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. 119
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 13, 1987, as supplemented January 6, 1988, March 10, 1988, April 6, 1988 and July 12, 1988 , 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:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154, 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

/S/

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 6, 1988

OFC	LA:PD21:DRPR	PE:PD21:DRPR	PM:PD21:DRPR	OGC	D:PD21:DRPR		
NAME	PAnderson	BMozafari:ch	BBuckley		EAdensam		
DATE	9/9/88	9/9/88	9/12/88	9/29/88	10/6/88		

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ATTACHMENT TO LICENSE AMENDMENT NO. 119

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 4-2	3/4 4-2
B3/4 4-1	B3/4 4-1
B3/4 4-2	B3/4 4-2
B3/4 4-3	B3/4 4-3

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1 and 2

ACTION:

With less than 20 jet pumps OPERABLE, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 Each of the above required jet pumps shall be demonstrated OPERABLE after entering OPERATIONAL CONDITION 1 but prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours, thereafter, by verifying that at least 2 of the following conditions do not occur:

a. The recirculation pump flow differs by more than 5% from the established speed-flow characteristics.

b. The jet pump loop flow differs by more than 5% from the established speed-flow characteristics.

c. The diffuser-to-lower plenum differential pressure reading on any individual jet pump varies by more than 10% from that jet pump's established operating characteristics.

4.4.1.2.2 Each of the above required jet pumps shall be demonstrated operable prior to entering OPERATIONAL CONDITION 2 and at least once per 24 hours, thereafter, with THERMAL POWER less than or equal to 25% of RATED THERMAL POWER by verifying the diffuser-to-lower plenum differential pressure reading is within the established normal operating characteristics of the jet pump.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation for longer than 24 hours with a reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design basis accident by increasing the blowdown area and eliminating the capability of reflooding the core--thus, the requirement for shutdown of the facility with a jet pump inoperable.

The established characteristics for the criteria of 4.4.1.2 are bands of values that encompass the normal scatter of the data. The scatter in the data can be attributed primarily to monitoring inaccuracies and instrumentation tolerances. An evaluation of these factors will be used to determine the widths of the bands. The bands will be centered about the expected values determined from operating data. The acceptance criteria will be these bands plus the appropriate percentage of the expected values at any point. The acceptance criteria will be updated as required to reflect any changes in the recirculation system that would affect the monitored variables.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within 50°F of each other prior to start-up of an idle loop.

Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than 145°F.

Neutron flux noise limits are established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1 to 12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels significantly larger than these values are considered in the thermal/mechanical fuel design and are found to be of negligible consequence. In addition, stability tests at operating BWR's have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles 5 to 10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Data to establish baseline APRM and LPRM neutron flux noise values is obtained at a point below the 100% rated rod line. A minimum of two detectors of one LPRM string per core octant and two detectors of one LPRM string near the center of the core should be monitored. Detectors used for monitoring should be selected to provide core wide representation. Substitutions are permitted for inoperable LPRM detectors.

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 operates 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.

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.

REACTOR COOLANT SYSTEM

BASES

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. Permitting operation to continue for limited time periods with higher specific activity levels accommodates short-term iodine spikes which may be associated with power level changes, and is based on the fact that a steam line failure during these short time periods is considerably less likely. Operation at the higher activity levels, therefore, is restricted to a small fraction of the unit's total operating time. The upper limit of coolant iodine concentration during short-term iodine spikes ensures that the thyroid dose from a steam line failure will not exceed 10 CFR Part 100 dose guidelines.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should the steam line rupture occur. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and start-up and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During start-up and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 119, 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

/S/

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 6, 1988

OFC	:LA:PD21:DRPR	:PE:PD21:DRPR	:PM:PD21:DRPR	: OGC	:D:PD21:DRPR	:	:
NAME	: PAnderson	:BMozafari:ch	: BBuckley	:	: EAdensam	:	:
DATE	:9/9/88	: 9/9/88	: 9/12/88	: 9/23/88	: 10/6/88	:	:

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UNITED STATES
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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. 154
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 13, 1987, as supplemented January 6, 1988, March 10, 1988, April 6, 1988 and July 12, 1988, 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. 154, 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

Elinor G. Adensam

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

Attachment:
Changes to the Technical
Specification.

Date of Issuance: October 6, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 154

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 4-2	3/4 4-2
B3/4 4-1	B3/4 4-1
B3/4 4-2	B3/4 4-2
B3/4 4-3	B3/4 4-3

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1 and 2

ACTION:

With less than 20 jet pumps OPERABLE, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 Each of the above required jet pumps shall be demonstrated OPERABLE after entering OPERATIONAL CONDITION 1 but prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours, thereafter, by verifying that at least 2 of the following conditions do not occur:

a. The recirculation pump flow differs by more than 5% from the established speed-flow characteristics.

b. The jet pump loop flow differs by more than 5% from the established speed-flow characteristics.

c. The diffuser-to-lower plenum differential pressure reading on any individual jet pump varies by more than 10% from that jet pump's established operating characteristics.

4.4.1.2.2 Each of the above required jet pumps shall be demonstrated operable prior to entering OPERATIONAL CONDITION 2 and at least once per 24 hours, thereafter, with THERMAL POWER less than or equal to 25% of RATED THERMAL POWER by verifying the diffuser-to-lower plenum differential pressure reading is within the established normal operating characteristics of the jet pump.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with a reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does present a hazard in case of a design basis accident by increasing the blowdown area and eliminating the capability of reflooding the core. Thus, the requirement for shutdown of the facility with a jet pump inoperable.

The established characteristics for the criteria of 4.4.1.2 are bands of values that encompass the normal scatter of the data. The scatter in the data can be attributed primarily to monitoring inaccuracies and instrumentation tolerances. An evaluation of these factors will be used to determine the widths of the bands. The bands will be centered about the expected values determined from operating data. The acceptance criteria will be these bands plus the appropriate percentage of the expected values at any point. The acceptance criteria will be updated as required to reflect any changes in the recirculation system that would affect the monitored variables.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within 50°F of each other prior to start-up of an idle loop.

Since the coolant in the bottom of the vessel is at a lower temperature than the water in the upper regions of the core, undue stress on the vessel would result if the temperature difference were greater than 145°F.

Neutron flux noise limits are established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1 to 12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels significantly larger than these values are considered in the thermal/mechanical fuel design and are found to be of negligible consequence. In addition, stability tests at operating BWR's have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles 5 to 10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

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REACTOR COOLANT SYSTEM

BASES

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

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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

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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.

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.

REACTOR COOLANT SYSTEM

BASES

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3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines in 10CFR 100. Permitting operation to continue for limited time periods with higher specific activity levels accommodates short-term iodine spikes which may be associated with power level changes, and is based on the fact that a steam line failure during these short time periods is considerably less likely. Operation at the higher activity levels, therefore, is restricted to a small fraction of the unit's total operating time. The upper limit of coolant iodine concentration during short-term iodine spikes ensures that the thyriod dose from a steam line failure will not exceed 10 CFR Part 100 dose guidelines.

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible, if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should the steam line rupture occur. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and start-up and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.2 of the FSAR. During start-up and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermally induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.