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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. DPR-71

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated September 6, 1985, 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. 94 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 30, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 94

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.

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INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The Emergency Core Cooling System (ECCS) actuation instrumentation shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable and place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.3.1 EACH ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.3.3 The ECCS RESPONSE TIME of each ECCS function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS function.

TABLE 3.3.5.1-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS, SENSOR LOCATIONS AND INSTRUMENT NUMBER	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Time-History Accelographs		
a. Reactor Building +89' 4" level on Drywell (ENV-XT-823-2)	0-1.0g	1
b. Reactor Building - 17' level (ENV-XT-823-1 and ENV-XT-823-3)	0-1.0g	1

TABLE 4.3.5.1-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS, SENSOR LOCATIONS, AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelographs			
a. Reactor Building +89' 4" level on Drywell (ENV-XT-823-2)	M*	SA	R
b. Reactor Building - 17' level (ENV-XT-823-1 and ENV-XT-823-3)	M*	SA	R

*Except seismic trigger

TABLE 3.3.5.6-1

CHLORIDE INTRUSION MONITORS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS[(a)]
1. Chloride leak detectors in the condenser hotwell outlet headers (CO-CR24)	4
2. Chloride leak detector in the condensate pump discharge (CO-CIS-3075-1 or TS-CR-863)	1
3. Chloride leak detector in the inlet to the condensate filter demineralizer (CFD-CIT-1)	1
4. Chloride leak detector in the inlet to the deep bed demineralizer (CDD-CIT-1)	1

a. Chloride intrusion can be detected if any of the functional units have their required minimum number of channels OPERABLE.

INSTRUMENTATION

3/4.3.7 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.7-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.7-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 113 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.7-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.7-1.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.1-1.

4.3.7.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant recirculation loops shall be in operation with the cross-tie valve closed, the pump discharge valves OPERABLE, and the pump discharge bypass valves OPERABLE or closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With one or both recirculation loops not in operation, operation may continue; restore both loops to operation within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each COLD SHUTDOWN which exceeds 48 hours, if not performed in the previous 31 days.

4.4.1.1.2 Each pump discharge bypass valve, if not OPERABLE, shall be verified to be closed at least once per 31 days.

*See Special Test Exception 3.10.4.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the primary containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 49 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.5 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations of primary containment internal pressure ensure that the containment peak pressure of 49 psig does not exceed the design pressure of 62 psig during LOCA conditions. The limit of 1.75 psig for initial positive containment pressure will limit the total pressure to 49 psig, which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.6 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 300°F during LOCA conditions and is consistent with the accident analyses.

4.7.7.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume is at least the minimum specified.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each pump and operating it for at least 15 minutes.

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

d. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

e. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each pump develops at least 2000 gpm at a system head of 125 psig,

   2. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3. Verifying that each fire pump starts sequentially to maintain the fire suppression water system pressure greater than or equal to 125 psig.

f. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

UNITED STATES
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WASHINGTON, D. C. 20555

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
License No. DPR-62

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

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated September 6, 1984, 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. 119, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 30, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 119

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.

Pages

3/4	1-5
3/4	3-39
3/4	3-45
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3/4	4-1
B3/4	6-2
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REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 6, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7.0 seconds, operation may continue and the provisions of Specification 3.0.4 are not applicable provided that:

a. The control rod with the slow insertion time is declared inoperable,

b. The requirements of Specification 3.1.3.1 are satisfied, and

c. The Surveillance Requirements of Specification 4.1.3.2.c are performed at least once per 92 days when operation is continued with three or more control rods with slow scram insertion times;

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

INSTRUMENTATION

3/4.3.4 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4 The control rod withdrawal block instrumentation shown in Table 3.3.4-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4-2.

APPLICABILITY: As shown in Table 3.3.4-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4-2, declare the channel inoperable until the channel is restored to OPERABLE status with its Trip Setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE channels not satisfied for one trip system, POWER OPERATION may continue provided that either:

   1. The inoperable channel(s) is restored to OPERABLE status within 24 hours, or

   2. The redundant trip system is demonstrated OPERABLE within 4 hours and at least once per 24 hours until the inoperable channel is restored to OPERABLE status, and the inoperable channel is restored to OPERABLE status within 7 days, or

   3. For the Rod Block Monitor only, THERMAL POWER is limited such that the MCPR will remain above 1.07, assuming a single error that results in complete withdrawal of any single control rod that is capable of withdrawal.

   4. Otherwise, place at least one trip system in the tripped condition within the next hour.

c. With the requirements for the minimum number of OPERABLE channels not satisfied for both trip systems, place at least one trip system in the tripped condition within one hour.

d. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.4 Each of the above required control rod withdrawal block instrumentation channels shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK, CHANNEL CALIBRATION, and a CHANNEL FUNCTIONAL TEST during the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.4-1.

TABLE 3.3.5.1-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS, SENSOR LOCATIONS, AND INSTRUMENT NUMBER	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Peak Shock Recorders		
a. Reactor Building -17' level (ENV-XRH-823-1)	2-25 Hz	1
b. Reactor Building +20' level (ENV-XRH-823-2)	2-25 Hz	1
c. Reactor Building +117' level (ENV-XRH-823-3)	2-25 Hz	1
2. Triaxial Time-History Accelographs		
a. Reactor Building +89' 4" level on Drywell (ENV-XT-823-2)	0-1.0g	1
b. Reactor Building -17' level (ENV-XT-823-1 and ENV-XT-823-3)	0-1.0g	1

TABLE 4.3.5.1-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS, SENSOR LOCATIONS, AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Peak Shock Recorders			
a. Reactor Building -17' level (ENV-XRH-823-1)	NA	NA	R
b. Reactor Building +20' level (ENV-XRH-823-2)	NA	NA	R
c. Reactor Building +117' level (ENV-XRH-823-3)	NA	NA	R
2. Triaxial Time-History Accelographs			
a. Reactor Building +89' 4" level on Drywell (ENV-XT-823-2)	M*	SA	R
b. Reactor Building -17' level (ENV-XT-823-1 and ENV-XT-823-3	M*	SA	R

*Except seismic trigger

TABLE 3.3.5.6-1

CHLORIDE INTRUSION MONITORS

FUNCTIONAL UNIT AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE CHANNELS[(a)]
1. Chloride leak detectors in the condenser hotwell outlet headers (CO-CR24)	4
2. Chloride leak detector in the condensate pump discharge (CO-CIS-3075-1 or TS-CR-863)	1
3. Chloride leak detector in the inlet to the condensate filter demineralizer (CFD-CIT-1)	1
4. Chloride leak detector in the inlet to the deep bed demineralizer (CDD-CIT-1)	1

a. Chloride intrusion can be detected if any of the functional units have their required minimum number of channels OPERABLE.

INSTRUMENTATION

3/4.3.6 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUEMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation trip systems shown in Table 3.3.6.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6.1-2, declare the trip system inoperable until the trip system is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the Minimum Number of OPERABLE Trip Systems per Operating Pump not satisfied for one Trip Function, restore the inoperable trip system to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.6.1.1 Each ATWS recirculation pump trip system instrumentation trip system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.1.1-1.

4.3.6.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant recirculation loops shall be in operation with the cross-tie valve closed, the pump discharge valves OPERABLE, and the pump discharge bypass valves OPERABLE, or closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With one or both recirculation loops not in operation, operation may continue; restore both loops to operation within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve and bypass valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each COLD SHUTDOWN which exceeds 48 hours, if not performed in the previous 31 days.

4.4.1.1.2 Each pump discharge bypass valve, if not OPERABLE, shall be verified to be closed at least once per 31 days.

*See Special Test Exception 3.10.4.

CONTAINMENT SYSTEMS

BASES

3/4.6.1.4 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the primary containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 49 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.5 PRIMARY CONTAINMENT INTERNAL PRESSURE

The limitations of primary containment internal pressure ensure that the containment peak pressure of 49 psig does not exceed the design pressure of 62 psig during LOCA conditions. The limit of 1.75 psig, for initial positive containment pressure will limit the total pressure to 49 psig, which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.6 PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 300°F during LOCA conditions and is consistent with the accident analyses.

TABLE 3.7.7.4-1 (Continued)

FIRE HOSE STATIONS

LOCATION	ELEVATION	HOSE RACK#
Diesel Generator Building	2'	DGB-1
	2'	DGB-2
	2'	DGB-3
	23'	DGB-4
	23'	DGB-5
	23'	DGB-6
	23'	DGB-7
	23'	DGB-8
	23'	DGB-9
	50'	DGB-10
	50'	DGB-11
	50'	DGB-12
	50'	DGB-13
	50'	AFFF HR-2
	50'	AFFF HR-3
Service Water Building	4'	SW-1
	20'	SW-2
	20'	SW-3
Control Building	23'	2-CB-1
	49'	2-CB-2
	70'	2-CB-3
Diesel Generator Tank Area	NA	AFFF HR-1