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. 168
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), December 31, 1992, as supplemented June 10, 1993, and August 23, 1993, and the request dated December 8, 1993, comply 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. 168 , 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

SBajwa

S. Singh Bajwa, Acting 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: February 8, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 168

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
1-2	1-2
3/4 3-29	3/4 3-29
3/4 9-5	3/4 9-5
B 3/4 9-1	B 3/4 9-1

CHANNEL FUNCTIONAL TEST (Continued)

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel.

Movement of source range monitors, local power range monitors, intermediate range monitors, traversing in-core probes, or special moveable detectors (including undervessel replacement) is not considered a CORE ALTERATION.

In addition, control rod movement with other than the normal control rod drive is not considered a CORE ALTERATION provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.3.1, 6.9.3.2, 6.9.3.3, and 6.9.3.4. Plant operation within these core operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in an assembly which is calculated, by application of an NRC approved CPR correlation, to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be concentration of I-131, μCi/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The following is defined equivalent to 1 μCi of I-131 as determined from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites": I-132, 28 μCi; I-133, 3.7 μCi; I-134, 59 μCi; I-135, 12 μCi.

$\bar{E}$ -AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 15 minutes making up at least 95% of the total non-iodine activity in the coolant.

TABLE 4.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
4. CORE STANDBY COOLING SYSTEMS ISOLATION				
a. High Pressure Coolant Injection System Isolation				
1. HPCI Steam Line Flow - High				
Transmitter:	NA[(a)]	NA	R[(b)]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3
2. HPCI Steam Line High Flow Time Delay Relay	NA	R	R	1, 2, 3
3. HPCI Steam Supply Pressure - Low	NA	M	R	1, 2, 3
4. HPCI Steam Line Tunnel Temperature - High	NA	SA	R	1, 2, 3
5. Bus Power Monitor	NA	R	NA	1, 2, 3
6. HPCI Turbine Exhaust Diaphragm Pressure - High	NA	M	Q	1, 2, 3
7. HPCI Steam Line Ambient Temperature - High	NA	SA	R	1, 2, 3
8. HPCI Steam Line Area Δ Temperature - High	NA	SA	R	1, 2, 3
9. HPCI Equipment Area Temperature - High	NA	SA	R	1, 2, 3
10. Drywell Pressure - High				
Transmitter:	NA[(a)]	NA	R[(b)]	1, 2, 3
Trip Logic:	D	M	M	1, 2, 3

3/4.9.3 CONTROL ROD POSITION

LIMITING CONDITION FOR OPERATION

3.9.3 All control rods shall be fully inserted*.

APPLICABILITY: OPERATIONAL CONDITION 5, during loading of fuel assemblies into the core**.

ACTION:

With all control rods not fully inserted, immediately suspend loading of fuel assemblies into the core. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 Verify all control rods to be fully inserted within 2 hours prior to the start of and at least once per 12 hours during loading of fuel assemblies into the core.

*Except control rods removed per Specification 3.9.10.1 or 3.9.10.2.
**See Special Test Exception 3.10.3.

3/4.9.1 REACTOR MODE SWITCH

Locking the reactor mode switch in the refuel position ensures that the restrictions on rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals, fuel assemblies and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

During a SPIRAL UNLOAD, the count rate of the SRM will decrease below 3 cps before all of the fuel is unloaded. The count rate of 3 cps is not necessary since there will be no reactivity additions during the spiral unload. The SRMs will be required to be OPERABLE prior to the SPIRAL UNLOAD, and each SRM will be verified operational by raising the count rate to 3 cps prior to the SPIRAL RELOAD by inserting up to four fuel assemblies around each SRM. This will ensure that the SRMs can be relied upon to monitor core reactivity during the reload.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during loading of fuel assemblies into the core ensures that fuel will not be loaded into a cell without a control rod and prevents two positive reactivity changes from occurring simultaneously.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 199
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), December 31, 1992, as supplemented June 10, 1993, and August 23, 1993, and the request dated December 8, 1993, comply 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. 199, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Bajwa

S. Singh Bajwa, Acting 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: February 8, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 199

FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

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Remove Pages	Insert Pages
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CHANNEL FUNCTIONAL TEST (Continued)

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel.

Movement of source range monitors, local power range monitors, intermediate range monitors, traversing in-core probes, or special moveable detectors (including undervessel replacement) is not considered a CORE ALTERATION.

In addition, control rod movement with other than the normal control rod drive is not considered a CORE ALTERATION provided there are no fuel assemblies in the associated core cell. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specifications 6.9.3.1, 6.9.3.2, 6.9.3.3, and 6.9.3.4. Plant operation within these core operating limits is addressed in individual specifications.

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3. HPCI Steam Supply Pressure - Low	NA	M	R	1, 2, 3
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