



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

December 15, 2008

Mr. Benjamin Waldrep, 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 REVISION OF CONTROL ROD NOTCH TESTING FREQUENCY AND CLARIFICATION OF A FREQUENCY EXAMPLE USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NOS. MD8992 AND MD8993)

Dear Mr. Waldrep:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 250 to Renewed Facility Operating License No. DPR-71 and Amendment No. 278 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2, respectively. The amendments are in response to your application dated June 19, 2008, as supplemented by your letter dated October 1, 2008. The amendments (1) revise the technical specification (TS) surveillance requirement frequency in TS 3.1.3, "Control Rod Operability," and (2) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The changes are consistent with the NRC-approved Technical Specification Task Force (TSTF)-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," Revision 1. This TS change was made available by the NRC in the *Federal Register* on November 13, 2007 (72 FR 63935) as part of the consolidated line item improvement process.

A copy of the related safety evaluation is also enclosed. A notice of issuance will be included in the NRC's biweekly *Federal Register* Notice.

Sincerely,

A handwritten signature in black ink, appearing to read "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. 250 to License No. DPR-71
2. Amendment No. 278 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page

Carolina Power & Light Company

**Brunswick Steam Electric Plant,
Units 1 and 2**

cc:

Sandra Spencer, Mayor
City of Southport
201 East Moore Street
Southport, North Carolina 28461

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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. 250
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 June 19, 2008, as supplemented by the letter dated October 1, 2008, 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.

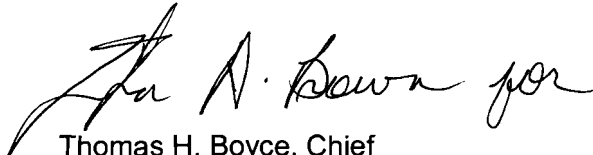
2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 250 , 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "Tom H. Boyce for". The signature is written in a cursive, flowing style.

Thomas H. Boyce, 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: December 15, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 250

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace Page 4 of Renewed Operating License DPR-71 with the attached Page 4.

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

1.4-4
3.1-8
3.1-10
3.1-11
3.1-14

Insert Pages

1.4-4
3.1-8
3.1-10
3.1-11
3.1-14

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 250, 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.

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. -----</p>	
<p>Perform channel adjustment.</p>	<p>7 days</p>

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.3 Perform SR 3.1.3.2 for each withdrawn OPERABLE control rod.</p> <p><u>AND</u></p> <p>A.4 Perform SR 3.1.1.1.</p>	<p>24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM</p> <p>72 hours</p>
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	<p>C.1 -----NOTE----- Inoperable control rod may be bypassed in the RWM or RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -----</p> <p>Fully insert inoperable control rod.</p> <p><u>AND</u></p>	<p>3 hours</p> <p>(continued)</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	<p>-----NOTE----- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -----</p> <p>Insert each withdrawn control rod at least one notch.</p>	31 days
SR 3.1.3.3	Verify each control rod scram time from fully withdrawn to notch position 06 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.4	Verify each control rod does not go to the withdrawn overtravel position.	<p>Each time the control rod is withdrawn to "full out" position</p> <p><u>AND</u></p> <p>Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling</p>

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

-----NOTES-----

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."
2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.3, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES WHEN REACTOR STEAM DOME PRESSURE ≥ 800 psig ^{(a)(b)} (seconds)
46	0.44
36	1.08
26	1.83
06	3.35

- (a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.
- (b) When reactor steam dome pressure is < 800 psig, established scram time limits apply.



UNITED STATES
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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. 278
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 June 19, 2008, as supplemented by the letter dated October 1, 2008, 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.

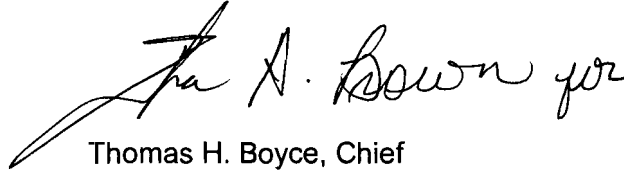
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 278, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "Tom H. Boyce for". The signature is fluid and cursive, with a long horizontal stroke extending to the left.

Thomas H. Boyce, 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: December 15, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 278

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace Page 3 of Renewed Operating License DPR-62 with the attached Page 3.

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

1.4-5
3.1-8
3.1-10
3.1-12
3.1-14

Insert Pages

1.4-4
3.1-8
3.1-10
3.1-11
3.1-14

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 or 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 possess, 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) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 278, are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. -----	
Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours (plus the extension allowed by SR 3.0.2) with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval (plus the extension allowed by SR 3.0.2), there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	<p>A.3 Perform SR 3.1.3.2 for each withdrawn OPERABLE control rod.</p> <p><u>AND</u></p> <p>A.4 Perform SR 3.1.1.1.</p>	<p>24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM</p> <p>72 hours</p>
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	<p>C.1 -----NOTE----- Inoperable control rod may be bypassed in the RWM or RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. -----</p> <p>Fully insert inoperable control rod.</p> <p><u>AND</u></p>	<p>3 hours</p> <p>(continued)</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours
SR 3.1.3.2	<p>-----NOTE----- Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. -----</p> <p>Insert each withdrawn control rod at least one notch.</p>	31 days
SR 3.1.3.3	Verify each control rod scram time from fully withdrawn to notch position 06 is ≤ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.3.4 Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

-----NOTES-----

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."
 2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.3, and are not considered "slow."
-

NOTCH POSITION	SCRAM TIMES WHEN REACTOR STEAM DOME PRESSURE \geq 800 psig ^{(a)(b)} (seconds)
46	0.44
36	1.08
26	1.83
06	3.35

- (a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.
- (b) When reactor steam dome pressure is < 800 psig, established scram time limits apply.



UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 250 AND 278

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 June 19, 2008 (Agencywide Document Access and Management System (ADAMS) Accession No. ML082810503), as supplemented by letter dated October 1, 2008 (ADAMS Accession No. ML081840066), the Carolina Power & Light Company (the licensee) requested amendments to Renewed Operating Licenses DPR-71 and DPR-62 for Brunswick Steam Electric Plant (BSEP), Units 1 and 2, respectively. The proposed amendments would (1) revise the TS surveillance requirement (SR) frequency in TS 3.1.3, "Control Rod Operability," and (2) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The changes are consistent with the U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF)-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," Revision 1. This TS change was made available by the NRC in the *Federal Register* on November 13, 2007 (72 FR 63935) as part of the consolidated line item improvement process (CLIIP).

These changes are based on TSTF-475, Revision 1, that revised the Standard Technical Specifications (STS) by: (1) revising the frequency of SR 3.1.3.2 notch testing of each fully withdrawn control rod from 7 days after the control rod is withdrawn and THERMAL POWER is greater than the Low Power Setpoint (LPSP) of the Rod Worth Minimizer (RWM) to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM" (NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4") and (2) revising Example 1.4-3 in Section 1.4 "Frequency" to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column (NUREG-1433).

The purpose of the surveillances is to confirm control rod insertion capability. This is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and the control rod drive (CRD) Mechanism (CRDM), by which the control rods are moved, are components of the CRD system, which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable with a tapered design of the index tube and is conducive to control rod insertion.

A stuck control rod is an extremely rare event and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test.

The purpose of these revisions is to reduce the number of control rod manipulations and, thereby, reduce the opportunity for reactivity control events.

The purpose of the change to Example 1.4-3 in Section 1.4 "Frequency" is to clarify the applicability of the 25 percent allowance of SR 3.0.2 to time periods discussed in NOTES in the "SURVEILLANCE" column as well as to time periods in the "FREQUENCY" column.

The licensee, in the letters dated June 19 and October 1, 2008, is not proposing any variations or deviations from the applicable TS changes described in the modified TSTF-475, Revision 1 and the NRC staff's model safety evaluation dated November 13, 2007. However, the licensee stated that an administrative change is being made to the Notes associated with SR 3.1.3.3 (the existing SR designation) for Amendment 210 for Unit 1 that was issued on May 23, 2000 (ADAMS Accession No. ML003718232) to modify the SR and include a cycle-specific Note associated with testing of Control Rod 26-47. The licensee stated that this note has expired and, as such, is being removed. The licensee concluded that this change is purely administrative in nature and does not affect the applicability of either the safety evaluation or the no significant hazards consideration determination published in the *Federal Register* as part of the CLIP.

The supplements dated October 1, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 7, 2008 (73 FR 58671).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, General Design Criterion (GDC) 29, Protection against anticipated occurrence, requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences. The design relies on the CRD system to function in conjunction with the protection systems under anticipated operational occurrences, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD system provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during anticipated operational occurrences. Meeting the requirements of GDC 29 for the CRD system prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during anticipated operational occurrences. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

The BSEP, Units 1 and 2 were designed and constructed based on the proposed GDC published by the Atomic Energy Commission (AEC) in the Federal Register (32 FR 10213) on July 11, 1967 (draft GDC). Carolina Power & Light reviewed the differences between the draft GDC and final GDC contained in Appendix A to 10 CFR, Part 50. As discussed in the NRC Staff

Requirements Memorandum for SECY-92-223, the NRC decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. As the BSEP, Units 1 and 2 were licensed before the final GDC were formally adopted, these units were evaluated on a plant-specific basis, determined to be safe, and licensed by the NRC. The draft GDC applicable to these units are maintained in Appendix F of the original approved Final Safety Analyses Report (FSAR). The NRC's "Safety Evaluation of the Brunswick Steam Electric Station Units 1 and 2," dated November 1973, found that the BSEP AEC-1971 criteria meets the intent of the GDC, published in the Federal Register on May 21, 1971, as Appendix A to 10 CFR Part 50. The licensee performed a comparison of between the draft (AEC-1971) and final GDC, which is contained in Section 3.1 of the Updated FSAR.

Draft AEC-1971, GDC 29, requires that the protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. This criterion is comparable to the current GDC 29, "Protection against anticipated occurrence," which requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences.

The design relies on the CRD system to function in conjunction with the protection systems under anticipated operational occurrences, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRD system provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during anticipated operational occurrences. Meeting these requirements for the CRD system prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during anticipated operational occurrences. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

3.0 TECHNICAL EVALUATION

The NRC staff previously reviewed the following information provided by the TSTF to justify the submitted license amendment request to revise the weekly control rod notch frequency to monthly (STS NUREG-1433) and revise the discussion of the applicability of the 25 percent allowance in Example 1.4-3. Specifically, the following documents were reviewed during the NRC staff's evaluation:

- TSTF letter, TSTF-04-07 (Reference 1) - Provided a description of the proposed changes in TSTF-475 that changes the weekly rod notch frequency to monthly and clarified the applicability of the 25 percent allowance in Example 1.4-3.
- TSTF letter, TSTF-06-13 (Reference 2) - Provided responses to NRC staff request for additional information (RAI) on (1) industry experience with identifying stuck rods, (2) tests that would identify stuck rods, (3) continue compliance with General Electric (GE) Services Information Letter (SIL) 139, (4) industry experience on collet failures, and (4) applying the 25 percent grace period to the 31-day control rod notch SR test frequency.

- Boiling Water Reactor Owners Group (BWROG) letter BWROG-06036 (Reference 3) - Provided the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," that evaluated CRD notching frequency and CRD performance.
- TSTF letter TSTF-07-19 (Reference 4) - Provided response to the NRC staff's RAI on CRD performance in Control Cell Core-designed plants, including TSTF 475, Revision 1.

The CRD System at BSEP, Units 1 and 2, is the primary reactivity control system for the reactor. The CRD system, in conjunction with the reactor protection system, provides the means for the reliable control of reactivity changes to ensure under all conditions of normal operation, including anticipated operational occurrences that specified acceptable fuel design limits are not exceeded. Control rods are components of the CRD system that have the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD system.

The CRD System consists of a CRDM by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD that houses the collet mechanism and consists of the locking collet, collet piston, collet return spring and an unlocking cam. The collet mechanism provides the locking/unlocking mechanism that allows the insert/withdraw movement of the control rod. The CRT has three primary functions: (a) to carry the hydraulic unlocking pressure to the collet piston, (b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and (c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

The following paragraphs describe the bases for the staff's approval of TSTF-475:

According to the BWROG, at the time of the first CRT crack discovery in 1975, each partially or fully withdrawn operable control rod was required to be exercised, one notch, at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits through the flow holes into the annulus on the outside. At the same time, hot water

from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through wall and circumferential temperature gradients during scrams that contribute to the observed CRT cracking.

Subsequently, many boiling water reactors (BWRs) have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change for partially withdrawn control rods was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on a weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.

In response to NRC's RAIs and to support their position to reduce the CRD notch testing frequency, the BWROG provided plant data and a GE Nuclear Energy report entitled, "CRD Notching Surveillance Testing for Limerick Generating Station." The GE report provided a description of the cracks noted on the original design CRT surfaces. These cracks, later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable and would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the technical specifications. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks have been observed in the final improved CRT design. Neither the BWROG nor the NRC staff was able to find evidence of a collet housing failure since 1975. To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. On a numerical basis for instance, based on BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6590 rod notch tests for a BWR/4 plant and 9284 for a BWR/6 plant. For example, if all BWRs operating in the U.S. are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

In addition, the IGSCC crack growth rates were evaluated, at Limerick Generating Station, using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

Also, the BWR scram system has extremely high reliability. In addition to notch testing, scram time testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks

and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod.

Also, the HCUs, CRD drives, and control rods are tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives, are inspected and, as required, their internal components are replaced. Therefore, increasing the CRD notch testing frequency to monthly would have very minimal impact on the reliability of the scram system.

The NRC staff reviewed TSTF-475 proposal to amend the (NUREG 1433) TS SR 3.1.3.2, "Control Rod OPERABILITY," from 7 days to monthly. Based on the following evaluation condition: (1) slow crack growth rate of the CRT; (2) the improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) GE chemistry recommendations; and (5) no known CRD failures have been detected during the notch testing exercise, the NRC staff concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRD system. The utilities should consider the replacement of CRT, when possible, with the GE CRT improved design.

The NRC staff has reviewed the TSTF-475 proposal to amend (NUREG-1433) Example 1.4-3 in Section e.4. "Frequency," to make the 1.25 provision in SR 3.0.2 to be equally applicable to time periods specified in the "FREQUENCY" column and in the NOTE in the "SURVEILLANCE" column. The NRC staff finds this change acceptable since the revision would make it consistent with the definition of the specified "Frequency" provided in the second paragraph of Section 1.4 which states that the specified "Frequency" is referred to throughout this sections and each of Specifications of Section 3.0, Surveillance Requirements Applicability. The specified "Frequency" consists of the requirements of the Frequency column of each SR, as well certain Notes in the Surveillance column that modify performance requirements.

The licensee stated in their application that they have reviewed the basis for the NRC's acceptance of TSTF-475, Revision 1 and concluded that the basis is applicable to BSEP, Units 1 and 2, and supports the adoption of the TSTF-475 changes into the TSs of both units. The NRC staff similarly concluded that the basis for TSTF-475 is applicable to BSEP, Units 1 and 2 and, therefore, appropriate for adoption by the licensee. In addition, the NRC staff reviewed the licensee's proposed changes against the corresponding changes made to the STS by TSTF 475, Revision 1, which the staff has found to satisfy applicable regulatory requirements, as described above.

Also, the licensee's letter dated June 19, 2008, states, "an administrative change is being made to the Notes associated with Surveillance Requirement (SR) 3.1.3.3 (i.e., the existing SR designation) for Amendment 210 for Unit 1, issued on May 23, 2000 (i.e., Accession Number ML003718232), modified the SR to include a cycle-specific Note associated with testing of Control Rod 26-47. This note has expired and, as such, is being removed. This change is purely administrative in nature and does not affect the applicability of either the safety evaluation or the no significant hazards consideration determination published in the *Federal Register* as part of the CLIP."

The staff found that the proposed changes are consistent with the changes approved by the NRC in TSTF-475, Revision 1 and, therefore, finds these changes acceptable. The staff also accepts the licensee's administrative change to the SR 3.1.3.3 Note.

3.1 Technical Conclusion

The NRC staff has reviewed the licensee's proposal to amend existing BSEP, Units 1 and 2 TSs: (1) revise the TS surveillance requirement frequency in TS 3.1.3, "Control Rod Operability," and (2) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff has concluded that the TS revisions will have a minimal effect on the high reliability of the CRD system while reducing the opportunity for potential reactivity events; thus, meeting the requirement of 10 CFR Part 50, Appendix A, GDC 29, and will clarify the applicability of the 1.25 provision in SR 3.0.2. Therefore, the staff concludes that the amendment request is acceptable. The staff also concurs with the licensee's proposed administrative change to the SR 3.1.3.3 Note, since it does not affect the applicability of either the safety evaluation or the no significant hazards consideration determination published in the *Federal Register* as part of the CLIP.

4.0 STATE CONSULTATION

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

5.0 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 NRC has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (73 FR 58671; October 7, 2008). 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.

6.0 CONCLUSION

The NRC staff 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) such activities will be conducted in compliance with the NRC'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.

7.0 REFERENCES

1. TSTF letter, TSTF-04-07, "TSTF-475, Revision 0, Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," August 30, 2004, ADAMS Accession No. ML042520035.
2. TSTF letter, TSTF-07-19, "Response to NRC Request for Additional Information (RAI) Regarding TSTF-475 Revision 0, Control Rod Notch Testing Frequency and SRM Insert Control Rod Action, dated February 28, 2007," May 22, 2007, ADAMS Accession No. ML071420428
3. TSTF letter, TSTF-06-13, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0, Control Rod Notch Testing Frequency and SRM Insert Control Rod Action dated March 21, 2005," July 3, 2006, ADAMS Accession No. ML061840342
4. BWROG letter, BWROG-06036, "Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0," Control Rod Notch Testing Frequency and SRM Insert Control Rod Action dated March 21, 2005," November 16, 2006, ADAMS Accession No. ML063250258

Principal Contributor: Ravi Grover

Date: December 15, 2008

December 15, 2008

Mr. Benjamin Waldrep, 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 REVISION OF CONTROL ROD NOTCH TESTING FREQUENCY AND CLARIFICATION OF A FREQUENCY EXAMPLE USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (TAC NOS. MD8992 AND MD8993)

Dear Mr. Waldrep:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 250 to Renewed Facility Operating License No. DPR-71 and Amendment No. 278 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2, respectively. The amendments are in response to your application dated June 19, 2008, as supplemented by your letter dated October 1, 2008. The amendments (1) revise the technical specification (TS) surveillance requirement frequency in TS 3.1.3, "Control Rod Operability," and (2) revise Example 1.4-3 in Section 1.4, "Frequency," to clarify the applicability of the 1.25 surveillance test interval extension. The changes are consistent with the NRC-approved Technical Specification Task Force (TSTF)-475, "Control Rod Notch Testing Frequency and SRM [Source Range Monitor] Insert Control Rod Action," Revision 1. This TS change was made available by the NRC in the *Federal Register* on November 13, 2007 (72 FR 63935) as part of the consolidated line item improvement process.

A copy of the related safety evaluation is also enclosed. A notice of issuance will be included in the NRC'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. 250 to License No. DPR-71
2. Amendment No. 278 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures: See next page

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NRR-058

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