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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
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 July 20, 1990, 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 Facility Operating License No. DPR-71 is hereby amended to read as follows:

9101020325 901220
PDR ADOCK 05000324
P PDR

December 20, 1990

Docket Nos. 50-325
and 50-324

DISTRIBUTION
See attached list

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. DPR-71 AND AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-62 REGARDING IGSCC REQUIREMENTS IN ACCORDANCE WITH GENERIC LETTER 88-01 - BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2, (TAC NOS. 77250 AND 77251)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 150 to Facility Operating License No. DPR-71 and Amendment No. 180 to Facility Operating License No. DPR-62 for Brunswick Steam Electric Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your submittal dated July 20, 1990.

The amendments change the Technical Specifications to comply with Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." The proposed changes are to the Surveillance Requirements.

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

Sincerely,

Original Signed By:

Ngoc B. Le, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

9101020317 901220
PDR ADOCK 05000324
P PDR

Enclosures:
1. Amendment No. 150 to License No. DPR-71
2. Amendment No. 180 to License No. DPR-62
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PD21:DRPR	:PE:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:
NAME	:PAnderson	:DSpaulding:sw	:NLe	: EAdensam	:	:	:
DATE	:11/26/90	:11/27/90	:11/27/90	:12/5/90	:	:	:

OFFICIAL RECORD COPY

DF01 1/1

Mr. L. W. Eury
Carolina Power & Light Company

Brunswick Steam Electric Plant
Units 1 and 2

cc:

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Vice President
Brunswick Nuclear Project
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Southport, North Carolina 28461

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Carolina Power & Light Company
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Mr. H. A. Cole
Special Deputy Attorney General
State of North Carolina
P. O. Box 629
Raleigh, North Carolina 27602

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
P. O. Box 29520
Raleigh, North Carolina 27626-0520

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 150 , 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronnie H. Lo/for

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 20, 1990

SUBJECT TO CORRECTION OF CHANGE PAGES.

OFC	:LA:PD21:DRPR	:PE:PD21:DRPR	:PM:PD21:DRPR	:EMCB:NRR	: OGC	:D:PD21:DRPR	:	:	:
NAME	: PAnderson	:DSpaulding:sw	:NLe	:CYCheng	: E Hollen	:EAdensam	:	:	
DATE	:11/28/90	:12/5/90	:12/4/90	:12/6/90	:12/13/90	:12/18/90	:		

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ATTACHMENT TO LICENSE AMENDMENT NO. 150

FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 0-2	3/4 0-2
3/4 0-3	3/4 0-3
3/4 4-6	3/4 4-6
B 3/4 0-3	B 3/4 0-3
B 3/4 4-2	B 3/4 4-2

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL CONDITIONS or other states specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable state shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

3/4.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods & personnel & sample expansion inlcuded in this letter.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE averaged over any 24-hour period.

c. 25 gpm total leakage averaged over any 24-hour period.

d. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24-hour period except for the first 24 hours of reactor startup commencing with entry into OPERATIONAL CONDITION 2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within the limits within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell and equipment drain sump flow rates at least once per 8 hours, and

b. Monitoring the primary containment atmosphere particulate and gaseous radioactivity at least once per 24 hours.

APPLICABILITY

BASES

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these technical specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And, for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

This specification includes a statement that the Inservice Inspection Program for pipe covered by the scope of GL 88-01 will be in conformance with the staff positions on schedule, methods and personnel, and sample expansion included in GL 88-01. This requirement may be removed form the technical specifications in the future in line with the Technical Specification Improvement programs.

These specifications are based on the guidance of General Electric SIL #380, Rev. 1, 2-10-84.

3/4.4.2 SAFETY/RELIEF VALVES

The reactor coolant system safety valve function of the safety-relief valves operate to prevent the system from being pressurized above the Safety Limit of 1325 psig. The system is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code Section III for the pressure vessel and ANSI B31.1, 1975, Code for the reactor coolant system piping.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE
3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action. Monitoring leakage at eight hour intervals is in conformance with the 12/21/89 NRC SER for GL 88-01.

3/4.4.4 CHEMISTRY

The reactor water chemistry limits are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus, the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides, and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity outside the limits, additional samples must be examined to ensure that the chlorides are not exceeding the limits.

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
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 July 20, 1990, 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 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. 180 , 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 of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronnie H. Lo/for

Elinor G. Adensam, Director
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 20, 1990

OFC	LA:PD21:DRPR	PE:PD21:DRPR	PM:PD21:DRPR	EMCB:NRR	OGC	D:PD21:DRPR		
NAME	PAnderson	DSpaulding	NLe	CYCheng	E. Holler	EAdensam		
DATE	11/26/90	11/27/90	11/27/90	11/27/90	12/13/90	12/11/90		

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ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 0-2	3/4 0-2
3/4 0-3	3/4 0-3
3/4 4-6	3/4 4-6
B 3/4 0-3	B 3/4 0-3
B 3/4 4-2	B 3/4 4-2

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL CONDITIONS or other states specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL CONDITION or other specified applicable state shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g) (6) (i).

3/4.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods & personnel & sample expansion inlcuded in this letter.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE averaged over any 24-hour period.

c. 25 gpm total leakage averaged over any 24-hour period.

d. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24-hour period except for the first 24 hours of reactor startup commencing with entry into OPERATIONAL CONDITION 2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within the limits within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell and equipment drain sump flow rates at least once per 8 hours, and

b. Monitoring the primary containment atmosphere particulate and gaseous radioactivity at least once per 24 hours.

APPLICABILITY

BASES

Under the terms of this specification, for example, during initial plant start-up or following extended plant outage, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status. Exceptions to some surveillance activities have been provided for in individual specifications.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these technical specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And, for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

This specification includes a statement that the Inservice Inspection Program for pipe covered by the scope of GL 88-01 will be in conformance with the staff positions on schedule, methods and personnel, and sample expansion included in GL 88-01. This requirement may be removed form the technical specifications in the future in line with the Technical Specification Improvement programs.

These specifications are based on the guidance of General Electric SIL #380, Rev. 1, 2-10-84.

3/4.4.2 SAFETY/RELIEF VALVES

The reactor coolant system safety valve function of the safety-relief valves operate to prevent the system from being pressurized above the Safety Limit of 1325 psig. The system is designed to meet the requirements of the ASME Boiler and Pressure Vessel Code Section III for the pressure vessel and ANSI B31.1, 1967, Code for the reactor coolant system piping.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE
3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action. Monitoring leakage at eight hour intervals is in conformance with the 12/21/89 NRC SER for GL 88-01.

3/4.4.4 CHEMISTRY

The reactor water chemistry limits are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low; thus, the higher limit on chlorides is permitted during full power operation. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides, and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity outside the limits, additional samples must be examined to ensure that the chlorides are not exceeding the limits.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. DPR-71

AND AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. 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 July 20, 1990, Carolina Power & Light Company submitted a request for changes to the Brunswick Steam Electric Plant, Units 1 and 2.

The amendments change the Surveillance Requirements of the Technical Specifications (TS) to comply with Generic Letter 88-01, "NRC Staff Position on IGSCC in BWR Austenitic Stainless Steel Piping." This Generic Letter outlines the NRC's position on intergranular stress corrosion cracking (IGSCC) in boiling water reactor austenitic stainless steel piping.

2.0 EVALUATION

One of the proposed changes is to delete Surveillance Requirement 4.0.5.a.1 and to renumber Surveillance Requirement 4.0.5.a.2 as Surveillance Requirement 4.0.5.a. The Surveillance Requirement 4.0.5.a.1 was only applicable during the time period from issuance of the Facility Operating License to the start of facility commercial operation. Both units have been in commercial operation for several years, therefore, 4.0.5.a.1 is no longer applicable. This change does not impact on the levels of safety that currently exist.

The second proposed change requested that Surveillance Requirement 4.0.6 be deleted and that a new Surveillance Requirement 4.0.5.f be added. Generic Letter 88-01 indicates that the technical specification should include a statement that the in-service inspection program for piping covered by the scope of the Generic Letter will be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion. The new Surveillance Requirement 4.0.5.f references the new guidance provided in Generic Letter 88-01 and incorporates requirements which supersede those provided in Surveillance Requirement 4.0.6. Therefore, the deletion of Surveillance Requirement 4.0.6 from the TS and addition of Surveillance Requirement 4.0.5.f does not impact on plant safety or operations.

The third proposed change requested that the reactor coolant system (RCS) leakage monitoring frequency, in Surveillance Requirement 4.4.3.2.a, be increased from every 24 hours to every 8 hours. This revision is in compliance with the NRC staff position on leakage monitoring as stated in Generic Letter 88-01. The proposed revision does not involve a physical change or alteration to the facility, therefore, there is no probability of reduction in safety as presently realized. There is no change in the method in which leakage is monitored, and the revision of the monitoring intervals will not impact on the level of safety currently seen. Therefore, this change in monitoring frequency is acceptable

3.0 ENVIRONMENTAL CONSIDERATIONS

These amendments change a requirement with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and change the surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released off site and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, these 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 these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the Federal Register (55 FR 34365) on August 22, 1990, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The 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 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.

Dated: December 20, 1990

Principal Contributors: D. Spaulding
N. Le

AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. DPR-71 - BRUNSWICK, UNIT 1
AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-62 - BRUNSWICK, UNIT 2

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
N. Le
D. Spaulding
R. Clark
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
G. Hill (8) (P1-137)
Wanda Jones (P-130A)
J. Calvo (11D3)
C. Cheng (7-D-4)
ACRS (10)
GPA/PA
OC/LFMB
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cc: Brunswick Service List

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