measurement database, when applied to channels

with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined.

This license condition provides assurance that, for those channels for which channel fast fluence gradients exceed the validated range of the channel bow model, the thermal-hydraulic performance would retain an adequate amount of margin to thermal-hydraulic limits to account for uncertainties that may exceed the known limits of the existing model.

In Enclosure 2 to its January 22, 2013, supplemental letter, the licensee provided data, which confirmed that the channels with the greatest cross-channel fluence gradients are located at or near the core periphery, in non-limiting, low-power locations. The licensee also provided a phenomenological explanation for this tendency: that the increased neutronic leakage at the core periphery will result in increased fluence gradients. The NRC staff reviewed the supplemental information submitted by the licensee and determined that, because the additional data shows that the fuel bundles with the highest fluence gradients will tend to be in non-limiting locations in the core, the licensee's application of the AREVA realistic channel bow model is acceptable despite that a small number of channels have negative fluence gradients that exceed the measurement database that verifies the model.

Enclosure 2 to the January 22, 2013, supplemental letter, also provided information to demonstrate, based on re-executing CPR calculations with the augmented uncertainties, that there was small change to the percentage of rods expected to experience boiling transition, but that this change was not significant enough to cause the predicted SLMCPR values to change. This information shows that the channel bow uncertainty, when increased, does not significantly affect the calculated SLMCPR value.

In consideration of the proposed license condition and supplemental information, the NRC staff concluded that the increased uncertainty approach will add a sufficient amount of conservatism to the SLMCPR calculation, since (1) the overall SLMCPR calculation is reasonably insensitive to the increased uncertainty value, and (2) the fuel bundles whose channels exceed the channel bow measurement data base tend to be in non-limiting locations in the core. Based on these considerations, the NRC staff determined that the BSEP licensee's application of the AREVA realistic channel bow model is acceptable.

In its March 6, 2012, submittal letter, the licensee also acknowledged that a small number of GE14 fuel assemblies remain in the core in non-limiting positions, but that the uncertainty associated with these channels was conservatively increased based on GE14-specific fuel channel bow data. The NRC staff accepts this approach, because the GE14 fuel is in non-limiting locations that are unlikely to experience transition boiling before other locations, and because the use of GE14-specific data is an appropriate numerical treatment.

3.5 Operability Assessment

The final proposal by the licensee was to change a license condition to incorporate ANP-3086(P), Revision 0, *Brunswick Unit 1 and Unit 2 SLMCPR Operability Assessment Critical Power Correlation for ATRIUM 10XM Fuel Improved K-Factor Model*, as the operability assessment methodology.[1] The proposed revision to the license condition reads as follows:

[1] A generic application to use ANP-10298PA, Revision 0, Supplement 1P, which addresses the same issue, is currently under review by the NRC staff. The staff questioned why the licensee wanted to use a plant-specific

Safety Limit Minimum Critical Power Ratio (SLMCPR), setpoint, and core operating limit values determined using the ANP-10298PA, ACE/ATRIUM 10XM Critical Power Correlation (i.e., TS 5.6.5.b.21), shall be evaluated to verify the values determined using the NRC-approved method remain applicable and the core operating limits include margin sufficient to bound the effects of the K-factor calculated issue described in AREVA Operability Assessment CR 2011-2274, Revision 1. SLMCPR shall be evaluated with methods described in AREVA Document ANP-3086(P), Revision 0, "Brunswick Unit 1 and 2 SLMCPR Operability Assessment Critical Power Correlation for ATRIUM 10XM Fuel – Improved K-factor Model." Setpoint and core operating limit values shall be evaluated with methods described in AREVA Operability Assessment CR 2011-2274, Revision 1. The results of the evaluation shall be documented and submitted to the NRC, for review, at least 60 days prior to startup of each operating cycle.

The license condition references ANP-10298PA (ADAMS Accession No. ML101190042) – the topical report describing the ACE critical power correlation – and AREVA Operability Assessment CR 2011-2274. The ACE critical power correlation is a vital element of the SLMCPR calculation, since it correlates the fuel bundle neutronic and thermal-hydraulic conditions to identify the MCPR, or to determine the number of fuel rods in the core that are subject to boiling transition for a given operating state. CR 2011-2274, also referenced in the license condition, identified an issue with the approved ACE correlation for the ATRIUM 10XM fuel design with regard to the calculation of K-factor[2] within the ACE correlation. As discussed in ANP-3086 (ADAMS Accession No. ML12076A087), AREVA integrates the K-factor along the entire heated length of the fuel assembly. This particular approach can lead to a condition where the local peaking factors in the upper elevations of a fuel assembly can non-physically affect the K-factor, despite that dryout[3] may occur much lower in the bundle.

AREVA performed evaluations on existing fuel cycle designs that used the ACE correlation to determine the impact of the K-factor issue. For certain fuel designs, AREVA determined that the issue may cause the critical power methods to predict a higher location where departure from nucleate boiling will occur than may be actually expected. In its September 21, 2012, response to NRC Question 3, the licensee explained that this tendency, which is nonconservative in safety analysis, occurs with bottom-peaked axial power shapes.

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methodology in their license condition. In Enclosure 1 to the August 29, 2012, supplemental letter, the licensee stated that this plant-specific application was employed, such that a linked amendment with ANP-10298PA was avoided. The licensee's response also indicated that this approach was discussed with NRC staff prior to submission of this request.

[2] The K-factor characterizes the fuel rod power peaking effect on the bundle critical power. A more comprehensive description of the K-factor and its relevance in the AREVA critical power methods can be found in AREVA Topical Report ANP-10249PA (ADAMS Accession No. ML093631415).

[3] Dryout, which may be characterized as a departure from nucleate boiling, is the thermal-hydraulic condition that is predicted to occur when a fuel assembly reaches its critical power. The dryout location is the elevation on the fuel rod or assembly where the departure from nucleate boiling occurs.

ANP-3086 describes improvements to the K-factor method to eliminate the deficiencies found in the axial averaging process. The improved method relies on a more explicit treatment of the K-factor, which is described in detail in Sections 3.0 through 3.2 of ANP-3086.

In the AREVA critical power methods, additive constants are used to distinguish the critical power performance of each fuel rod. The additive constants account for the influence that spacers and bundle geometry characteristics have on the critical power behavior of the individual rods within a bundle. Section 3.3 of ANP-3086 describes the process to determine the additive constants; the additive constants were re-determined when the K-factor approach was revised. In its review, the NRC staff considered Figure 3.5 of ANP-3086, which compares additive constants supporting the original method, and the new additive constants supporting the more explicit method.

The NRC staff requested that the licensee explain the physical significance of the sign convention of the additive constants and clarify whether a trend was discernible between the original and revised additive constants. The licensee explained in its September 21, 2012, response to NRC Question 5, that the sign convention reflects a comparison of predicted to actual critical power performance, based on the particular K-factor approach and the specific characteristics of the particular rod and bundle. The licensee did not indicate that there was a discernible trend, but rather that some K-factors were different, and therefore some of the additive constants changed. The staff determined that this information was acceptable, because it clarified that the additive constants are determined based on a particular K-factor, and a particular trend in any direction is not expected.

Despite that no readily discernible trend in additive constants was identified, Section 3.4 of ANP-3086 discusses the uncertainty assessment for the additive constants. The staff observed that the uncertainty associated with revised additive constants is slightly lower than the previous additive constant uncertainty. In addition, as discussed in Section 3.5, AREVA provided information that showed that the improved K-factor approach and revised additive constants, when used to assess its experimental database, had a slightly higher ratio of predicted-to-measured rods in boiling transition. The slightly lower additive constant uncertainty is an indication that the improved K-factor approach is slightly more accurate, and the increased ratio is an indication that the overall method, when used with the revised K-factor approach, tends to be more conservative. Based on these considerations, the NRC staff determined that the operability assessment approach is acceptable.

3.6 Technical Evaluation Conclusions

The NRC staff has determined, based on the considerations discussed above, that the proposed changes supporting the implementation of the AREVA SAFLIM-3D SLMCPR method, in combination with the improvements to the K-factor method described in ANP-3086, are acceptable. The proposed revisions to TS 5.6.5 are consistent with GL 88-16 guidance, the revised SLMCPR value contained in TS 2.1.1.2 is supported by plant-specific results performed using NRC-approved methods, and those methods have been augmented to reflect limitations in the available data to validate the channel bow model and to address a concern with the generically approved K-factor method.

The NRC staff determined that the use of ANP-10307PA is acceptable for BSEP, and that the proposed SLMCPR values have been so chosen to provide the requisite level of confidence that

the fuel rods will avoid boiling transition during all conditions of normal operation, with margin for anticipated operational occurrences. The proposed changes are therefore consistent with the guidance in SRP Chapter 4.4, and with GDC 10.

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. 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 (75 FR 4114, January 26, 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.

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Date: March 1, 2013

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* Notice.

Sincerely,

/RA by Araceli Billoch for/

Christopher Gratton, 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. 262 to DPR-71
2. Amendment No. 290 to DPR-62
3. Safety Evaluation

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ADAMS Accession No.: ML13037A551 * By memo

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